Technical Aspects of ALWR Emergency Planning

TR-113509

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REPORT SUMMARY

The very high level of safety achieved by the U.S. advanced light water reactor (ALWR) designs merits emergency planning rulemaking that is cost-effectively tailored to ALWRs. This report defines and evaluates the technical basis for such rulemaking and defines a supportable ALWR emergency planning concept.

Background

EPRI's ALWR Utility Requirements Document (URD) defines both broad and detailed requirements to be met in the design of ALWRs, including comprehensive requirements to assure a very high level of safety. All three U.S. ALWR designs achieve a very high degree of compliance with the detailed URD requirements, and all three designs have now received the formal design approval of the U.S. Nuclear Regulatory Commission (NRC). In particular, the designs are intended to meet specific URD criteria relevant to emergency planning, such as in the areas of core damage prevention, containment performance, and offsite dose.

Objectives

To quantify the performance of the three U.S. ALWR designs, in the areas of core damage prevention, containment performance, and offsite dose, and to use the results to define an emergency planning concept cost-effectively tailored to ALWRs.

Approach

The objectives are achieved in this report via essentially three steps. The first identifies the URD criteria particularly relevant to emergency planning and compares the ALWR designs with those criteria, thereby confirming detailed conformance. The second quantifies the performance of the ALWR designs, in the areas of core damage prevention, containment performance, and offsite dose, and compares that performance with emergency planning criteria currently applied to existing U.S. nuclear plants. The third step defines a cost-effective, ALWR-specific emergency planning concept in which the ALWRs perform as well as currently required of existing plants in their emergency planning concept.

Results

All three ALWR designs conform to the relevant detailed URD criteria. As a result, core damage accident sequences are extremely unlikely, and even if such sequences occur, the releases will be small and slow.

The ALWRs' probabilities of exceeding various doses at 0.5 miles are lower than the probabilities assessed for existing plants at 10 miles in emergency planning criteria currently applied to existing plants.

An appropriate ALWR emergency planning concept has two concentric areas: a response area, for rapid response, out to 0.5 miles, and an awareness area, for assuring public awareness, beyond 0.5 miles.

EPRI Perspective

This report not only contains the details of the approach and results summarized above, but also reflects extensive interaction with the U.S. nuclear utility community (on the URD and emergency planning elements) and with the U.S. reactor designers (on the ALWR design conformance and performance aspects). The report has also been reviewed by and discussed with a small team of very experienced peers, and incorporates the recommendations of that team.

The resulting report should prove to be the thorough technical support needed by the nuclear utility community to gain NRC approval of a cost-effective, ALWR-specific emergency planning concept.

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Keywords

Advanced light water reactors
Emergency planning

ABSTRACT

This report evaluates the technical aspects of a proposed rule being developed by industry to address advanced light water reactor (ALWR) emergency planning, including the technical basis and the ALWR emergency planning concept which builds on this technical basis. The foundation of the technical basis is the ALWR Utility Requirements Document (URD). This report evaluates the URD provisions on accident prevention and mitigation, including specific design criteria for emergency planning. In addition, the report evaluates and compares the ALWR in a generic manner against NUREG 0396 considerations. Finally, an emergency planning concept has been developed for the ALWR which is consistent with and supported by the technical basis.

The results of the above work are as follows:

- Detailed design criteria have been provided in the URD to support improved emergency planning for ALWRs. The criteria include deterministic requirements (containment design features and associated accident analyses) intended to address specific severe accident challenges, as well as a protective action guide (PAG) dose requirement and a PRA requirement. For plant designs meeting these criteria, core damage accident sequences are extremely unlikely, and even if such sequences occur, the releases would be small and slow. All three ALWR designs (AP600, System 80+, and ABWR) satisfy these design criteria.
- The ALWR accident sequence probabilities and source terms are significantly reduced compared to the WASH 1400 accident sequence probabilities and source terms which were used as the basis for the 1978 NUREG 0396 evaluation. The table below summarizes the ALWR conditional probabilities of exceeding various doses at 0.5 miles against the corresponding conditional probabilities from NUREG-0396, Figure I-11 at 10 miles.

ALWR NUREG-0396 Dose Assessment Results

	ALWR (0.5 mile)	**NUREG-0396, Fig I-11 (10 miles)**
Conditional prob. of exceeding 1 rem	0.25	0.3
Conditional prob. of exceeding 5 rem	0.06	0.25
Conditional prob. of exceeding 50 rem	0.006	0.1
Conditional prob. of exceeding 200 rem	<0.001	0.01 – 0.001

- An ALWR emergency planning concept has been developed which is based on showing that the URD emergency planning design criteria are met and that the NUREG-0396 considerations are satisfied. The plan concept is for two concentric zones or areas: a response area, for rapid response, out to 0.5 miles, and an awareness area, for assuring public awareness, beyond 0.5 miles. The response area is the responsibility of the licensee. Awareness area activities would be the responsibility of the offsite agencies and would be administered as part of the all-hazards emergency plan required by the Federal Emergency Management Agency (FEMA). The table below provides a summary of the ALWR emergency planning concept and a comparison with existing emergency planning.

Summary of ALWR Emergency Planning

	ALWR Emergency Planning	Existing Emergency Planning
Plume exposure pathway planning distance	0.5 mile distance based on meeting 1 rem PAG and no early injuries	10 mile distance based on meeting 1 rem PAG and no early injuries
Planning basis for expansion of response	Onsite plan and all-hazards plan	Onsite plan and offsite plan
Ingestion exposure pathway planning area	25 mile distance based on meeting limiting ingestion PAG	50 mile distance based on meeting limiting ingestion PAG

- The planning basis for expansion of response for ALWR emergency planning is the fact that the response area (onsite) planning and awareness area (offsite) planning, as well as the implementation actions taken in an actual emergency, would facilitate protective actions beyond the plume exposure pathway planning distance boundary (i.e., the 0.5 mile response area), and in fact even beyond the awareness area boundary, should such actions be necessary.

- For the ingestion exposure pathway, existing emergency planning, which is based on NUREG-0396 dose calculations using WASH 1400 sequence probabilities and source terms, meets the dose limits at about 50 miles. ALWR ingestion exposure pathway emergency planning, which is based on dose calculations which use ALWR sequence probabilities and source terms, meets the ingestion exposure pathway dose limits at about 25 miles.

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1 INTRODUCTION

This report discusses the technical aspects of the proposed rule for ALWR emergency planning, including the technical basis and the emergency planning concept which builds on this technical basis. The foundation of the technical basis is the ALWR Utility Requirements Document (URD) [1] provisions on accident prevention and mitigation, including specific design criteria for emergency planning. The URD design requirements are discussed in Section 2. To establish the capabilities of the three ALWR designs, each design is assessed against these design criteria as discussed in Section 3. In addition, the ALWR plume exposure pathway has been evaluated in a generic manner against NUREG 0396 [2] considerations in Section 4.

The ALWR emergency planning concept has been developed to be fully consistent with and supported by the technical basis. The plan concept starts with four general principles which industry has defined for ALWR emergency planning. These general principles are then used as guidance in characterizing the plan at a conceptual level. This is discussed in Section 5.

The ALWR emergency planning basis is summarized, and compared with that of existing emergency planning, in Section 6.

A number of appendices have been included with details and supporting information. Appendix A reproduces relevant URD sections which are discussed in Section 2. Appendices B, C, and D provide supporting information for the Section 3 assessment of the three ALWR designs against the URD design requirements. Appendix E provides details of the assessment of ingestion pathway exposure for ALWRs. Appendix F provides information on severe accident management guidelines applicable to ALWRs. Appendix G documents an analysis of fission product aerosol retention for unisolated steam generator tube rupture accidents. Appendix H is a summary of the industry response to recommendations on this report from a peer review committee.

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ALWR URD DESIGN REQUIREMENTS

The first main element of the technical basis for ALWR emergency planning is the URD requirements. The URD contains a broad array of policies and principles, as well as specific design requirements, to produce ALWR designs which are reliable, economical, and safe. With respect to emergency planning the URD establishes specific design criteria, and associated methodology for demonstrating that the design criteria have been met, in the areas of containment performance and offsite dose. In addition, a supplemental PRA evaluation is required by the URD in support of the demonstration of the criteria. Together, the design criteria and supplemental PRA evaluation form the technical foundation for emergency planning for the ALWR. While they would likely be met for the ALWR designs apart from emergency planning considerations, the design criteria have been specified and assembled in the emergency planning context in order to clearly articulate this technical foundation.

It should be noted that while the emergency planning requirements focus mainly on containment and accident mitigation capability, core damage prevention is key to overall plant safety and is an important part of the technical foundation for ALWR emergency planning. Core damage prevention of the ALWR is rooted in the URD emphasis on simplicity, engineering margin, and human factors throughout the design process. Examples of requirements in these areas include:

- No recirculation lines in the BWR
- Improved PWR reactor coolant pump design which reduces the likelihood of pump seal loss of coolant accident (LOCA)
- No loop seals and a minimal number of welds in passive PWR primary system piping
- Increased thermal margin in the fuel (15% above regulatory limits)
- Reduced PWR primary system hot leg temperature of 600°F or less to decrease steam generator tube corrosion
- Use of improved alloys for PWR steam generator tube materials to reduce tube failures
- Improved resistance to embrittlement in the reactor vessel
- Increased reactor coolant system (RCS) coolant inventory which delays core uncovery in the event of an accident
- Improved redundancy and reliability in safety injection, emergency feedwater, and plant cooling water systems
- Decreased dependence on operator action after an accident
- Improved control room which makes the plant easier and safer to operate

- Improved accessibility for maintenance
- Decreased dependence of safety systems on support systems
- Provision of an alternate ac power source

In addition, there are requirements specifically directed toward avoiding core uncovery during shutdown conditions. The ALWR Program reviewed existing shutdown risk issues and the URD provisions to address these issues [3]. Additional requirements were defined as a result of this review. With proper plant specific implementation of these requirements and appropriate administrative controls and procedures provided by the licensee and operator, core uncovery during shutdown conditions will not be a credible event.

Finally, accident management requirements exist to prevent as well as limit the extent of core damage. Equipment and procedures for accident management are being considered as part of the plant design process, thus increasing the likelihood of successful recovery actions.

In summary, while the URD emergency planning design criteria focus on containment and accident mitigation matters, the ALWR emphasis on core damage prevention and the resulting extremely low probability of an accident are important factors in the consideration of emergency planning requirements. Further discussion of accident sequence probability as an input to the emergency planning technical basis is contained in Section 4 on NUREG 0396 considerations.

2.1 URD Design Criteria for Emergency Planning

Design criteria, and associated methodology for demonstrating that the design criteria have been met, have been defined for ALWR emergency planning in the areas of containment performance and offsite dose. The complete set of criteria and methodology are specified in URD Volumes II and III (for evolutionary and passive plants, respectively), Chapter 1 and are reproduced in Appendix A.

A summary of the criteria and the associated methodology is as follows:

Containment Performance Criterion

Plant design characteristics and features shall be provided to withstand core damage sequence loads and to preclude core damage sequences which could bypass containment. Containment loads from low pressure core damage sequences shall not exceed ASME Service Level C/Unity Factored Load limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours, there shall be no uncontrolled release.

The methodology for demonstrating the containment performance criterion includes the following:

- Incorporate the design characteristics and features specified in the URD to address severe accident challenges.

- Demonstrate using best estimate severe accident methods that the loads associated with core damage sequences are no more limiting than the peak LOCA plus hydrogen loads or ASME Service Level C limits.
- Protection of the containment for overpressurization beyond 24 hours shall be provided. Overpressure protection may be provided by the size and strength of the containment. Protection on the order of two to three days is judged to be adequate for actions by the plant staff to bring the accident under control.

Dose Criterion

Dose at 0.5 mile from the reactor due to fission product source term release from a damaged core shall not exceed the Protective Action Guides (PAGs) for approximately 24 hours.

The methodology for demonstrating the dose criterion includes the use of a physically-based source term using NUREG 1465 parameters [4] and/or plant-specific severe accident results, a probabilistic dose method (e.g., MACCS 1.5), use of a range of meteorological conditions, and use of effective dose equivalent with a 50 year commitment. The PAGs are projected dose levels for evacuation (1 to 5 rem) which are specified by the Environmental Protection Agency in a 1992 report [5] as guidance for actions to protect the public in the early phase of a nuclear incident.

The criteria and associated methodology are primarily deterministic. A supplemental PRA evaluation is also required by the URD in support of the two criteria. This reliance on deterministic criteria with PRA as a supplement is consistent with the NRC Severe Accident Policy [6]. The supporting requirements for the containment performance criterion, the dose criterion, and the supplemental PRA evaluation are described in more detail below in Sections 2.2, 2.3, and 2.4, respectively.

It should be noted that the ALWR emergency planning design criteria are intended to be applied together with the methodology specified in the URD. Thus, for example, it would be inappropriate to require plants to meet 1 rem at 0.5 mile with a more conservative dose evaluation methodology than that in URD Volume III, Chapter 1, Section 2.6.5. Application of the criteria with the specified methodology is considered to provide adequate margin based on the following:

- The bounding nature of the core damage progression and associated fission product release specified in the URD methodology, given any credible severe accident. The fission product release includes that from the gap, early in-vessel, ex-vessel, and late in-vessel core damage progression phases.
- The very low likelihood of any severe accident in an ALWR. Given this extremely low likelihood, conservatism beyond that noted above is considered unwarranted.
- The margin in the 24 hour, PAG dose requirement. The 1 rem is the low end of the U.S. Environmental Protection Agency (EPA) range of 1 to 5 rem for evacuation [5], and 24 hours provides significant margin to perform offsite protective measures.

- A more conservative dose evaluation methodology (90th percentile meteorology) is, in fact, considered with dose limits from ICRP 63 [7] which recommends a dose limit for evacuation no lower than 50 mSv (i.e., 5 rem) under any circumstance.

2.2 Discussion of Containment Performance Requirements

The licensing design basis for the ALWR containment is the traditional set of deterministic loads and load combinations compared against ASME Section III limits. Loads associated with events including loss of coolant accidents and the safe shutdown earthquake are combined in the design of the plant. Further, the licensing design basis includes loads associated with generation of hydrogen in accordance with 10CFR50.34(f) [8].

In addition to the licensing design basis, the URD includes the safety margin basis which contains requirements that provide margin beyond the licensing design basis. The safety margin basis specifies severe accident requirements which support the emergency planning containment performance design criterion defined above. These URD severe accident design requirements were developed through the cooperative efforts of the ALWR plant designers and other ALWR Program organizations using knowledge gained over the last 15 to 20 years from work on severe accident matters. To provide confidence in the completeness and effectiveness of the severe accident design requirements, the requirements were evaluated against a comprehensive set of severe accident challenges. The set of potential severe accident challenges was identified based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. Table 2-1 contains a list of these potential challenges. There are 23 challenges in the table. The first 13 challenges represent events which could occur independent of or precede core damage, such as bypass accidents. The remaining 10 challenges could occur as a result of a severe accident, such as containment pressure loads from a core damage event.

Evaluations of the URD were performed to assess the degree to which each of the 23 potential severe accident challenges was addressed in the evolutionary and passive plant requirements. These evaluations contain a challenge-by-challenge assessment of the requirements. Reference [9] documents this assessment for passive plants. A similar assessment was performed for evolutionary plants. Tables A.5-1 and A.5-2 in Section A.5 of Appendix A summarize the results of this challenge-by-challenge assessment for challenges which could occur independent of or precede core damage, and challenges which could occur as a result of a severe accident, respectively.

Table 2-1
Potential Severe Accident Containment Challenges

CHALLENGES/FAILURE MODES THAT ARE INDEPENDENT OF OR COINCIDENT WITH A SEVERE ACCIDENT
1. Containment Isolation
2. Interfacing System LOCA
3. Blowdown Forces
4. Pipe Whip and Jet Impingement
5. Steam Generator Tube Rupture (PWR)
6. Anticipated Transient Without Scram (ATWS)
7. Suppression Pool Bypass (BWR)
8. Reactor Pressure Vessel (RPV) Failure
9. Internal Vacuum
10. Internal (Plant) Missiles
11. Tornado and Tornado Missiles
12. Man-Made Site Proximity Hazards
13. Seismic
CHALLENGES/FAILURE MODES POTENTIALLY RESULTING FROM A SEVERE ACCIDENT
14. High Pressure Melt Ejection (HPME)
15. Hydrogen Detonation/Deflagration
16. In-vessel Debris-Water Interaction
17. Ex-vessel Debris-Water Interaction
18. Noncondensable Gas Generation During Core-Concrete Interaction
19. Containment Basemat Erosion or Reactor Pressure Vessel Support Degradation During Core-Concrete Interaction
20. Core Debris in Containment Sump
21. Core Debris Contact with Containment Shell Liner
22. Decay Heat Generation
23. Steam Generator Tube Rupture (SGTR) from Natural Circulation of Hot Gases (PWR)

It is evident from the Table A.5-1 and A.5-2 results that potential severe accident challenges, regardless of the extremely low likelihood of the challenge, have been systematically and explicitly addressed in the URD. This includes challenges which could pose an early threat to containment integrity (i.e., the first group of 13 challenges, which are the containment bypass type challenges, as well as high pressure melt ejection, hydrogen detonation, steam explosion, basemat erosion or pressure vessel support degradation, core debris contact with shell liner, and steam generator tube rupture from hot gases). For example, interfacing system LOCA and hydrogen phenomena, which on the basis of past PRAs have been considered significant threats to early containment integrity, are essentially precluded by the design requirements specified in the URD. Interfacing system LOCA requirements include increased design pressure of the interfacing low pressure systems to preclude rupture in the event that these systems are exposed to full RCS pressure. Hydrogen requirements include hydrogen control systems to prevent or mitigate deflagration and detonation loads such that these loads would not threaten containment. Additional discussion on the manner in which ALWR requirements have addressed accident sequences and containment failure modes considered important in past PRAs is provided below.

An additional factor relative to containment challenges is that, even if it was assumed that containment systems do not perform as designed, the plant operators have the ability to perform accident management actions to assure containment integrity. An example in this regard is containment isolation. Accident management procedures have been developed and implemented in operating plants to address containment isolation as follows [10, 11]:

- Confirmation of containment isolation. In the event of a containment isolation signal, emergency operations and/or alarm response procedures call for the operator to confirm that containment isolation valves have closed using valve position indications in the control room. For the ALWR, sufficient time (minimum of 1 to 2 hours) is expected to be available for the operator to perform any necessary valve closures before significant release of radioactivity into the containment.
- Continuous survey of radiation in key plant areas, providing indication of the existence and location of non-isolated or leaking lines. Monitoring systems have been designed for areas such as building ventilation stacks, sampling lines, and sumps such that if excessive leakage begins to occur, it can be detected immediately.
- In case of leakage, complementary confirmation of containment isolation including local verifications and/or operator actions when necessary.

Generally, it is considered that a relatively small, well-trained team of plant personnel can be effective in accident management for containment isolation as well as other containment challenges. As noted in Section 2.1 above, the ALWR URD specifies that accident management equipment and procedures be developed as part of the design process. Appendix F provides further information.

On the basis that challenges which pose an early threat to containment integrity are being addressed in the design, and considering the extremely low likelihood of core damage in the first place as well as the capability of accident management to address problems, it is expected that accident sequences involving early containment failure will not be credible in ALWR designs which meet the URD requirements. This has implications on the type of severe accident sequences for which containment loads should be evaluated against the Service Level C/Unity Factored Load limits as specified in the emergency planning containment performance design criterion. It also has implications on the range of severe accident sequences which should be considered in the ALWR NUREG 0396 assessment in Section 4.

The remaining Table 2-1 challenges (i.e., hydrogen plus LOCA loads, pressurization from debris-water interactions, the potential for core concrete interaction, and decay heat loads) should be considered in establishing the accident sequences for which containment response should be evaluated. In considering these remaining challenges, the effect of plant design characteristics and features on the containment loads and fission product release should be included. For example, passive containment heat removal does not depend upon any electrical or mechanical equipment which must function in a severe accident environment. Thus it is reasonable to assume that passive containment heat removal functions as designed during the accident. Similarly, it is reasonable to assume that an evolutionary PWR containment, which is large enough to withstand many hours of decay heat and noncondensible gas generation without failure, will have a significant time delay before any fission product release.

On this basis, the types of severe accident sequences for which containment response should be evaluated against the Service Level C/Unity Factored Load limits are as follows:

Core Damage

- Rapid core damage progression, i.e., beginning at approximately one hour after the initiating event, and occurring over a time frame of a few hours
- Large scale core melt and associated gas and aerosol release
- Steam release out of phase with aerosol release
- Consideration of in-vessel core damage and the potential for ex-vessel core damage

Reactor Coolant System Condition

- Limited aerosol plateout in the RCS
- A vapor pathway exists in the RCS (i.e., from the core to the containment atmosphere)
- RCS is depressurized to about 100 psig or less

Containment Condition

- Containment is isolated and otherwise intact at the time of core damage (i.e., no containment bypass has occurred)
- Water exists in the reactor cavity/lower drywell prior to or immediately upon reactor vessel lower head penetration
- Containment systems are functioning as designed (heat removal, fission product removal, hydrogen control, pH control)
- Containment leaking at design basis leak rate (or leak rate proportional to pressure)

Secondary Building Condition

- Containment leakage released into secondary building volume
- Building volume mixing and exchange with the environment is based upon plant design characteristics (e.g., safety envelope leakage)
- Building volume bypass pathways taken into account‘

As noted in Appendix A, the above severe accident sequence types are specified in URD Chapter 5, Section 2.6, Criteria and Methodology for ALWR Emergency Planning. The loads associated with these severe accident characteristics must not exceed specified ASME limits for approximately 24 hours.

ASME Service Level C/Unity Factored Load limits were specified in order to provide high confidence that containment leakage would, at most, be a linear extrapolation of design basis leakage. This is based on several factors including:

- Service Level C assures stress levels below yield in steel containments, and unity factored load assures limits on linear deformation in concrete containments; leaks are not expected in membranes with such small deformations.
- A review of experimental and analytical evidence [12] which indicates that there is essentially no increase in penetration leakage under severe accident conditions up to Service Level C/Unity Factored Loads.
- Nuclear plant containment leak test data indicating that, for pressure increases up to design pressure, leak rate does not exceed a value proportional to the pressure [12].

An additional point is the fact that the fission product mass is almost exclusively particulate and as noted in reference [13], aerosol plugging of leak paths is expected which should significantly reduce the actual mass leaked during an accident compared to that assumed in design basis leakage.

The 24 hour limit is consistent with the 24-hour limit specified in the dose criterion and allows appropriate time for necessary offsite protective actions.

No uncontrolled release beyond 24 hours has been specified to provide additional margin for emergency planning. While approximately 24 hours is considered more than adequate for offsite protective actions, it is desirable to avoid long-term overpressure failure.

Reference [14] contains further information on the URD containment performance requirements.

<u>Probabilistic Perspective for Containment Performance Requirements</u> To provide additional confidence that the appropriate severe accident sequence characteristics are being considered in the evaluation of containment response against the Service level C/Unity Factored Load limits, the URD requires that accident sequence types with frequency greater than approximately 10^{-7} per year be evaluated for containment response. Lower frequency functional sequence types are to be reported for discussion (i.e., identification of design characteristics and features which are credited in reaching this low frequency), but are not required to be evaluated for containment response. Further discussion of the 10^{-7} per year frequency is contained in Section 4 below.

As described in Appendices B, C, and D, review of the ALWR plant specific PRAs indicates that accident sequences which are of the order of 10^{-7} per year or greater involve core damage into an intact containment with the reactor coolant system at least partially depressurized and containment systems functioning as designed. That is, the characteristics of these sequences from the PRAs are similar to the characteristics which are defined above from a primarily deterministic perspective.

ALWR Risk Characterization vs. WASH 1400 Given the above ALWR design requirements, it is useful at this point to examine the accident types and failure modes which dominated the risk in the WASH 1400 PRA [15] and the manner in which these sequence types and failure modes are addressed by the ALWR design. At the time of the development of the existing emergency planning basis, defined in NUREG 0396, WASH 1400 provided the most detailed perspective on the types of accident scenarios which made up the collection of "Class 9" events. Accident scenario types and containment failure modes which dominated the risk in WASH 1400 are summarized in Table 2-2, and it is these events which formed the basis for existing emergency planning requirements. Also included in Table 2-2 are important challenges identified as a result of PRA work subsequent to WASH 1400. More recently, improved understanding of severe accident behavior as well as modifications to plants and procedures have changed the characteristics of accident scenarios which dominate risk compared to WASH 1400. This applies to a significant extent in existing plants and to an even greater extent in ALWRs. ALWR design requirements directly address those events which dominated the risk in WASH 1400 and subsequent PRAs. Appendix A, Section A.6 describes the ALWR design characteristics and features that have been provided to preclude or accommodate the accident sequence types and failure modes listed in Table 2-2 as contributors to core damage and containment failure.

It is apparent from this reexamination that the Table 2-2 WASH 1400 issues which dominated the risk and formed the basis for existing emergency planning, as well as subsequently identified containment challenges (shown in Table 2-2 with a footnote), have been addressed explicitly in the ALWR requirements. Therefore, the characterization of risk for ALWRs will differ significantly from a WASH 1400 type characterization, or even from the characterization in subsequent PRAs. Table 2-3 provides clear illustration of this difference in risk characterization. The probability of dose exceedance for the ALWR at 0.5 mile is one to two orders of magnitude less than the probability of dose exceedance for WASH 1400 at 10 miles. This is the case for 1 rem and for the dose at which significant early injuries start to occur. This ALWR risk characterization, which reflects the above design characteristics and features and the improved phenomenological understanding of severe accidents, should be used in formulating ALWR emergency planning regulatory requirements.

2.3 Discussion of Dose Requirements

As part of the technical basis for emergency planning in ALWRs, a dose limit is required. There are several aspects of the dose criterion in Section 2.1 above to be discussed.

- The 0.5 mile distance is specified for the dose calculation; this corresponds to the distance of the site boundary from the reactor which is assumed in the URD.
- The physically-based source term utilizes release parameters from NUREG 1465 and/or plant specific severe accident evaluation results. A physically-based source term has been used for ALWR licensing (DBA) applications and is also proposed here for emergency planning applications in the ALWR. The physically-based source term specifies fission product release timing and magnitude to containment,

Table 2-2
Accident Sequence Types Which Tend to Dominate Risk for Existing Emergency Planning Basis

DOMINANT ACCIDENT SEQUENCES LEADING TO CORE DAMAGE [1]	
PWRs	**BWRs**
LOCAs (large or small) - Loss of injection (AD, SD) - Loss of recirculation (AH, SH) Vessel Rupture (R) Interfacing LOCA (V)	LOCAs (large or small) - Loss of injection (AE, SE) Vessel Rupture (R)
Transients - Loss of secondary heat removal (TML) - Station blackout (TMLB′) ATWS (TKQ)	Transients - Loss of containment heat removal (TW) - Loss of all injection (TQUV) ATWS (TC)
Shutdown Conditions [2]	Shutdown Conditions [2]

Table 2-2 (continued)
Accident Sequence Types Which Tend to Dominate Risk for Existing Emergency Planning Basis

POTENTIAL CONTAINMENT FAILURE MODES [1]	
PWRs	**BWRs**
Overpressure (δ) In-Vessel Steam Explosion (α) Hydrogen Combustion (γ) Containment Isolation (β) Basemat Penetration (ε) Direct Containment Heating [2] Steam Generator Tube Rupture [2]	Overpressure (γ) In-vessel Steam Explosion (α) Ex-Vessel Steam Explosion (β) Containment Isolation (δ,ε) Liner Melt-Through [2] Overtemperature [2]

(1) Characters in parentheses are sequence and failure mode designators from WASH 1400
(2) Issues which were identified in PRA work subsequent to WASH 1400.

Table 2-3
Comparison Between WASH-1400 and ALWR Requirements

	Mean Core Damage Frequency	Probability of Exceeding 1 Rem	Probability of Exceeding Early Injury Dose
WASH-1400 (doses at 10 miles from reactor)	~$1.5x10^{-4}$/yr	$4x10^{-5}$/yr (1)	1×10^{-6}/yr (2)
ALWR Requirements (doses at 0.5 miles from reactor)	<10^{-5}/yr	<10^{-6}/yr	<10^{-7} / yr (3)
ALWR Plant Specific (doses at 0.5 miles from reactor)	~$2x10^{-6}$/yr (4)	~$1x10^{-7}$/yr (4)	~$1x10^{-8}$/yr (4)

(1) Based on mean core damage frequency of ~1.5 $x10^{-4}$/yr (i.e., 3 x the WASH-1400 median value of 5 $x10^{-5}$) and, from Figure I-11 of NUREG-0396, ~0.3 conditional probability of exceeding 1 rem at 10 miles.

(2) Based on mean core damage frequency of ~1.5 $x10^{-4}$/yr and, from Figure I-11 of NUREG-0396, ~0.03 conditional probability of exceeding prompt effects dose at 10 miles.

(3) The URD specifies that functional sequences which could threaten containment must be <10^{-7}/yr.

(4) Based on the average of the three ALWRs using information from the plant specific PRAs.

chemical form of the fission products, fission product removal from containment, and fission product holdup in secondary buildings. The main difference between the DBA application and the emergency planning application of the physically-based source term is the fact that for emergency planning the ex-vessel release and late in-vessel release are considered (whereas for DBA only gap and early in-vessel release are considered). The physically-based source term has been defined so as to generally envelope potential source terms from sequences having the characteristics defined above in Section 2.1. Thus, the physically-based source term provides significant margin beyond the actual fission product release which would be expected if a core melt accident were assumed to occur at an ALWR. Tables 2-4 and 2-5 depict the release fractions from NUREG 1465, and Table 2-6 depicts release fractions based upon ALWR Program work [16, 17] which differ slightly from NUREG 1465. The differences are mainly in the low volatile releases. Since these differences are not expected have a significant effect on offsite dose and since it is expected that the ALWR will have margin to dose limits, NUREG 1465 release parameters will be used, where applicable, for the emergency planning PAG dose calculation.

Table 2-4
NUREG 1465 PWR Release Fractions to Primary Containment1

Nuclide	Gap Release	Early In-Vessel	Ex-Vessel[2]	Late In-Vessel[3]	Total
Duration (hr)	0.5	1.3	2.0	10.0	-
Nobles	0.05	0.95	0	0	1.0
I	0.05	0.35	0.25	0.1	0.75
Cs	0.05	0.25	0.35	0.1	0.75
Te		0.05	0.25	0.005	0.305
Sr, Ba		0.02	0.1	0	0.12
Ru		0.0025	0.0025	0	0.005
Cerium		0.0005	0.005	0	0.0055
Lanthanum		0.0002	0.005	0	0.0052

Table 2-5
NUREG 1465 BWR Release Fractions to Primary Containment1

Nuclide	Gap Release	Early In-Vessel	Ex-Vessel[2]	Late In-Vessel[3]	Total
Duration (hr)	0.5	1.5	3.0	10.0	-
Nobles	0.05	0.95	0	0	1.0
I	0.05	0.25	0.30	0.01	0.61
Cs	0.05	0.20	0.35	0.01	0.61
Te		0.05	0.25	0.005	0.305
Sr, Ba		0.02	0.1	0	0.12
Ru		0.0025	0.0025	0	0.005
Cerium		0.0005	0.005	0	0.0055
Lanthanum		0.0002	0.005	0	0.0052

1. All numbers except durations are fraction of original core fission product inventory released into the containment.
2. The ex-vessel release is from the ex-vessel debris into a water pool overlying the ex-vessel debris since the ALWR uses a flooded cavity design
3. The late in-vessel release is from the fuel remaining in the reactor vessel after lower vessel head meltthrough.

Table 2-6
Alternative Release Fractions Based on ALWR Program Work

Nuclide	Gap Release	Early In-Vessel	Ex- Vessel	Late In-Vessel	Total
Sr, Ba	0	0.004	0.002	0	0.006
Ru	0	0.0025	0.01	0	0.0125
Cerium	0	0.0001	0.001	0	0.0011
Lanthanum	0	0.0001	0.001	0	0.0011

- The PAGs are expressed in the EPA guidance as a range of 1 to 5 rem. It is further stated in the EPA guidance that evacuation should normally be initiated at a projected dose of 1 rem. The NUREG 0396 basis for existing emergency planning depended in part on establishing that "most" core melt accidents would not exceed the PAG. This is taken to be the lower value (i.e., 1 rem) based on the EPA guidance. A similar approach is used here for ALWR emergency planning. There are two sources of variability in determining the meaning of "most" in this situation: the magnitude of the source term, and the meteorology. Median dose (i.e., 50th percentile meteorology) in combination with the physically-based source term, which tends to bound the source term expected for nearly all core melt accidents in an ALWR, assures that the dose from most core melt accidents will not exceed 1 rem.
- More extreme (e.g., very stable atmospheric conditions, low wind speed) meteorology could cause higher doses for a given source term. While doses exceeding 1 rem would not be expected as noted above, a 5 rem limit has been specified for 90th percentile meteorology in order to address more extreme meteorological conditions. A 5 rem limit for such conditions is considered reasonable on several grounds. First, ICRP 63 [7] recommends a dose limit for evacuation no lower than 50 mSv (i.e., 5 rem). Second, under stable, low wind speed conditions the plume is concentrated (only about 100 feet wide at 0.5 mile) and is moving slowly, so the need for rapid evacuation, if any, would be limited to a relatively small sector. Finally, 5 rem is the upper end of the 1 to 5 rem range recommended by EPA and thus is a reasonable limit for emergency planning purposes under low probability weather conditions.
- The methodology specified for the dose evaluation is similar in concept to what is typically done in Level 3 PRA evaluations, e.g., a MACCS 1.5 calculation. The site meteorology which has been specified for ALWR design certification applicant dose calculations is that which is in the ALWR URD Key Assumptions and Groundrules for PRA. This site was selected to have a Chi/Q greater than 80 to 90 percent of U.S. operating nuclear plant sites to provide siting margin and flexibility for the ALWR. Total effective dose equivalent (TEDE) is to be used (as opposed to the older whole body concept) on the basis of the EPA report [5] and revised 10CFR20.
- A period of approximately 24 hours after the beginning of fission product release to containment has been specified on the basis of EPA guidance for actions to protect the public in the early phase of a nuclear incident. As noted in NUREG 1338 [18], based on evacuation experience for non-radiological emergencies, evacuations take from two to eight hours, including time to notify the public. Not exceeding the PAG for approximately 24 hours

would provide significant margin for ALWR accident detection, notification, and, if necessary, evacuation.

2.4 Supplementary PRA Evaluation

As described in Section 2.1 the two ALWR emergency planning criteria, containment performance and offsite dose, stress a deterministic approach. To complement the deterministic approach associated with the criteria, a supporting PRA evaluation has also been specified. The PRA is required to demonstrate that core damage frequency is less than 10^{-5} per year and that the cumulative frequency for sequences that result in greater than 1 rem for 24 hours at 0.5 mile from the reactor is less than 10^{-6} per year. As part of the PRA evaluation, it is also to be demonstrated that the prompt accident quantitative health objective of the NRC Safety Goal Policy [19] is met with no credit for offsite evacuation prior to 24 hours.

The purpose of the PRA evaluation is to demonstrate the integrated effectiveness of the two emergency planning criteria and to serve as a tool for the Plant Designer for refining and optimizing the design. Also, the PRA will provide additional confidence to the NRC in the overall safety of the design and in the margin to NRC guidelines on core damage frequency and large release. Finally, the NRC Safety Goal Policy quantitative health objective provision demonstrates that an acceptable level of radiological risk to the public, as defined by the NRC Safety Goal Policy, can be achieved without evacuation.

As noted above, this approach of mainly deterministic criteria, with PRA used as a supporting evaluation, is consistent with the industry interpretation of the NRC Severe Accident Policy [6] which states that safety acceptability should be based on an approach which stresses deterministic engineering analysis and judgment, complemented by PRA.

3
ASSESSMENT OF ALWR DESIGN CONFORMANCE WITH REQUIREMENTS

The second main element of the technical basis for ALWR emergency planning is a design specific assessment of individual ALWR designs against the URD design requirements and criteria discussed above. In this section, the results of design specific assessments of the conformance of the three ALWRs with the emergency planning-related URD requirements are summarized. The details of these assessments are contained in Appendices B (AP600), C (System 80+), and D (ABWR).

With regard to containment performance, the steps used for the assessment of conformance with the requirements were as follows:

1. Confirm that the design meets the requirements of the URD, Chapter 5, Section 6.6.2.1 by identifying the plant specific design characteristics and features which meet the requirements.

2. Confirm that containment loads representing those from core damage sequences do not exceed ASME limits specified in the URD Chapter 5, Section 6.6.2.2 for approximately 24 hours under realistic severe accident assumptions.

3. Confirm that no uncontrolled release will occur beyond approximately 24 hours.

With regard to dose, the assessment is to confirm that the plant specific dose at 0.5 mile using the physically-based source term meets the 1 and 5 rem PAGs, and to compare these results with the doses from PRA sequences with frequency $>10^{-7}$ per year to confirm that the physically-based source term is reasonably bounding.

Finally, it is necessary to confirm that the supplemental PRA goals of less than 10^{-5} per year core damage frequency and less than 10^{-6} per year cumulative frequency for sequences resulting in greater than 1 rem are met.

3.1 AP600 Assessment Results

For AP600, the comparison of plant specific design characteristics and features in the AP600 Standard Safety Analysis Report (SSAR) against the URD indicated that all the URD requirements were met. Table B-2 in Appendix B list items for which supplemental information was requested from Westinghouse. This information was provided in reference [20].

Confirmation that core damage containment loads do not exceed ASME limits specified in the URD Volume III, Chapter 5, Section 6.6.2.2 for approximately 24 hours was provided by reviewing AP600 PRA results for accident class 3BE (which is the dominant contributor to core damage frequency with a frequency of 7.8 x 10^{-8} per year) as discussed in Appendix B. Results indicate a peak pressure and gas temperature for 3BE of 36 psia and 370 F, respectively. The LOCA plus hydrogen pressure and temperature are 90.8 psia and 400 F, respectively. Thus there is significant margin to the limits.

Confirmation that no uncontrolled release will occur beyond 24 hours was provided by reviewing a variety of accident sequence classes for longer times. The review indicates that with vessel cavity flooding, overpressure failure will not occur. Even without reactor vessel cavity flooding, overpressure failure will not occur for well after the URD specified time of approximately 2 to 3 days.

As discussed in Appendix B, the AP600 dose at 0.5 mile was evaluated against the PAG using the final NUREG 1465 source term with all release phases (i.e., gap, early in-vessel, ex-vessel, and late in-vessel) considered. The median and 90th percentile 0.5 mile effective doses were determined to be 0.72 and 3.5 rem, respectively, well below the URD limits of 1 rem and 5 rem. Also, as noted in Appendix B, PRA dose results indicate that the physically-based source term is reasonably bounding.

Total core damage frequency for AP600 (internal events, external events, and low power/shutdown conditions) is about 10^{-6} per year which is well under the 10^{-5} goal. Similarly, the cumulative frequency for sequences resulting in greater than 1 rem at 0.5 mile is 2 x 10^{-8} per year which provides significant margin to the 10^{-6} goal. The 2 x 10^{-8} per year also assures large margin to the URD requirement that the NRC Safety Goal Policy quantitative health objective be achieved without evacuation.

3.2 System 80+ Assessment Results

For System 80+, the comparison of plant specific design characteristics and features in the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC) against the URD indicated that all the URD requirements were met. Table C-2 in Appendix C lists URD items for which supplemental information was requested from ABB. This information was provided in reference [21].

Confirmation that core damage containment loads do not exceed ASME limits specified in the URD Volume II, Chapter 5, Section 6.6.2.2 for approximately 24 hours was provided by reviewing System 80+ PRA results for accident sequence LL-3E as discussed in further detail in Appendix C. LL-3E is the representative sequence which is evaluated in the PRA for the smaller large break LOCA accident sequence grouping which, at 5 x 10^{-7} per year, is the dominant contributor to core damage frequency and is the only accident sequence grouping with frequency greater than 10^{-7} per year. Results indicate a 24 hour peak pressure and temperature for LL-3E of 100 psia and 320 F, respectively. The ASME Service Level C limits are 140 psia at 360 F. Thus there is significant margin to the limits.

Confirmation that no uncontrolled release will occur beyond 24 hours was provided by reviewing a variety of accident sequence classes for longer times. The review indicates that with vessel cavity flooding, overpressure failure will not occur. Even without reactor vessel cavity flooding, overpressure failure will not occur for well after the URD specified time of approximately 2 to 3 days.

As discussed in Appendix C, the System 80+ dose at 0.5 mile was evaluated against the PAG using a draft version of the NUREG 1465 source term. This evaluation is reported in Chapter 15 of the CESSAR-DC. This draft NUREG 1465 was an early version of the final NUREG 1465 [4] source term and the draft release parameters are generally conservative with respect to reference [4] as noted in Appendix C. All release phases (i.e., gap, early in-vessel, ex-vessel, and late in-vessel) were considered. The median and 90th percentile 0.5 mile effective doses were determined to be 0.33 rem and 1.65 rem, respectively, well below the URD limits of 1 rem and 5 rem. Also, as noted in Appendix C, PRA dose results indicate that the physically-based source term is reasonably bounding.

Total core damage frequency for System 80+ is about 3×10^{-6} per year which is well under the 10^{-5} goal. Similarly, the cumulative frequency for sequences resulting in greater than 1 rem at 0.5 mile is 2.8×10^{-7} per year which provides over a factor of 3 margin to the 10^{-6}, 1 rem goal. The 2.8×10^{-7} per year also assures large margin to the URD requirement that the NRC Safety Goal Policy quantitative health objective be achieved without evacuation.

3.3 ABWR Assessment Results

For ABWR, the comparison of plant specific design characteristics and features in the ABWR Standard Safety Analysis Report (SSAR) against the URD indicated that all the URD requirements were met. Table C-2 in Appendix C lists URD items for which supplemental information was obtained from ABB. This information was provided in references [22] and [23].

Confirmation that core damage containment loads do not exceed ASME limits specified in the URD Volume III, Chapter 5, Section 6.6.2.2 for approximately 24 hours was provided by reviewing ABWR PRA results for accident class ID (which is the dominant contributor to core damage frequency with a frequency of 7.0×10^{-8} per year) as discussed in Appendix D. Results indicate a peak pressure for accident class ID of 104 psia at 500 F. The LOCA plus hydrogen pressure is 111.7 psia at 500 F. Thus there is margin to the limit.

Confirmation that uncontrolled release will not occur beyond 24 hours was provided by reviewing accident sequence progression and the Containment Overpressure Protection System (COPs). The COPs is a rupture disc-actuated system which provides a scrubbed pathway for fission product release in the event of containment overpressure. The review indicates that the COPs is designed to open beyond 24 hours for most severe accidents and will prevent containment overpressure failure.

As discussed in Appendix D, the ABWR dose at 0.5 mile was evaluated against the PAG using a source term which was based on a conservative approximation to the NUREG 1465 source term with all release phases (i.e., gap, early in-vessel, ex-vessel, and late in-vessel) considered. The median and 90th percentile 0.5 mile effective doses were determined to be 0.88 rem and 4.18 rem, respectively which is below the URD limits of 1 rem and 5 rem. Also, as noted in Appendix D, PRA dose results indicate that the physically-based source term is reasonably bounding.

Internal events core damage frequency for ABWR was estimated to be 1.6×10^{-7} per year. External events and low power/shutdown conditions add about 2×10^{-7} per year. Thus the total core damage frequency is well under the 10^{-5} goal. Similarly, the cumulative frequency for sequences resulting in greater than 1 rem at 0.5 mile is 4×10^{-8} per year which provides significant margin to the 10^{-6} goal. The 4×10^{-8} per year also assures large margin to the URD requirement that the NRC Safety Goal Policy quantitative health objective be achieved without evacuation.

4
ALWR NURE-0396 ASSESSMENT

4.1 Four Considerations from NUREG 0396

The third main element of the technical basis for ALWR emergency planning is an assessment of the ALWR against NUREG-0396. NUREG-0396 was published in 1978 by a joint NRC-EPA task force, which addressed a request for federal guidance on emergency planning from a conference of state radiation control directors.

Four considerations were addressed in NUREG 0396 in determining the recommended EPZ. These considerations were later restated in NUREG 0654:

a. projected dose levels from the most severe design basis accident (DBA) should not exceed the protective action guide (PAG) levels outside the zone,

b. projected dose levels from less severe (i.e., "most") core melt accidents should not exceed the protective action guide (PAG) levels outside the zone,

c. for more severe core melt accidents, doses would generally not cause early injuries outside the zone, and

d. the planning which is performed should provide a substantial base for expansion of response efforts in the event this proved necessary.

In addressing these four considerations, the stated approach in NUREG 0396 was to base the rationale on a “full spectrum of accidents and corresponding consequences tempered by probability considerations.” The probabilities and consequences of severe accidents which were used in NUREG-0396 came primarily from WASH-1400 [15]. WASH-1400, published nearly 25 years ago, was the first LWR PRA performed in the U.S. and reflected the perspectives and state of knowledge on severe accidents which existed in the early-1970s. The WASH 1400 results were used in NUREG-0396 to generate curves of conditional probability of dose exceedance versus distance from the reactor (i.e., conditioned on the assumed occurrence of a core melt). These curves were generic in that they were for a combined PWR and BWR. Figure I-11 of NUREG-0396 (reproduced as Figure 4-1 below) shows these conditional probability of dose exceedance curves, and this figure was the main basis for the recommended 10 mile plume exposure planning distance in NUREG-0396.

4.2 Methodology for ALWR NUREG 0396 Assessment

For the ALWR NUREG-0396 assessment, the four considerations above have been addressed. In addressing these considerations, industry has utilized the NUREG 0396 assumptions and methods where practical in order to keep the ALWR comparison with NUREG 0396 on a common basis. Several aspects of the NUREG 0396 methodology and its application to ALWRs are discussed below.

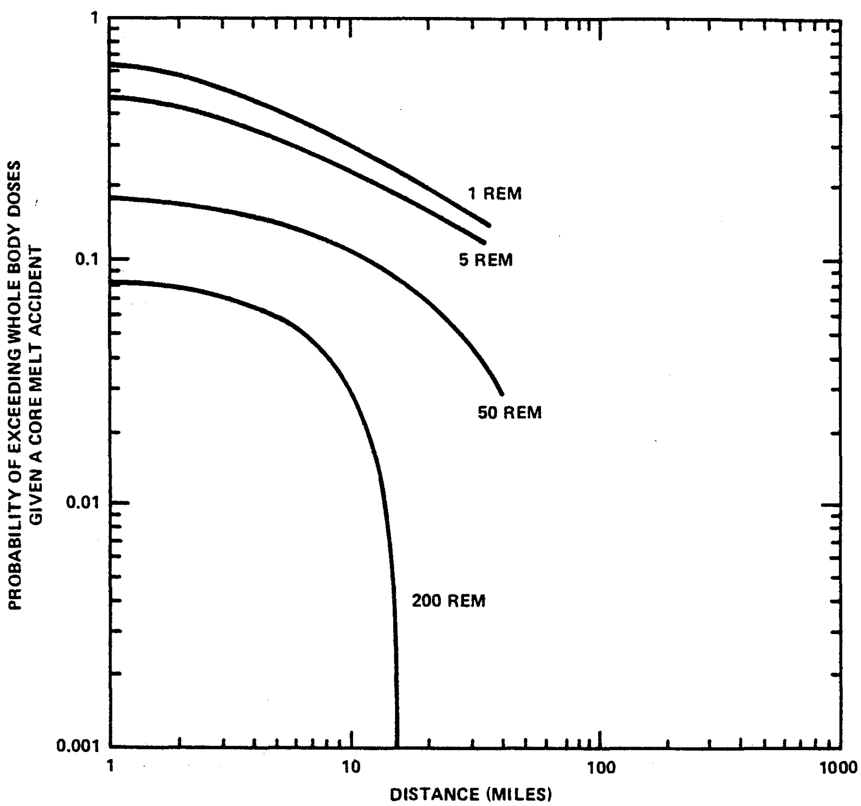

Figure 4-1
Reproduction of NUREG 0396, Figure I-11-Conditional Probability of Exceeding Whole Body Dose Versus Distance. Probabilities are Conditional on a Core Melt Accident (5×10^{-5})

4.2.1 Accute Whole Body vs. EDE

In NUREG-0396 the severe accident dose evaluation (Figure I-11 and related figures) was based on acute whole body dose with one year committed inhalation. For consistency, the ALWR NUREG 0396 assessment uses this same approach.

Use of acute whole body dose is not today's recommended practice for smaller dose levels where the concern includes dose commitment over many years. For example, current EPA guidance [5] specifies EDE as the dosimetric quantity for comparison against the PAG. Accordingly, the ALWR plant specific dose evaluations, which are required by the ALWR URD emergency planning criteria and methodology and are summarized in Section 3 above (and described in more detail in Appendices B, C, and D for AP600, System 80+, and ABWR, respectively), have been based on EDE with 50 year dose commitment for comparison with the PAG.

4.2.2 Combined ALWR Treatment

For the ALWR NUREG-0396 assessment, the ALWR has been evaluated in a generic manner. That is, an analysis has been produced which combines the three ALWR designs. This has been done for two reasons. First, this generic approach is similar to the NUREG-0396 methodology in which the curve of probability of dose exceedance vs. distance was generated for a combined PWR and BWR. Second, a generic approach avoids the tendency to compare one ALWR design against another for the very low frequency accident sequence types. Such comparisons would not be as meaningful as an evaluation which demonstrates that the ALWRs, as a group, are in fact greatly improved in severe accident performance compared to the WASH 1400 characterization of LWRs which was used for the NUREG-0396 evaluation in the 1970s.

4.2.3 Dose Calculation Model

For dose calculations, the intent is to utilize NUREG-0396 modeling assumptions and parameters where practical. The following dose modeling assumptions and parameters were used in the NUREG-0396 evaluation in the 1970s:

- As evident from Figure I-11, core melt was assumed with probability unity.
- Source terms from Table 5-1 of the WASH 1400 Main Report were used as input. Core inventories from Section 3, Appendix VI of WASH 1400 were used.
- Based on footnotes in NUREG-0396, Figures I-11, I-12, I-15, and I-16, the dose reported in Figure I-11 is considered to be whole body acute consisting of three plume pathways: cloud (24 hour exposure), inhalation (1 year exposure), and ground (24 hour exposure).
- Straight line plume trajectory is used per Figure I-11 footnote.
- The shielding factor for radionuclides deposited on the ground is 0.7 per Figure I-15. No shielding factor is assumed for cloud dose. No inhalation protection factor is assumed for inhalation dose.

No attempt was made to determine the site meteorology data base used in the NUREG-0396 work (it may in fact have been a composite site), nor was the dose code investigated in any detail.

To be certain that the ALWR NUREG-0396 assessment provides a common basis for comparison with NUREG-0396, it was necessary to perform a calculation which approximately reproduced Figure I-11 of NUREG-0396. The MACCS 1.5 dose code was used. MACCS is the state-of the-art dose code for severe accident dose calculations in the U.S. and is the successor to the CRAC code, an early version of which was most likely used in NUREG-0396 work. Red bone marrow dose was used to represent whole body dose since MACCS does not report "whole body dose". Comparisons performed between MACCS and later revisions of CRAC indicate significant differences are not expected.

The ALWR site as defined in the URD was used in the benchmark. While it is not known how this site compares to what was used for NUREG-0396 studies, this site is relatively demanding from a meteorological standpoint in that it bounds about 80% of existing U.S. sites based on short term Chi/Q. It is also known that meteorological differences from site-to-site are generally not large compared to other variables in the dose calculation problem.

The results of the benchmark calculation to reproduce the original NUREG 0396 curves are given in Figure 4-2. As is evident from comparing the four NUREG-0396 Figure I-11 curves (i.e., Figure 1) to the Figure 4-2 benchmark (note both figures span 3 decades of conditional probability on the ordinate and 1 to 1000 miles on the abscissa), the agreement is quite good. This provides confidence that curves calculated using the ALWR source terms can be meaningfully compared with the NUREG-0396 results.

4.2.4 Use of Tempering Based on Design Features and Probability

Basis for Use of Tempering Appendix I of NUREG-0396 describes the various rationales for establishing a planning basis including risk, probability, cost effectiveness, and consequence spectrum. The study based the rationale for the planning basis on a “spectrum of consequences, tempered by probability considerations.” NUREG-0396 also states that, "accident probability is important and does have a place in terms of evaluating the range of the consequences of accident sequences and setting some reasonable bounds on the planning basis." Conditional probabilities of various consequences were used to provide perspective on critical doses of concern to emergency planners and probabilities were used in reviewing the planning basis finally chosen. However, none of the set of severe accidents considered in the planning basis was judged so

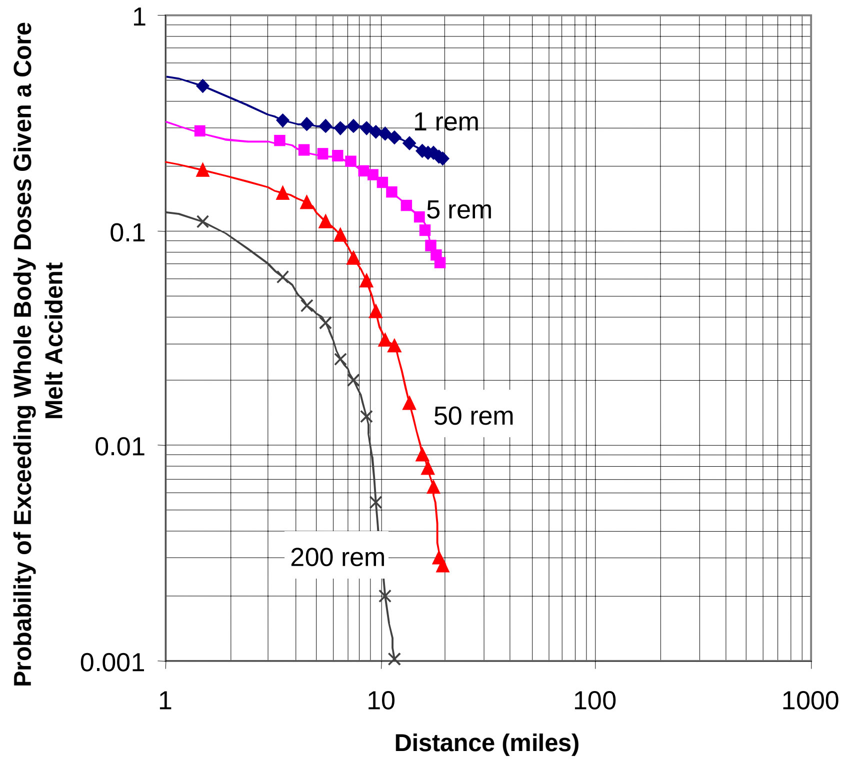

Figure 4-2
Benchmark to Reproduce NUREG 0396, Figure I-11*

* Conditional probability of exceeding whole body dose vs.distance; probabilities are conditional on a core melt accident (median probability ~5 x 10^{-5} per year for WASH 1400)

unlikely that its inclusion in the basis would be unreasonable based on its WASH-1400 estimated probabilities of occurrence. One of the reasons given in NUREG-0396 against using probability as the principal rationale was that, “A generic 'probability of an event' appropriate for planning has many implications felt to be outside the scope of the Task Force objective.”

Since the release of NUREG-0396 over 20 years ago, a number of factors have changed which make the use of a tempering process more appropriate. (Tempering in the ALWR context is intended to mean consideration of severe accident design features and accident management, and the low probabilities which result, as an input to the planning rationale.) The Commission's Safety Goal Policy has been issued which provides guidelines on probabilities for large releases. There have been two decades of substantial progress in severe accident understanding and assessment methodologies. Significant advances have occurred in accident management equipment, procedures, and training as discussed in Appendix F. Further, as summarized in Sections 2.0 and 3.0 above, there are design features and capabilities in ALWRs which specifically address severe accidents. Relative to designs characterized in NUREG-0396 using WASH-1400, these advances in severe accident understanding, accident management, and design have led to significantly reduced accident probabilities and source terms compared to those estimated in WASH-1400.

The use of a tempering process is also consistent with the Commission statement that the staff should remain open to suggestions to simplify emergency planning requirements that are designed with greater safety margins [24]. It is further reinforced by the ACRS letter to the Commission where the ACRS describes the need to reconsider the emergency planning basis for ALWRs and to consider risk arguments [25]. Consideration of the ALWR emergency planning basis is an example of an issue where a risk-based approach would provide useful insights. Most recently, in SECY-97-020 [26] the NRC indicated that it was receptive to industry petitions for changes in emergency planning requirements, and indicated that to justify such changes it is necessary to address the probability level, if any, below which accidents will not be considered for emergency planning.

Based on the above, a tempering process has been used in determining the range of accident sequence types to be included in the severe accident dose evaluation performed as part of the ALWR NUREG-0396 assessment. The steps in applying the tempering process are as follows:

a. Use the complete set of accident sequences in the ALWR plant specific PRAs as a starting point.

b. Identify those sequence classes which have probability of occurrence which is so low that it would be unreasonable to include the sequences in the NUREG-0396 assessment, or for which the time to the beginning of significant release provides adequate warning time; the remaining sequences are to be included in the ALWR NUREG-0396 assessment.

c. Review design aspects of the low probability sequence classes to confirm the existence of design features and capabilities which would tend to support the low probability of occurrence (or time delay) of the dominant accident sequence types which make up the sequence class.

<u>Specification of Probability Screen</u> To assess what would be a reasonable probability screen for the tempering process for severe accidents, guidance and precedents for consideration of probability in risk-based work were investigated.

The guidance and precedents for power reactors include the following:

- NUREG-1150 [27] used a frequency cutoff of 10^{-7} per year for PRA accident sequence progression
- NUREG-1420 [28] discusses probability cutoff criteria for PRAs and their relation to the Commission's Safety Goal Policy, and indicates that consequences with frequencies lower than about 10^{-7} per year "...are not meaningful for decision making."
- The Standard Review Plan [29] guidance specifies evaluation of potential accidents from hazards which exceed 10^{-7} per year
- NUREG-0396, Figure I-11, has a conditional probability range down to 10^{-3} which corresponds to ~10^{-7} per year absolute probability (since WASH 1400 mean core damage frequency is ~10^{-4} per year)

- NUREG-1338 [18], draft Pre-Application SER for Modular HTGR, states as part of justification for reduced emergency planning that sequences appearing to have a frequency in the range of about 10^{-7} per year will be examined for residual risk
- Regulatory Guide 1.174 [30] specifies that an increase of 10^{-7} per year in large early release frequency (LERF) is permitted for proposed plant changes

A final point is to note that large natural or manmade catastrophic events (e.g., meteorites) that would make emergency planning irrelevant, from a nuclear plant licensing perspective, have a frequency of occurrence of the order of 10^{-7} per year.

Additional perspective on the 10^{-7} per year can be obtained by considering the NRC evaluation of probability bounds for credible events in connection with pre-closure of the high level waste repository [31]. This NRC evaluation concluded that events with probability lower than 10^{-6} per year can be screened from further consideration in repository analysis. The basis for this conclusion was the observation that events with probability less than 10^{-6} per year, with consequences which would likely not exceed several tens of rem, result in estimated risk of cancer fatality which is less than 10^{-8} per year. Reference [31] went on to put this risk in perspective by noting that the ICRP has indicated that a fatal cancer risk in the range of 10^{-5} to 10^{-6} per year from exposure to radiation would likely be acceptable to members of the public. Reference [31] thus concludes that events which result in fatal cancer risks on the order of 10^{-8} per year do not contribute significantly to individual risk.

In the case of reactor accidents with probability in the range of 10^{-7} per year and consequences which would likely not exceed several tens of rem (say 30 rem), the fatal cancer risks are of the order of

$$(10^{-7}/\text{yr})(5\text{x}10^{-4}\ \text{fatalities/rem})(\sim 30\ \text{rem}) = \sim 10^{-9}\ \text{per year}$$

This is well below the level which would contribute significantly to individual risk.

The above guidance and precedents support a probability screen of $\sim 10^{-7}$ per year. While a 10^{-7} screen is appropriate in general, it was considered too high for ALWRs since the very low ALWR core damage frequency ($\sim 10^{-6}$ per year) and the ALWR containment design features have resulted in no containment challenges above 10^{-7}.

A probability screen of 10^{-8} per year was considered for ALWRs and would have included some containment challenges. However, the peer review report (see Appendix H) recommended that sequences down to 3 orders of magnitude below core damage frequency be included since the NUREG-0396 Figure I-11 methodology is based on this relative frequency range. Thus, a probability screen of 2×10^{-9} per year (i.e., 3 decades below the average ALWR core damage frequency of 2×10^{-6}) was used in the ALWR assessment against NUREG-0396. As noted in step (3) of the tempering process, design aspects of low probability sequences were reviewed to confirm existence of plant design features and capabilities which support the low probabilities. All sequence classes with frequency less than 10^{-7} were reviewed.

4.2.5 Use of More Realistic Credit for Mitigating Factors

The sequences which are to be included in the ALWR NUREG 0396 assessment are very low in probability. Since these low probability sequences tend to control the results of the NUREG 0396 assessment, an effort was made to take more realistic credit for mitigating factors than was generally done in the ALWR PRAs, including:

- Credit severe accident management actions which are provided for in ALWR design documentation. Examples include RCS depressurization, steam generator injection, and manual containment spray. Appendix F discusses severe accident management as applied to ALWRs.
- Credit more realistic phenomenological assumptions where the ALWR PRAs were excessively conservative (for example in cases where the conservatism expedited completion of the PRA for purposes of obtaining 10 CFR 52 Final Design Approval, but did not impact overall risk) or did not consider a phenomenon. Appendix G discusses an important phenomenon, not considered in the PRAs, for fission product aerosol retention in unisolated SGTR accidents. Other phenomena and excess conservatisms are noted below in Section 4.3 and in Tables 4-1, 4-2, and 4-3.
- Eliminate sequences with time to beginning of release of 24 hours or longer. This is more than adequate time on the basis of historical ad hoc evacuations in the US which have ranged between 2 and 8 hours [18].

4.3 ALWR Source Terms

Applying steps (1) and (2) from 4.2.4 above, the release categories to be considered in the ALWR severe accident MACCS evaluation are identified in Tables 4-1, 4-2, and 4-3 for AP600, System 80+, and ABWR, respectively (a release category is generally a group of accident sequences where the most limiting release sequence is selected to represent the release for source term purposes). The frequency, release timing, and release magnitude for noble gas and iodine for the various release categories are taken from the respective ALWR PRAs. The basis for including or not including the release category in the ALWR NUREG-0396 assessment is summarized in the last column. The following supplemental information was used in the MACCS evaluation for addressing the more realistic credit for mitigating factors per Section 4.2.5 above:

- AP600 release categories BP (6E, 6L) and BP (3A) were combined since the release magnitude and timing are essentially the same. The resulting combined category (8.4×10^{-9} per year) was then split into two types, per Appendix G, one with tube rupture at the tube sheet (probability $0.3 * 8.4 \times 10^{-9}$), and one with tube rupture at or above the tube bundle midpoint (probability $0.7 * 8.4 \times 10^{-9}$). For tube rupture at the tube sheet, an aerosol DF of 100 is applied per Appendix G, thus reducing the iodine release fraction from 0.17 to 0.0017 (with corresponding reductions to the other non-noble gas radionuclide group release fractions). For tube rupture at the tube bundle midpoint, the aerosol is assumed to be essentially completely removed per Appendix G. The noble gas release is assumed to be unchanged (i.e., the release magnitude from the AP600 PRA is used for both).

- In the AP600 PRA, release category CFE has a release magnitude and timing based on early hydrogen detonation. Basing the CFE release on early hydrogen detonation was done by Westinghouse for expediency in the AP600 PRA since it had little effect on risk [32]. The actual frequency of early hydrogen detonation is 2.5×10^{-11}. Thus, CFE was not included in the MACCs evaluation. The AP600 PRA accident classes which make up the bulk of the probability of release category CFE are vessel failure accidents which, per Appendix B of the AP600 PRA, have been determined to not fail containment [32].
- System 80+ release categories RC4.22E and RC4.30E were combined since the release magnitude and timing are essentially the same. The resulting combined category (1.3×10^{-8} per year) was then split into two types, per Appendix G, one with tube rupture at the tube sheet (probability $0.3*1.3 \times 10^{-8}$), and one with tube rupture at or above the tube bundle midpoint (probability $0.7*1.3 \times 10^{-8}$). For tube rupture at the tube sheet, an aerosol DF of 100 is applied per Appendix G, thus reducing the iodine release fraction from 0.25 to 0.0025 (with corresponding reductions to the other non-noble gas radionuclide group release fractions). For tube rupture at the tube bundle midpoint, the aerosol is assumed to be essentially completely removed per Appendix G. The noble gas release is assumed to be unchanged (i.e., the release magnitude from the System 80+ PRA is used for both).

Table 4-1
Treatment of AP600 Release Categories for NUREG-0396 Assessment[1]

AP600 Release Category	PRA Frequency (yr^{-1})	Time of Beginning of Core Damage (hr)[2]	Approximate Time of Containment Failure[3] (hr)	PRA Noble Gas, Iodine Release[4]	Basis for Including or Not Including Release Category in NUREG-0396 Assessment
IC	1.7E-7	1.1	N/A	9E-4, 5E-6	Included since probability $>2x10^{-9}$
BP (6E, 6L)	4.2E-9	13	Containment bypass	1.0E0, 1.7E-1	Included since probability $>2x10^{-9}$ (see Section 4.3)
BP (1A)	1.8E-9	2	5	8.6E-1, 3.8E-1	Not included based on low probability; note that there are accident management provisions to inject into the SG which will tend to prevent tube failure and mitigate long term release even if tube rupture occurs
BP (1AP)	1.1E-9	25	28	2.5E-1, 4.3E-1	Not included based on low probability and significant time (>24 hours) leading up to core melt; note that there are accident management provisions to inject into the SG which will tend to prevent tube failure and mitigate long term release even if tube rupture occurs
BP (3A)	4.2E-9	13	Containment bypass	1.0E0, 1.7E-1	Included since probability $>2x10^{-9}$ (see Section 4.3)
CI	3.6E-10	1.3	Containment isolation failure	8.4E-1, 3.4E-2	Not included based on low probability; note also that accident management containment spray will mitigate the release[5]

Table 4-1 (continued)
Treatment of AP600 Release Categories for NUREG-0396 Assessment[1]

AP600 Release Category	Frequency (yr^{-1})	Time of Beginning of Core Damage (hr)[2]	Approximate Time of Containment Failure[3] (hr)	Noble Gas, Iodine Release[4]	Basis for Including or Not Including Release Category in NUREG-0396 Assessment
CFE	6.6E-9	1.3	1.6	7.0E-1, 8.3E-2	Not included based on low probability (see Section 4.3); note also that accident management containment spray will mitigate the release[5]
CFI	1.3E-11	1.3	5	6.2E-1, 3.4E-3	Not included based on low probability; note also that accident management containment spray will mitigate the release[5]
CFL	1.5E-11	1.1	46.5	1.1E-3, 1.2E-5	Not included based on low probability and time to containment failure

1. All information in the first five columns of the table is taken from the AP600 PRA, Rev.8.
2. This is the time after the initiating event at which the fuel clad begins to fail.
3. This is the time after the initiating event at which the containment fails.
4. This is the fraction of the noble gas and iodine core inventory released to the environment for 24 hrs after the time of containment failure .
5. Per reference [33], the reduction in aerosol release for release category CFE due to the AP600 accident management spray system is about a factor of 6, and the reduction in aerosol release for release category CI is a factor of 8. A reduction of a factor of 6 is assumed for CFI on the basis of its similarity to CFE.

Table 4-2
Treatment of System 80+ Release Classes for NUREG-0396 Assessment[1]

System 80+ Release Class[2]	Frequency (yr^{-1})	Time of Beginning of Core Damage (hr)[3]	Approximate Time of Containment Failure (hr)[4]	Noble Gas, Iodine Release[5]	Basis for Including or not Including Release Class in Backup Assessment
RC1.1E	1.4E-6	4	N/A	5E-3, 2E-7	Included since probability $>2x10^{-9}$
RC1.1M	4E-7	16	N/A	5E-3, 2E-5	Included since probability $> 2x10^{-9}$
RC4.22E	6E-9	4	Containment bypass	1E0, 2.4E-1	Included since probability $>2x10^{-9}$ (see Section 4.3)
RC4.30E	7E-9	4	Containment bypass	1E0, 2.5E-1	Included since probability $>2x10^{-9}$ (see Section 4.3)
RC4.36L	3E-8	25	Containment bypass	1E0, 3.5E-1	Not included due to significant time leading up to core melt; note also that there is aerosol retention in SG tube bundle, and accident management provisions to inject into the SG to mitigate the release
RC3.1E-RC3.6E	1.8E-8	4	5	1E0, 2E-2	Not included based on supplemental information indicating that containment will not fail even if cavity fails (see Section 4.3)
RC3.2M, RC3.6M	3.6E-9	16	17	1E0, 6E-2	Not included based on supplemental information indicating that containment will not fail even if cavity fails (see Section 4.3)

Table 4-2 (continued)
Treatment of System 80+ Release Classes for NUREG-0396 Assessment[1]

System 80+ Release Class[2]	Frequency (yr^{-1})	Time of Beginning of Core Damage (hr)[3]	Approx. Time of Containment Failure (hr)[4]	Noble Gas, Iodine Release[5]	Basis for Including or not Including Release Class in Backup Assessment
RC5.1E	5E-10	2	Containment bypass	1E0, 6E-2	Not included based on low probability; note also there is supplemental information indicating a more realistic release fraction of ~5E-3 for iodine/cesium[6]
RC4.8E	1E-9	4	Containment Bypass	1E0 8E-3	Not included based on low probability
RC2.1E, RC2.2E	6.7E-9	4	11	1E0, 1E-4	Included since probability >$2x10^{-9}$
RC2.4E- RC2.7E	1.2E-7	4	65	5E-2, 3E-4	Not included since containment failure is so far out in time
RC2.2M, RC2.6M	1.3E-8	16	65	1E0, 6E-5	Not included since containment failure is so far out in time
RC2.5M, RC2.7M	1.6E-8	10	65	1E0, 6E-3	Not included since containment failure is so far out in time

1. All information in the first five columns of the table is taken from the System 80+PRA, Amendment W.
2. For entries with more than one Release Class (RC), the frequency is the sum of the RC frequencies for that entry.
3. This is the time after the initiating event at which the fuel clad begins to fail.
4. This is the time after the initiating event at which the containment fails.
5. This is the fraction of the noble gas and iodine core inventory released to the environment for 24 hrs after the time of containment failure.
6. See ABB letter, reference [34], which states that more realistic, but still conservative analyses of the interfacing LOCA (RC5.1E) indicate that the fraction of core inventory of CsI released is of the order of $5x10^{-3}$.

Table 4-3
Treatment of ABWR Release Categories for NUREG-0396 Assessment[1]

ABWR Release Category	Frequency (yr^{-1})	Beginning of Core Damage (hr)[2]	Approximate Time of Containment Failure (hr)[3]	Noble Gas, Iodine Release[4]	Basis for Including or Not Including Release Category In Backup Assessment
NCL[5]	1.3E-7	0.5	N/A	4.4E-2, 2.3E-5	Included since probability >$2x10^{-9}$
Case 1	2.1E-8	0.5	20	1E0, 1.5E-7	Included since probability >$2x10^{-9}$
Case 7 (LCHPPFPM, LCLPFSBR)	1.4E-10	0.5	18	1E0, 5E-3	Not included based on low probability
Case 8 (LCHPPFEH)	2.1E-10	0.5	2	1E0, 1.9E-1	Not included based on low probability; note also there is supplemental information indicating that containment is not expected to fail even if vessel breach at high pressure occurs[6]
Case 9 (SBRCPFD90)	1E-12	9	24	1E0, 1.7E-1	Not included based on low probability and the significant time to containment failure
Case 8 (LCHPPFBD)	2.6E-12	0.5	18	1E0, 5E-3	Not included based on low probability
Case 8 (LCHPPFBR)	2E-13	0.5	9	1E0, 8E-2	Not included based on low probability
Case 7 (LCLPFSD90)	2.6E-10	0.5	31	Negligible	Not included based on low probability and the significant time to containment failure
Case 2	<1E-10	0.5	20	1E0, 5E-6	Not included based on low probability

Table 4-3 (continued)
Treatment of ABWR Release Categories for NUREG-0396 Assessment[1]

ABWR Release Category	Frequency (yr^{-1})	Beginning of Core Damage (hr)[2]	Approximate Time of Containment Failure (hr)[3]	Noble Gas, Iodine Release[4]	Basis for Including or Not Including Release Category In Backup Assessment
Case 3	<1E-10	0.5	50	1E0, 3E-4	Not included based on low probability and significant time to containment failure
Case 4	<1E-10	0.5	20	1E0, 1.6E-3	Not included based on low probability
Case 5	<1E-10	0.5	19	1E0, 6E-3	Not included based on low probability
Case 6	<1E-10	0.5	19	1E0, 3.1E-2	Not included based on low probability

1. All information in the table is taken from the ABWR PRA, Rev.4, Amendment 34, with the following clarifications.
   - Case 9 (SBRCPFD90) frequency is assumed to be that of STC 33 of Figure 19D.5-3 of ABWR SAR
   - Case 8 (LCHPPFBD), which is an unsprayed release, has the sprayed and unsprayed branch probabilities reversed (STC 9 and 10 of Figure 19D.5-3 of ABWR SAR); thus the frequency of the unsprayed release has been assumed to be that of the sprayed branch (STC 9 of Figure 19D.5-3).
   - Case 8 (LCHPPFBR), which is an unsprayed release, has the sprayed and unsprayed branch probabilities reversed (STC 7 and 8 of Figure 19D.5-3 of ABWR SAR); thus the frequency of the unsprayed release has been assumed to be that of the sprayed branch (STC 7 of Figure 19D.5-3); an additional mu;tiplier of ~0.2 has been included to account for the conditional probability of the vacuum breaker being fully open given vacuum breaker failure to close (based on Figure 19EE-1 of the ABWR SAR)
2. This is the time after the initiating event at which the fuel clad begins to fail.
3. This is the time after the initiating event at which the containment fails.
4. This is the fraction of the noble gas and I core inventory released to the environment within 24 hrs after core damage.
5. Normal Containment Leakage.
6. Reference [35], a report prepared for Polestar by Dr. Brian Cantwell of Stanford University, indicates that even if high pressure breach of the reactor vessel lower head occurred, the geometry of the lower drywell, together with consideration of lateral expansion of the shock wave, would mitigate the pressure rise in the drywell so that upper drywell head failure is not expected.

- System 80+ release categories RC2.1E and RC2.2E were combined since the release magnitude and timing are essentially the same.

- In the System 80+ PRA, the release magnitude for release categories RC3.1E – RC3.6E and RC3.2M – RC3.6M is based on a steam explosion-induced failure of the reactor cavity leading to containment failure. Supplemental information from reference [34] indicates that even if the cavity fails from a steam explosion, there would not be significant movement of the reactor vessel and corresponding containment failure. Thus release categories RC3.1E – RC3.6E and RC3.2M – RC3.6M were not included in the NUREG-0396 assessment.

Using the release categories from Tables 4-1, 4-2, and 4-3 and the supplementary information discussed above, probability of dose exceedance calculations were performed with MACCS as described in Section 4.2.

4.4 Application of ALWR Source Terms for ALWR Assessment Against the Four NUREG 0396 Considerations

4.4.1 Radiological DBA-LOCA

For emergency planning purposes, meeting the PAG for the design basis accident (i.e., the radiological DBA-LOCA), which is consideration a. in Section 4.1, should not be considered separately as was done in NUREG-0396 and NUREG-0654. Rather, the DBA should be subsumed in the spectrum of severe accidents. The separation of the DBA from severe accidents was done in NUREG-0396 when it was developed in the late 1970s since at that time PRA and severe accident evaluation were very new concepts and were not widely understood or accepted. Thus the radiological DBA and associated meteorology (which were part of the regulatory process and were the only well-documented accident and dose calculations available for essentially all plants and sites) was retained in the context of emergency planning.

Today, however, PRA and severe accident evaluation are much more mature technologies. Further, as a result of the TMI-2 accident and severe accident research, it is now recognized that the radiological DBA-LOCA and the less severe core melt accidents, from a radiological standpoint, are essentially the same accident. NUREG 1465 recognizes this and bases radiological DBA-LOCA release magnitude, timing, and chemical form on severe accident phenomena. Thus considerations a. and b. above have been combined by integrating the radiological DBA-LOCA into the spectrum of severe accidents for the ALWR NUREG-0396 assessment. This also eliminates the inconsistent treatment of meteorology in NUREG-0396 in which a conditional probability of unity was assumed for the worst-case, DBA meteorology whereas a probabilistic approach was used for meteorology for severe accidents in WASH 1400 and Figure I-11. Assuming that the worst-case, DBA meteorology occurs 100% of the time is unrealistic and contrary not only to use of probabilities in other aspects of the NUREG-0396 assessment, but also to today's emphasis on consideration of risk in decision making.

4.4.2 Less Severe Core Melt Accidents

Meeting the PAG for less severe (i.e., most) core melt accidents is consideration b. in Section 4.1. Using the ALWR source terms from Tables 4-1, 4-2, and 4-3 and the NUREG-0396 approach as outlined in Section 4.2, 1 rem and 5 rem dose curves have been calculated as shown on Figure 4-3 to characterize the less severe core melt accidents. Note that Figure 4-3 spans 0.1 to 10 miles on the abscissa in order to show necessary detail close to the plant, and three decades on the ordinate as is the case in NUREG-0396, Figure I-11. It is evident from this curve that the conditional probability of exceeding 1 rem at 0.5 mile is of the order of 0.25 which is lower than the NUREG-0396, Figure I-11 value of 0.3 conditional probability of exceeding 1 rem at 10 miles. Similarly, the conditional probability of exceeding 5 rem at 0.5 miles is of the order of 0.06 which is lower than the corresponding NUREG 0396, Figure I-11 value of 0.25 at 10 miles.

In addition to this NUREG 0396 assessment, the less severe core melt accidents have been compared with the PAGs on a plant specific basis as required in the ALWR URD and as summarized in Section 3 above and discussed in detail in Appendices B, C, and D. The results of these evaluations indicate that for each ALWR plant design, the TEDE at 0.5 miles does not exceed the PAGs.

4.4.3 More Severe Core Melt Accidents

No early injuries at the emergency planning boundary for more severe core melt accidents is consideration c. in Section 4.1. Using the ALWR source terms from Tables 4-1, 4-2, and 4-3, the probability of dose exceedance for 50 rem and 200 rem acute whole body dose have been calculated as shown on Figure 4-3. As is evident from Figure 4-3, the ALWR conditional probability of exceeding 50 rem at 0.5 mile is of the order of 0.006 which is lower than the NUREG-0396, Figure I-11 value of 0.1 conditional probability of exceeding 1 rem at 10 miles. The ALWR conditional probability of exceeding 200 rem at 0.5 mile is less than 0.001, comparable to the NUREG-0396, Figure I-11 probability of exceeding 200 rem at 10 miles.

To further evaluate the ALWR 200 rem curve, a four decade plot is provided in Figure 4-4. It is evident from Figure 4-4 that the ALWR 200 rem curve is essentially vertical at ~0.5 mile.

Table 4-4 summarizes the Figure 4-3 results for the ALWR and compares the conditional probabilities of exceeding various doses against the corresponding conditional probabilities from NUREG-0396, Figure I-11.

To assess the sensitivity of the Figure 4-3 results to the unisolated SGTR aerosol DF of 100 discussed in Section 4.3 and detailed in Appendix G, an aerosol DF of 30 was applied. The results are given in Figure 4-5. It is evident from Figure 4-5 that the conclusion that the ALWR curve conditional probabilities at 0.5 mile are less than the NUREG-0396, Figure I-11 curve conditional probabilities at 10 miles still applies.

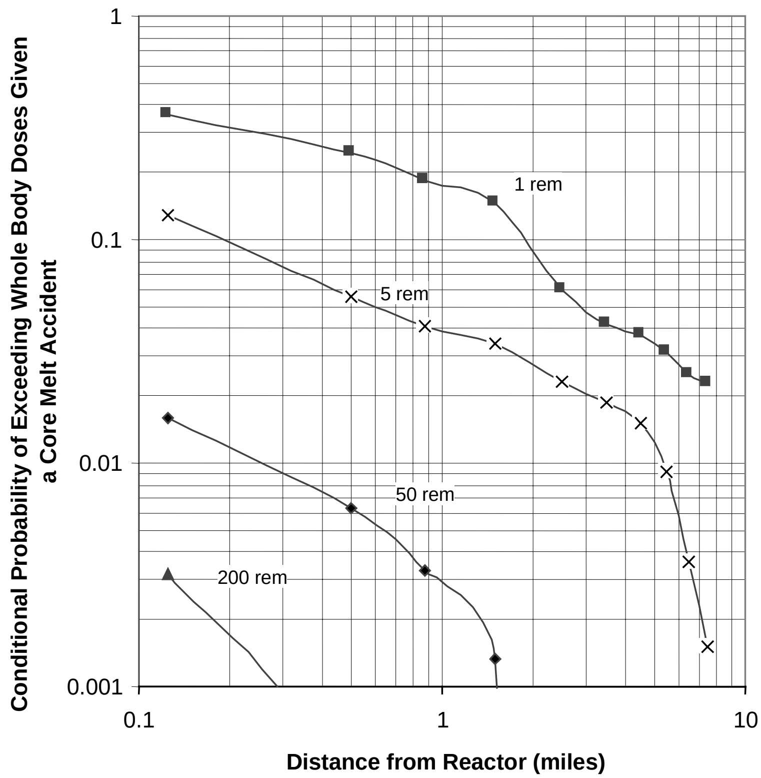

Figure 4-3
ALWR Assessment for Comparison with NUREG 0396, Figure I-11*

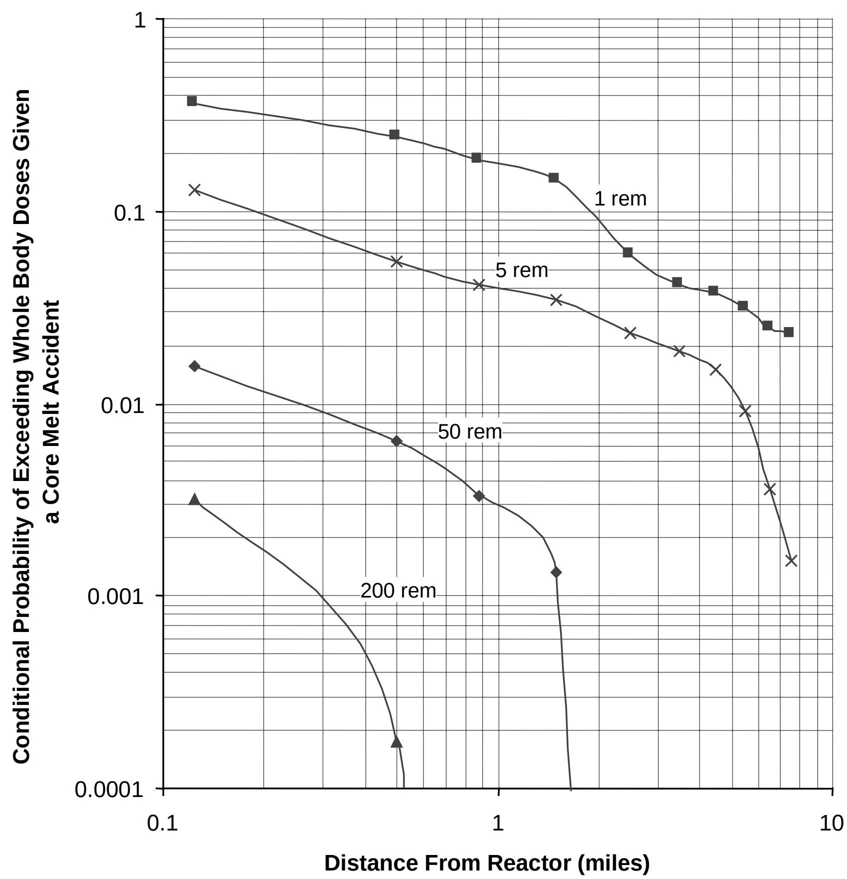

Figure 4-4
Expansion of ALWR NUREG-0396 Assessment to Four Decades

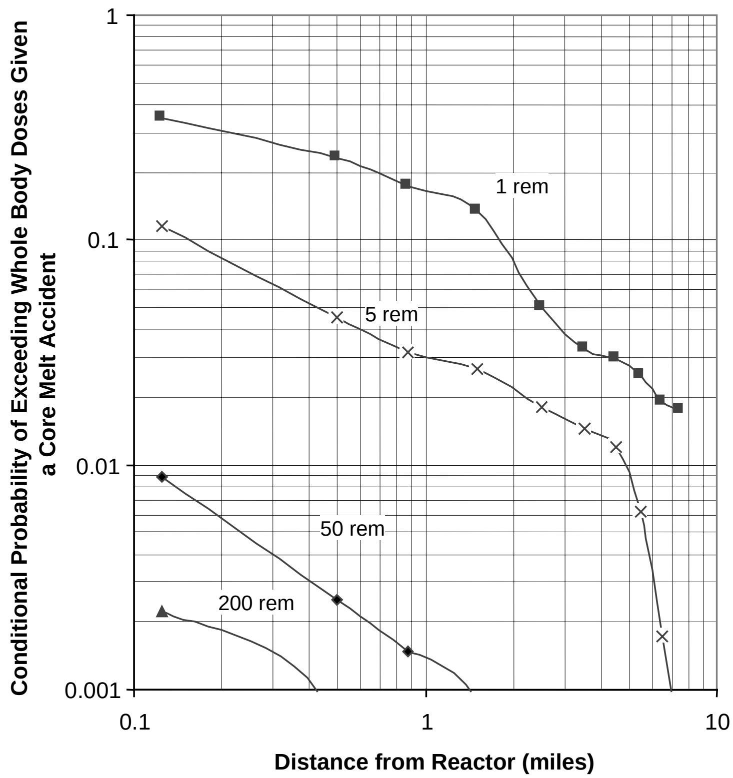

Figure 4-5
ALWR NUREG-0396 Assessment with Reduced SGTR DF.

Table 4-3
ALWR NUREG-0396 Dose Assessment Results

	ALWR (0.5 mile)	**NUREG-0396, Fig I-11 (10 miles)**
Cond. prob. of exceeding 1 rem	0.25	0.3
Cond. prob. of exceeding 5 rem	0.06	0.25
Cond. prob. of exceeding 50 rem	0.006	0.1
Cond. prob. of exceeding 200 rem	<0.001	0.01 – 0.001

4.4.4 Expansion of Base of Response

The last NUREG-0396, NUREG-0654 consideration (consideration d. in Section 4.1) is that the planning which is performed should provide a substantial base for expansion of response efforts in the event this proved necessary. This is being accomplished for ALWR emergency planning through use of the awareness area as described in Section 5 below.

4.5 Review of Design Aspects of Sequences Less than 10^{-7} Per Year

In accordance with step (3) in Section 4.2.4, a review of the design aspects of ALWRs has been performed to confirm the existence of design features and capabilities which support the low probability of occurrence (or the time delay) ALWR accident sequences.

The results of the review of release categories which were $<10^{-7}$ per year are presented in Tables 4-5, 4-6, and 4-7 for the AP600, System 80+, and ABWR, respectively. It is evident from the information in the tables that for all release categories, plant design features and capabilities exist which provide a basis for a low probability of occurrence for the accident sequence types which make up the release category. It is further evident from the tables that there are aspects of each $<10^{-7}$ release category which would substantially mitigate the consequences (i.e., delay and/or reduce the magnitude of release).

Table 4-4
Design Aspects of AP600 Release Categories <10^{-7} Per Year

Release Category (Accident Class)[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspects of Release
BP (6E, 6L)	Spontaneous SG tube rupture, stuck open SG safety valve, core damage at 13 hrs	•Improved water chemistry, tube material, tubesheet and tube support plate design •Automatic, redundant RCS depressurization	Even if it occurred, the release is estimated to be delayed for 13 hours after the initiating event, and aerosol release would be reduced (to less than 1% of the maximum release assumed in NUREG 0396) due to aerosol removal in the SG tube bundle (see Appendix G)
BP (1A)	Induced rupture of SG tubes at 5 hrs, stuck open SG safety valve	•Improved water chemistry, tube material, and tubesheet and tube support plate design •Passive RHR •Automatic, redundant RCS depressurization	Accident management actions to inject into the SG (e.g., using startup a.c. powered feedwater pump) would preclude induced SG tube rupture (See Appendix F); several hours is available after the initiating event to depressurize (if necessary) and inject into the SG
BP (1AP)	Induced rupture of SG tubes at 28 hrs, stuck open SG safety valve	•Improved water chemistry, tube material, and tubesheet and tube support plate design •Automatic, redundant RCS depressurization	Even if it occurred, the release would be delayed for over 24 hours after the initiating event

Table 4-5 (continued)
Design Aspects of AP600 Release Categories <10^{-7} Per Year

Release Category (Accident Class)[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspects of Release
BP (3A)	Rupture of SG tubes from ATWS (overpressure), stuck open SG safety valve	•Control rod insertion •Automatic, redundant RCD depressurization	Same as BP (6E, 6L)
CI (3BE-5)	Containment isolation failure	•Redundant valve isolation •Improved reliability of isolation (e.g., reduced number of penetrations, capability for on-line check of containment integrity)	Even if it occurred, with containment sprays the aerosol release magnitude is estimated to be less than 1% of the maximum release assumed in NUREG 0396
CFE (3BE-8)	Containment failure at ~2 hrs due to hydrogen detonation	•Redundant hydrogen igniter system •Large, strong containment which will withstand hydrogen burns	Even if it occurred, with containment sprays the aerosol release magnitude is estimated to be less than 1% of the maximum release assumed in NUREG 0396
CFI (3BE-9)	Containment failure at ~5 hrs due to hydrogen detonation	•Redundant hydrogen igniter system •Large, strong containment which will withstand hydrogen burns	Even if it occurred, the aerosol release magnitude is estimated to be less than 1% of the maximum release assumed in NUREG 0396
CFL	Containment failure at ~47 hrs due to overpressure	•Cavity flooding system •Passive containment cooling system	Even if it occurred, the release is estimated to be delayed for ~2 days and release magnitude is estimated to be negligible

[1]The release category and accident class numbers are taken from the AP600 PRA, Tables 34-4 and 45-6.

Table 4-5
Design Aspects of System 80+ Release Classes <10^{-7} Per Year

Release Class[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspect of Release
RC4.22E, RC4.30E	Spontaneous SG tube rupture, safety injection failure, unisolated SG	•Improved water chemistry, tube material, tubesheet and tube support plate design •Automatic, redundant RCS depressurization	Even if it occurred, the aerosol release would be reduced (to less than 1% of the maximum release assumed in NUREG 0396) due to aerosol removal in the SG tube bundle (see Appendix G)
RC4.36L	Spontaneous SG tube rupture, safety injection success, IRWST inventory depletion, unisolated SG	•Improved water chemistry, tube material, tubesheet and tube support plate design •Automatic, redundant RCS depressurization	Even if it occurred, the release is estimated in the PRA to be delayed for 25 hours after the initiating event, and aerosol release would be reduced (to less than 1% of the maximum release assumed in NUREG 0396) due to aerosol removal in the SG tube bundle (see Appendix G)
RC3.1E-RC3.6E	Steam explosion at time of RV breach (~5 hrs after initiating event due to ECCS failure) leads to failure to support RV and which in turn leads to penetration failure	•Per reference [34], the upper cavity and adjacent structures are strong enough to support the RV even with failure of the entire lower cavity wall	Even if it occurred, reference [34] states that the 3.5% iodine release estimate from the PRA, which is mainly gaseous evolution and reevolution, is overestimated since the gaseous iodine will be small due to pH control, and the containment temperature will be relatively low due to containment sprays

Table 4-6 (continued)
Design Aspects of System 80+ Release Classes <10^{-7} Per Year

Release Class[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspect of Release
RC3.2M RC3.6M	Steam explosion at time of RV breach (~17 hrs after initiating event due to long term decay heat removal failure) leads to failure to support RV and which in turn leads to penetration failure	•Per reference [34], the upper cavity and adjacent structures are strong enough to support the RV even with failure of the entire lower cavity wall	Even if it occurred, the release is estimated in the PRA to be delayed until 17 hours after the initiating event, and the iodine release magnitude is overestimated as indicated above
RC5.1E	Interfacing LOCA in a line in subsphere	•High to low pressure interfaces provided with isolation valves with leak testing capability, position indication in control room, and high pressure alarm •Interlocks prevent isolation valve opening when RCS pressure exceeds interfacing system pressure •Interfacing systems designed to withstand full RCS without rupture	Even if it occurred, the aerosol release magnitude is estimated in reference [34] to be about 1% of the maximum release assumed in NUREG 0396
RC4.8E	Containment isolation failure	•Redundant valve isolation •Improved reliability of isolation (e.g., capability for periodic on-line check of containment integrity)	Even if it occurred, the aerosol release magnitude is estimated in the PRA to be about 1% of the maximum release assumed in NUREG 0396

Table 4-6 (continued)
Design Aspects of System 80+ Release Classes <10^{-7} Per Year

Release Class[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspect of Release
RC2.1E, RC2.2E	Containment failure at ~11 hrs due to hydrogen burn	•Redundant hydrogen igniter system	Even if it occurred, the release is estimated to be delayed until 11 hours after the initiating event, and aerosol release magnitude is estimated to be negligible compared to the maximum release assumed in NUREG 0396
RC2.4E-RC2.7E	Containment failure at ~65 hrs due to basemat penetration	•Cavity flooding system	Even if it occurred, the release is estimated to be delayed for over 2 days, and the aerosol release magnitude is estimated to be negligible compared to the maximum release assumed in NUREG 0396
RC2.2M, RC2.6M	Containment failure at ~65 hrs due to overpressure	•Automatic, redundant containment spray system	Even if it occurred, the release is estimated to be delayed for over 2 days, and the aerosol release magnitude is estimated to be negligible compared to the maximum release assumed in NUREG 0396
RC2.5M, RC2.7M	Containment failure at ~65 hrs due to basemat penetration	•Cavity flooding system	Even if it occurred, the release is estimated to be delayed for over 2 days, and the aerosol release magnitude is estimated to be about 1% of the maximum release assumed in NUREG 0396

[1]The release class numbers are taken from the System 80+ PRA, Table 19.12.3-1.

Table 4-6
Design Aspects of ABWR Release Categories <10^{-7} Per Year

Release Category (Source Term Category [STC] and Acc Group)[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspects of Release
Case 7 (STC 19/LCLP)	Suppression pool bypass due to vacuum breaker failure at 18 hours	•Vacuum breaker position indication in control room •Loads associated with breaker operation accounted for in the design so as to minimize the potential for failure with the breakers in open or closed position	Even if pool bypass occurred, operator action could be taken to actuate ADS and/or drywell sprays, both of which would limit pressure rise in wetwell; even if release occurred, it is estimated to be delayed until about 18 hours after the initiating event, and aerosol release magnitude is estimated to be about 1% of the maximum release assumed in NUREG 0396
Case 7 (STC 5/LCHP)	RV breach at high pressure, failure of moveable containment penetrations at 18 hours due to high temperature	•Automatic, redundant RCS depressurization would prevent debris from dispersing into the upper drywell. •Operator action to actuate DW spray which would prevent the high temperature failure	Even if it occurred, the release magnitude is estimated to be negligible compared to the NUREG 0396 release. In addition, the release is delayed significantly since the release rate is governed by revaporization. finally, the penetration failure would allow fission products to enter the reactor building which would result in some fission product deposition.
Case 7 (STC 21/LCLP)	DW head failure at 31 hours due to overpressure	•Containment RHR would prevent overpressure failure •Containment overpressure system (COPS) averts DW head failure and enables a scrubbed release from the wetwell to the stack	Even if it occurred, the release is delayed for over 24 hours due to firewater being added to containment.

Table 4-7 (continued)
Design Aspects of ABWR Release Categories <10^{-7} Per Year

Release Category (Source Term Category [STC] and Acc Group)[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspects of Release
Case 7 (STC 22/LCLP)	DW head failure at 20 hours due to overpressure	•Containment RHR would prevent overpressure failure •Containment overpressure system (COPS) to avert DW head failure and enable a scrubbed release from the wetwell to the stack	Even if it occurred, the release is delayed due to firewater being added to containment . At 24 hours after the start of release only 0.5% of the cesium and iodine have been released from the containment building. The release rate is governed by fission product revaporization.
Case 8 (STC 11/LCHP)	RV breach at high pressure causing mechanical failure of DW head at 2 hours	•Automatic, redundant RCS depressurization would prevent debris from dispersing into the upper drywell.	Per reference [35], even if high pressure breach of RV occurred, the geometry of the lower drywell, together with consideration of lateral expansion of the shock wave, would mitigate the pressure rise in the drywell so that upper DW head failure is not expected. Also, operator action to actuate DW spray would scrub fission products entering drywell.
Case 8 (STC 8/LCHP)	Suppression pool bypass due to vacuum breaker failure at 9 hours	•Vacuum breaker position indication in control room •Loads associated with breaker operation accounted for in the design so as to minimize the potential for failure with the breakers in open or closed position.	Even if pool bypass occurred, operator action could be taken to actuate ADS and/or drywell sprays, both of which would limit pressure rise in wetwell; even if it occurred, the release is estimated to be delayed until about 9 hours after the initiating event, and aerosol release magnitude is estimated to be about 14% of the maximum release assumed in NUREG 0396

Table 4-7
Design Aspects of ABWR Release Categories <10^{-7} Per Year

Release Category (Source Term Category [STC] and Acc Group)[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspects of Release
Case 8 (STC 9/LCHP)	RV breach at high pressure, DW head failure from overpressure at ~18 hours	•Automatic, redundant RCS depressurization would prevent debris from dispersing into the upper drywell. •Containment RHR would prevent overpressure failure •Containment overpressure system (COPS) to avert DW head failure and enable a scrubbed release from the wetwell to the stack	Even if it occurred, the aerosol release magnitude is estimated to be less than 1% of the maximum release assumed in NUREG 0396
Case 9 (STC 34/SBRC)	DW head failure from overpressure at ~24 hours	•Combustion turbine generator reduces likelihood of blackout	Even if it occurred, the release would be delayed for about 24 hours after the initiating event
Case 2 (STC 18/LCLP)	Containment overpressure system (COPS) functions so that release is from wetwell air space to the stack	•Containment RHR would prevent containment overpressure challenge	Even if it occurred, the release is estimated to be delayed until 20 hours after the initiating event, and, because of scrubbing in the suppression pool, aerosol release magnitude is estimated to be negligible compared to the release in NUREG 0396

Table 4-7 (continued)
Design Aspects of ABWR Release Categories <10^{-7} Per Year

Release Category (Source Term Category [STC] and Acc Group)[1]	Description of Containment Failure	Design Features and Capabilities which would Preclude Containment Failure	Mitigating Aspects of Release
Case 3 (STC 9/LCHP)	DW head failure at 50 hours due to overpressure	•Automatic, redundant RCS depressurization would prevent debris from dispersing into the upper drywell •Containment overpressure system (COPS) to avert DW head failure and enable a scrubbed release from the wetwell to the stack	Operator action to actuate DW sprays and scrub fission products entering drywell. Even if it occurred, the release is estimated to be delayed for over 2 days, and the aerosol release magnitude is estimated to be negligible compared to the release in NUREG 0396
Case 5 (STC 42/LBLC	Containment overpressure system (COPS) functions so that release is from wetwell air space to the stack	•Containment RHR would prevent containment overpressure challenge	Even if it occurred, the release is estimated to be delayed for 19 hours after the initiating event, and the aerosol release magnitude is estimated to be about 1% of the maximum release assumed in NUREG 0396
Case 6 (STC 46/LBLC)	DW head failure at 19 hours due to overpressure	•Containment RHR would prevent containment overpressure challenge	Even if it occurred, the release is estimated to be delayed for 19 hours after the initiating event, and the aerosol release magnitude is estimated to be about 5% of the maximum release assumed in NUREG 0396

[1] The source term category and accident group numbers are taken from ABWR PRA, Tables 19D.5-3 and 19E.3-6.

5

DEVELOPMENT OF ALWR EMERGENCY PLANNING CONCEPT INCLUDING PROVISIONS OF BASE FOR EXPANSION OF RESPONSE

As noted in the introduction to Section 1, the ALWR emergency planning concept has been designed to build on and be consistent with the technical basis. Development of the plan concept starts with four general principles which industry has defined for ALWR emergency planning. These general principles are then used as guidance in characterizing the emergency plan concept which is being proposed. The description of the ALWR plan is at a conceptual level in Section 5. Reference [36], which is an industry prepared ALWR supplement to NUREG 0654 [37] in draft form, contains more details on the planning actions.

5.1 General Principles

The four general principles defined to guide the development of a new emergency planning concept are as follows:

Principle 1 - Emergency planning is a necessary part of the defense-in-depth philosophy of nuclear safety and should be provided for power reactors. This is the case notwithstanding the much improved accident prevention and mitigation capabilities of ALWRs, i.e., emergency planning provides defense-in-depth beyond the design capabilities. This philosophy of defense-in-depth would also call for planning that supports the capability for expansion of the actions taken during an actual emergency should this be necessary.

Principle 2 - The concept and details of the emergency plan should be commensurate with the facility design, that is, with the risk associated with the specific design. The design affects the likelihood of an offsite release as well as the timing and magnitude of the release. Thus, the emergency plan for a particular class of plants should reflect the likelihood, timing, and magnitude of the offsite release for that class of plants. The NRC has certainly recognized this as existing regulations specify different emergency planning measures for gas-cooled reactors, test reactors, fuel storage facilities, and the current generation of LWRs. As described above in the discussion of technical basis, ALWRs offer fundamental improvements in severe accident prevention and mitigation design. These fundamental improvements in design should be reflected in the degree and details of ALWR emergency planning.

Principle 3 - ALWR emergency planning should reflect the experience from existing emergency planning regulations and implementation at operating plants. Particularly important are the need for organizational responsibilities for emergency planning which are matched to the authority of the organization, and a plan which addresses the project investment risk which results from existing emergency planning.

Principle 4 - A common framework should be considered for offsite emergency planning for non-nuclear industrial hazards and nuclear power plants. This is desirable to avoid the complexity and confusion of overlapping emergency response systems and procedures, and to provide a consistent level of protection for comparable risks. It is also consistent with recent FEMA action to implement an all-hazards approach, combining emergency management, technical assistance, and resources for all emergencies, both radiological and non-radiological [38].

These four principles are considered in characterizing the plan concept below.

5.2 Characterization of Plan Concept

The existing plume exposure pathway EPZ is a single area out to 10 miles. For the ALWR it is proposed that the plume exposure pathway consist of two areas: the response area and the awareness area. The response area would be that area closest to the reactor, within which a severe accident could cause radiological consequences of sufficient concern that a rapid response should be included in planning. The awareness area would be the area beyond the response area within which the radiological effects would be smaller.

The ALWR would also have an ingestion exposure pathway planning area similar in concept to that of existing plants.

5.2.1 Size of Response Area

Consistent with general principle 2, the size of the emergency planning areas should be commensurate with the plant design capability. As discussed in detail in Sections 2 and 3 above, the containment performance and dose mitigation aspects of the ALWR designs provide assurance that even if core damage should occur, the fission product release from containment would be slow and small relative to the EPA PAGs. In addition, as shown in Section 4 the offsite releases from more severe core damage events would generally be much smaller and slower than that from WASH 1400 which was used as the basis for NUREG 0396.

The suggested size of the response area is 0.5 mile. This is based on the following:

- The URD requirement that median dose for 24 hours for most core damage events is less than 1 rem EDE is met at 0.5 mile from the reactor.
- The URD requirement that 90th percentile dose for 24 hours for most core damage events is less than 5 rem EDE is met at 0.5 mile from the reactor.
- The fact that the first three (i.e., dose-related) NUREG 0396 considerations, including the consideration that doses from more severe core damage accidents would generally not cause early injuries, are met at 0.5 mile.

Thus the response area, with an ALWR design, is comparable to the 10 mile EPZ with a design characterized by WASH 1400.

5.2.2 Awareness Area as Base for Expansion of Response

To address general principle 1 on defense-in-depth and the fourth NUREG 0396 consideration of providing a substantial base for expansion of response in the plume exposure pathway in the event that this is necessary, the awareness area has been defined. This base for expansion of response will be accomplished by: (1) providing increased awareness on the part of the public residing in the area as to how they will be advised in the event of an emergency at the plant site, (2) providing increased awareness on the part of State and/or local jurisdictions so as to facilitate actions which they take in discharging their emergency management responsibilities.

The size of the awareness area should be large enough that the increased awareness noted above could be expanded beyond the awareness area distance if necessary. The size of the awareness area will be somewhat site specific since it depends upon political boundaries, the characteristics of the terrain, and the emergency planning capabilities of State and local jurisdictions. Generally, a distance of about 3 miles is judged to be large enough such that it could be expanded if necessary, and the three mile distance itself would reduce dose by over an order of magnitude relative to the dose at 0.5 mile.

It is noted that this two area plume exposure pathway plan concept provides flexibility to better match organizational responsibility with authority (consistent with general principle 3) and provides a natural means to incorporate the all-hazard approach for offsite emergency planning (consistent with general principle 4).

5.2.3 Size of Ingestion Exposure Pathway Area

The existing emergency planning ingestion exposure pathway EPZ is 50 miles. This is based on maintaining the projected infant thyroid dose from cow's milk below the EPA PAG (1.5 rem). As discussed in Appendix E, this distance for ALWRs is 25 miles.

5.2.4 Timing of Response Actions

Consistent with general principle 2, emergency planning should be based on timing of response actions which is commensurate with the plant design capability. Significant atmospheric release (i.e., release which is expected to cause doses exceeding ~1 rem inside 24 hours) would occur only if and when core damage and containment failure (or at least excessive containment leakage) occur (i.e., more severe core melt accidents).

Based on the discussion in Section 4.2 and as summarized Tables 4-5, 4-6, and 4-7, sequences which could result in significant atmospheric release have been effectively precluded by design in the ALWR. Thus significant time (>24 hours) would be expected to be available for response actions. However, for purposes of providing guidance on timing of response actions, the accident categories in Tables 4-1, 4-2, and 4-3 which were included in the NUREG-0396 assessment are utilized. The following guidance is provided on timing of response actions:

- In two of the Table 4-1, 4-2, 4-3 accident categories (AP600 IC and ABWR NCL), core damage occurs sooner than several hours, but neither of these sequences involve significant release. Furthermore, based on the Table 4-1, 4-2, 4-3 accident categories included in the NUREG-0396 assessment, significant atmospheric release, even if it is assumed to occur, generally begins 2 hours or more after the accident initiating event. For conservatism, the minimum time from initiating event to start of significant atmospheric release is taken to be 2 hours.
- The time period over which radioactive material may be released is taken to be 0.5 hour to one or more days after the start of significant atmospheric release.
- The time at which the major portion of release may occur is taken to be 0.5 hour to approximately 1 day after the start of significant atmospheric release.
- The time from the initiating event to significant integrated dose at 0.5 mile for most core melts is beyond 24 hours.
- In addition to larger releases, extreme weather (very stable conditions leading to a concentrated plume) would be necessary in order to cause doses in excess of 1 rem beyond the awareness area. Such weather conditions generally involve low wind speeds (0 to 1 meter per second [39]), leading to expected plume arrival times at the awareness area distance of about 3 hours after the beginning of the release. Thus the time from the initiating event to significant integrated dose at 3 miles is several hours to one day.

6
SUMMARY OF ALWR EMERGENCY PLANNING BASIS

As discussed in this report, the technical foundation of ALWR emergency planning is the ALWR design for accident prevention and mitigation. This ALWR design capability results in accident sequence probabilities and source terms which are significantly reduced compared to the WASH 1400 accident sequence probabilities and source terms which were used as the basis for the 1978 NUREG 0396 evaluation.

Table 6-1 summarizes NUREG 0396 results for the ALWR and compares the conditional probabilities of exceeding various doses against the corresponding conditional probabilities from NUREG-0396, Figure I-11.

The proposed ALWR emergency planning concept is commensurate with the ALWR design capability and is based on showing that the URD emergency planning design criteria as well as NUREG-0396 considerations are satisfied. These ALWR emergency planning technical criteria are very similar to the existing emergency planning technical criteria, with updates having been provided in two areas. One is that the ALWR criteria include deterministic requirements (design features and associated accident analyses) intended to address specific severe accident challenges. These deterministic requirements are discussed in Sections 2 and 3. The other is that the ALWR NUREG-0396 assessment, discussed in Section 4 with results summarized in Table 6-1 below, updates the NUREG 0396 tempering process so as to consider the severe accident design features and accident management, and the low probabilities which result, as an input to the planning rationale.

Table 6-1
ALWR NUREG-0396 Dose Assessment Results

	ALWR (0.5 mile)	NUREG-0396, Fig I-11 (10 miles)
Cond. prob. of exceeding 1 rem	0.25	0.3
Cond. prob. of exceeding 5 rem	0.06	0.25
Cond. prob. of exceeding 50 rem	0.006	0.1
Cond. prob. of exceeding 200 rem	<0.001	0.01 – 0.001

Table 6-2 provides a summary of the basis for ALWR emergency planning and a comparison with the basis for existing emergency planning. As is evident from Table 6-2, the ALWR emergency planning concept is based on essentially the same criteria as existing emergency planning.

Table 6-2
Summary of Emergency Planning Basis

	Existing Emergency Planning	ALWR Emergency Planning
Plume exposure pathway planning distance	Distance based on meeting dose limits[1]	Distance based on meeting dose limits[1]
Planning basis for expansion of response	Onsite plan and offsite plan[2]	Onsite plan and all-hazards plan[2]
Ingestion exposure pathway planning area	Distance based on meeting dose limits[3]	Distance based on meeting dose limits[3]

1. Applicable dose limits are the 1 rem EDE PAG for most core melts and early injury dose (200 rem whole body) for more severe core damage accidents. For existing emergency planning the dose limits are met at 10 miles. For ALWR emergency planning the dose limits are met at 0.5 mile.
2. The onsite and offsite plan provide a basis for expansion of response for existing emergency planning beyond the plume exposure pathway planning distance. The ALWR onsite plan and the associated all-hazards plan (i.e., the awareness area) provide a basis for expansion of response of ALWR emergency planning beyond the plume exposure pathway planning distance.
3. The applicable dose limit is the 1.5 rem milk pathway thyroid PAG . For existing emergency planning this dose limit is met at 50 miles. For ALWR emergency planning this dose limit is met at 25 miles.

Existing emergency planning, which is based on NUREG-0396 dose calculations using WASH 1400 sequence probabilities and source terms, meets the plume exposure pathway dose limits at about 10 miles, hence the 10-mile EPZ. ALWR emergency planning, which is based on dose calculations which use ALWR sequence probabilities and source terms, meets the plume exposure pathway dose limits at 0.5 mile, hence the 0.5 mile response area.

The planning basis for expansion of response for existing emergency planning is the fact that the onsite and offsite planning, as well as the implementation actions taken in an actual emergency, would facilitate protective actions beyond the plume exposure pathway planning distance boundary (i.e., the 10-mile EPZ), should such actions be necessary. Similarly, for ALWR emergency planning, the onsite and offsite planning (which includes preparations made in the awareness area per the all-hazards plan) as well as the implementation actions taken in an actual emergency, would facilitate protective actions beyond the plume exposure pathway planning distance boundary (i.e., the 0.5 mile response area), and in fact even beyond the awareness area boundary, should such actions be necessary.

Finally, for the ingestion exposure pathway, existing emergency planning, which is again based on NUREG-0396 dose calculations using WASH 1400 sequence probabilities and source terms, meets the dose limits at about 50 miles. ALWR ingestion exposure pathway emergency planning, which is based on dose calculations which use ALWR sequence probabilities and source terms, meets the ingestion exposure pathway dose limits at about 25 miles.

7
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A
ALWR UTILITY REQUIREMENTS DOCUMENT EMERGENCY PLANNING RELATED REQUIREMENTS

Section A.1
ALWR URD Emergency Planning Criteria
(Volume II, Chapter 1, Section 2.6)

VOLUME II, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	Requirement	Rationale	Rev.
2.6	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING**	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING**	8
	ALWRs shall be designed to allow simplification and standardization of emergency planning. The Plant Designer shall perform an evaluation of the plant design against two ALWR emergency planning technical criteria prescribed below for containment performance and offsite dose. The methodology which is specified for demonstrating the criteria shall be utilized in this evaluation. The Plant Designer shall also perform a supplemental PRA evaluation in support of the evaluation against the two ALWR emergency planning criteria.	Technical criteria and methodology are provided so as to specify what a Plant Designer seeking approval of ALWR emergency planning for a particular plant design must demonstrate. It is intended that these criteria and methodology form the technical basis for any necessary regulatory action on emergency planning for both the plume exposure pathway and the ingestion pathway. The criteria and methodology are intended to be used in an integrated manner and the criteria should not be applied without utilizing the methodology specified in this section. The criteria and methodology for containment performance and dose evaluation are primarily deterministic. The PRA evaluation is not a criterion itself but rather is intended to complement the two criteria. This is consistent with the NRC Severe Accident Policy which states that safety acceptability should be based on an approach which stresses deterministic engineering analysis, complemented by PRA. The requirements in this section are generally unique to emergency planning although the containment performance criterion draws heavily on containment performance requirements in other locations of the Utility Requirements Document.	8

VOLUME II, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	Requirement	Rationale	Rev.
2.6.1	**Containment Performance Criterion**	**Containment Performance Criterion**	8
	For ALWR emergency planning, the plant shall be provided with the capability to address severe accident containment challenges, including design features and characteristics to preclude core damage sequences which could bypass containment, and to withstand loads representing those associated with core damage sequences. The methodology in Section 2.6.4 below shall be used to evaluate that capability. The LOCA plus hydrogen load from Chapter 5, Section 6.6.2.2 should not be exceeded for a period of approximately 24 hours after the start of release of fission products from the fuel.	While ALWR accident prevention design features make the possibility of core damage extremely remote, specifying the capability to address severe accident containment challenges, including avoiding containment bypass and withstanding loads which are expected to envelope best estimate pressure and temperatures associated with severe accident conditions, provides confidence that the containment can withstand a severe accident. Not exceeding LOCA plus hydrogen loads for approximately 24 hours provides low leakage for the period corresponding to the site boundary dose criterion.	8
	Beyond approximately 24 hours, means for preventing uncontrolled fission product release from containment shall be provided in accordance with Chapter 5, Section 6.6.2.6.	Even if a core damage event should occur, the ALWR Program considers that it is very likely that the ALWR containment would be able to meet appropriate limits for an indefinite time period, i.e., no containment overpressure would occur. This is based on LWR accident management capabilities and the TMI-2 accident experience which suggest that it is likely that core damage events will be recovered in-vessel, and on ALWR reactor cavity design features (e.g., debris spreading area, flooding of debris) which are designed to quench the ex-vessel debris. Nevertheless, for defense-in-depth purposes, a requirement has been specified for no uncontrolled release beyond approximately 24 hours to provide protection against long-term containment overpressure failure. Radioactive decay and removal of fission products in containment is such that a release at 16 hours, or even earlier depending on the plant design, would result in no acute health effects at the site boundary. Thus, the approximately 24-hour period provides significant margin	

VOLUME II, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	Requirement	Rationale	Rev.
		to that time at which the acute health effects dose threshold could be exceeded.	
2.6.2	**Dose Criterion**	**Dose Criterion**	8
	Dose at 0.5 mile shall be evaluated per the methodology in Section 2.6.5 below and shall be shown not to exceed the Protective Action Guides (PAGs) for a period of approximately 24 hours from the start of release of fission products from the fuel.	The PAGs are projected dose levels for evacuation (1 to 5 rem) which are specified by the Environmental Protection Agency (EPA) in a 1992 report as guidance for actions to protect the public in the early phase of a nuclear incident (i.e., the plume exposure pathway). As noted in NUREG-1338, based on experience for non-radiological emergencies, evacuations take from two to eight hours, including time to notify the public. Not exceeding the PAG for approximately 24 hours would provide significant margin for ALWR accident detection, notification, and evacuation, if necessary. A separate ingestion pathway requirement has not been specified since the ingestion exposure planning distance will be determined on a generic basis for all ALWRs as noted in Section 2.6.5 below.	8
2.6.3	**Supplemental PRA Evaluation**	**Supplemental PRA Evaluation**	8
	A PRA evaluation shall be performed per the methodology in Section 2.6.6 below to demonstrate that the following goals are met: • A mean core damage frequency ≤ 10-5/yr; • A cumulative frequency <10-6/yr for sequences resulting in greater than 1 rem total effective dose equivalent (TEDE) over 24 hours at the site boundary.	The requirement to perform the supplemental PRA evaluation and the associated goals are intended to demonstrate the integrated effectiveness of the two emergency planning criteria (Sections 2.6.1 and 2.6.2 above). The supplemental PRA also serves as a tool for the Plant Designer for refining and optimizing the design. Finally, the supplemental PRA will provide confidence to the NRC in the overall safety of the plant and in the margin to NRC guidelines on core damage frequency and large release. Given the guidance in the NRC Severe Accident	8

VOLUME II, CHAPTER 1: OVERALL REQUIREMENTS

Paragraph No.	Requirement	Rationale	Rev.
		Policy Statement, it is not intended that the PRA goals be made part of design certification or of any rulemaking on emergency planning.	
	In addition, it shall be demonstrated that ALWR designs are consistent with the prompt accident quantitative health objective of the NRC Safety Goal Policy with no credit for evacuation prior to 24 hours.	This requirement demonstrates that an acceptable level of radiological risk to the public, as defined by the prompt accident quantitative health objective of the NRC Safety Goal Policy, can be achieved with evacuation, which as noted in Section 2.6.2, can be accomplished with significant margin within 24 hours.	
2.6.4	**Methodology for Demonstrating Containment Performance Criterion**	**Methodology for Demonstrating Containment Performance Criterion**	8
	The Plant Designer shall demonstrate that the pressure and temperature loads associated with core damage sequences are no more limiting than the peak LOCA plus hydrogen loads of Chapter 5, Section 6.6.2.2. For plant designs meeting the requirements of Chapter 5, Section 6.6.2.1, the characteristics of the core damage sequences shall be as follows:	Chapter 5, Section 6.6.2.2, requires that the peak LOCA plus hydrogen loads not exceed applicable ASME limits. The loads associated with core damage sequences must therefore be no more limiting than the LOCA plus hydrogen loads.	8
	• Containment is isolated and otherwise intact (i.e., no bypass has occurred); • Reactor coolant system is depressurized to <100 psig; • Ample water is in the reactor cavity/lower drywell prior to or immediately upon vessel penetration for cooling ex-vessel core debris; • Containment heat removal is adequate; • BWR containments are inerted, and hydrogen control	Consistent with Chapter 5, Section 6.6.2, design characteristics and features are to be provided which address severe accident challenges, including bypass and loads from core damage sequences. An exhaustive set of severe accident challenges, regardless of the probability of occurrence of the challenge, have been addressed based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. The conclusion from the technical work in support of this requirement is that if core damage should occur, it will be into an intact containment with the RCS at low pressure and with containment systems	

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	system is functioning.	functioning as designed.	
	Best estimate severe accident methods shall be utilized in evaluating the loads. Accepted industry computer codes such as MAAP shall be applied.	Best estimate methods are appropriate for the severe accident evaluation since the evaluation relates to matters beyond the design basis, i.e., the ALWR Safety Margin Basis, and since the ALWR plant features for addressing severe accident challenges significantly reduce the uncertainty in severe accident phenomena.	
	In accordance with Chapter 5, Section 6.6.2.3, functional sequence types from the PRA with mean frequencies greater than 10-7/yr shall be analyzed for containment response. Functional sequence types with mean frequency less than 10-7/yr shall be reported for discussion.	Evaluation of core damage sequence loads with sequence selection based on PRA provides additional assurance that risk-significant sequences are being considered.	
2.6.5	**Methodology for Demonstrating Dose Criterion**	**Methodology for Demonstrating Dose Criterion**	8
	The demonstration that the dose criterion is met shall utilize a physically-based source term release into an intact containment as defined in Chapter 5, Section 2.4.1. The methodology for the PAG dose evaluation shall consist of the following.	The physically-based source term is based on release and removal phenomena from actual core damage sequences and should be reasonably bounded for source terms from the probabilistically significant sequences. The intact containment is based on ALWR containment performance requirements which have been specified such that severe accident challenges to containment are effectively precluded or can be accommodated, thus providing integrity of the containment.	8
2.6.5.1	**Approach**	**Approach**	8
	A probabilistic dose (PD) method (e.g., CRAC2 or MACCS) shall be used.	A PD method is chosen for consistency with the basis for existing emergency planning and the fact that PD methods have provision for the particulate component of the source term and thus are an appropriate method for calculating PAG comparison doses. The use of CRAC2,	8

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Paragraph No.	Requirement	Rationale	Rev.
		MACCS, or another similar code is consistent with current level 3 PRA evaluations and ALWR PRA Key Assumptions and Groundrules (KAG).	
2.6.5.2	**Meteorological Database**	**Meteorological Database**	8
	The meteorological database shall be that provided in Annex B to Appendix A to Chapter 1 of the URD.	This meteorological database is that provided in the PRA KAG. It is an actual site meteorological database for which the RG 1.145 two-hour Exclusion Area Boundary X/Q is estimated to be greater than the X/Q for 80 to 90 percent of U.S. operating sites.	8
2.6.5.3	**Direction-Dependent vs. Direction-Independent**	**Direction-Dependent vs. Direction-Independent**	8
	The dose calculation shall be direction-independent.	The calculations supporting existing emergency planning are direction-independent, i.e., the frequency of exceeding given dose levels is provided independent of direction. The NRC safety goals use a direction-independent approach as well. The use of a direction-independent approach is also consistent with the methods to be used in preparing the complementary cumulative distribution function (CCDF) for the exceedance frequency of off-site doses at the site boundary required by the PRA KAG.	8
2.6.5.4	**Statistical Measure of Dose to be Compared to 1 Rem PAGs**	**Statistical Measure of Dose to be Compared to 1 Rem PAGs**	8
	The dose to be compared to the 1 rem PAG for ALWR emergency planning shall be the median dose.	Existing emergency planning was based in part on establishing that "most" core melt accidents would not exceed the PAG. There are two sources of variability in determining the meaning of "most" in the situation for existing emergency planning (i.e., NUREG 0396): the	8

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		magnitude of the source term, and the meteorology. A similar approach is used here for ALWR emergency planning. Median dose (i.e., 50th percentile meteorology) together with the physically-based source term, which tends to bound the source term expected for nearly all core melt accidents in an ALWR, assures that the dose from most core melt accidents will not exceed 1 rem.	
2.6.5.5	**Statistical Measure of Dose to Compared to 5 Rem PAG**	**Statistical Measure of Dose to Compared to 5 Rem PAG**	8
	The dose to be compared to the 5 rem PAG shall be the 90th percentile dose.	More extreme (e.g., very stable atmospheric conditions, low wind speed) meteorology could cause higher doses for a given source term. While doses exceeding 1 rem would not be expected as noted above, a 5 rem limit has been specified for 90th percentile meteorology in order to address more extreme meteorological conditions. A 5 rem limit for such conditions is considered reasonable on several grounds. First, ICRP 63 recommends a dose limit for evacuation no lower than 50 mSv (i.e., 5 rem). Second, under stable, low wind speed conditions, the plume is concentrated (only about 100 feet wide at 0.5 mile) and is moving slowly, so the need for rapid evacuation would be quite limited. Finally, 5 rem is the upper end of the 1 to 5 rem range recommended by EPA and thus is a reasonable limit for emergency planning purposes under low probability weather conditions.	8
2.6.5.6	**Whole Body Dose vs. Effective Dose Equivalent**	**Whole Body Dose vs. Effective Dose Equivalent**	8
	The dose to be calculated is the sum of the effective dose equivalent (EDE) resulting from exposure to external sources (cloud shine and ground shine) and the committed effective dose equivalent (CEDE) from plume inhalation.	The May 1992 revision to Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (PAG Manual) calls for the use of EDE as the basis for determining off-site doses in relation to the 1 to 5 rem PAG. MACCS already employs this concept, as does the	8

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		current 10CFR20. A separate thyroid dose limit is unnecessary since the EDE includes the organ weighted contribution from thyroid exposure. Not specifying a separate thyroid dose limit is also consistent with the recent NRC 10CFR 50/100 rule change which specifies EDE as defined in Section 2.6.5.6. A separate ingestion exposure pathway requirement has not been specified since the ingestion exposure planning distance will be determined, using the May, 1992 PAG Manual guidelines, on a generic basis for all ALWRs. This will be accomplished by assuming that the 0.5 mile dose is equal to the PAG (i.e., the EDE limit for plume exposure), determining the maximum iodine contribution to this dose, and using this maximum iodine release as the basis for calculating the distance at which the ingestion dose equals the controlling ingestion pathway PAG (i.e., a projected infant thyroid dose from cow's milk of 1.5 rem on a preventative basis and 15 rem as a basis for emergency contamination).	
2.6.5.7	**Inclusion of Organic Iodide in the PAG Calculation**	**Inclusion of Organic Iodide in the PAG Calculation**	8
	In calculating doses for comparison with the PAG values to justify ALWR emergency planning, the contribution from organic iodide can be neglected.	The I and HI are quite reactive and are likely to undergo natural deposition as rapidly (or more rapidly) than the particulate. Given that pH is controlled as specified in the Utility Requirements Document, the actual dose contribution from organic iodide is expected to be very small (a few percent of thyroid dose) and thus can be omitted from the dose calculation.	8

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.8	**Dose Commitment**	**Dose Commitment**	8
	A dose commitment of 50 years shall be used for TEDE from plume inhalation.	In the May 1992 revision of the PAG Manual, plume inhalation dose commitment is assumed to be the "lifetime". It is judged that a 50-year commitment is adequate on a generic basis to fulfill that requirement; it is also the duration used in the current 10CFR20. This differs from the PRA as specified in the KAG where the intent is to compare calculated doses to the 25 rem threshold for acute health effects (based on the current 25 rem whole body requirement in 10CFR100). It also differs from NUREG 0396 which utilizes one year commitment for inhalation.	8
2.6.5.9	**Radionuclides to be Included**	**Radionuclides to be Included**	8
	The radionuclides identified in Table II-2 of the CRAC2 User's Guide (NUREG/CR-2326) shall be the minimum list of radionuclides included in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning.	There are 54 radionuclides identified in this list. In MACCS there are six additional radionuclides: Sr-92, Y-92, Y-93, Ba-139, La-141, and La-142. These are not critical for the PAG comparison calculation; the impact of the Sr, Y, Ba and La isotopes already included in the CRAC2 list is much greater, given their relative quantities, half-lives and dose conversion factors; therefore, the CRAC2 list is acceptable.	8
2.6.5.10	**Dose Conversion Factors**	**Dose Conversion Factors**	8
	External dose conversion factors (plume and ground exposure) shall be based on Kocher, D.C., "Dose Rate Conversion Factors for External Exposure to Photons, and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," Health Phys., Volume 38, pp. 543-621 (1980). Inhalation dose	Federal Guidance Report No. 11 is the document referenced by the May, 1992 revision of the PAG Manual. However, in this guide, external dose conversion factors are provided only for noble gases. The external dose conversion factors used in MACCS for NUREG-1150 calculations are referenced in NUREG/CR-4551 to the	8

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Paragraph No.	Requirement	Rationale	Rev.
	conversion factors shall be based on Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," Office of Radiation Programs, USEPA (1988).	specified Health Physics article. These are judged to be acceptable for the use described herein. The inhalation dose conversion factors provided in the guide are for a 50-year "lifetime" commitment, consistent with 2.6.5.8 above.	
2.6.5.11	**Plume Modeling**	**Plume Modeling**	8
	The model used to treat dispersion in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning shall be a straightline Gaussian plume. Plume centerline doses shall be reported. The values of σ_y and σ_z that are used to characterize the Gaussian plume expansion shall be based on Pasquill-Gifford curves. If the analytical model used in the analysis employs a uniform approximation of the expansion in the crosswind (y)direction (e.g., CRAC2), the final result shall be increased by an appropriate factor to provide centerline doses. In the case of CRAC2 (which employs a 3- σ_y "top hat" approximation of the cross-wind Gaussian distribution), the factor shall be 1.2. The initial σ_y shall be the building width divided by 4.3 if some other factor is used to determine the initial σ_y (e.g., a factor of 3 in CRAC2), and the building width specifiction shall be changed at the input level to compensate (e.g., the building width for CRAC2 shall be input as 70% of its actual value). The correlation for dispersion in the vertical direction (z) shall be the form $\sigma_z = ax^b + c$ where x is the distance the plume has traveled. The values for a, b and c shall be the fixed values in CRAC2. In the event a simpler form has been	The plume modeling in MACCS differs somewhat from that in CRAC2. The differences have been resolved as follows: • To demonstrate that the PAGs will not be exceeded within the exclusion area boundary (EAB) radius, the peak centerline value is the value that should be reported. To obtain this value, the CRAC2 results must be multiplied by a factor of 1.2. In addition, to compensate for the initially more disperse plume in CRAC2 (which results from setting the initial σ_y equal to building width/3 instead of building width/4.3), it is necessary to set the CRAC2 building width at the input level to 70% of its actual value. • In CRAC2, the expansion in the z-direction (vertical) is controlled by an expression for σ_z as a function of plume travel, x. The expression has the form	8

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	employed for calculational ease (e.g., $\sigma_Z = ax^b$ in MACCS), the coefficients shall be set to provide the same value of σ_Z at a site boundary of 0.5 mile and at a low population zone (LPZ) radius of two miles as would be calculated using the fixed values for a, b and c in CRAC2. Those values are as follows:	$\sigma_Z = ax^b + c$ with the constants fixed in the coding. In MACCS, a different correlation which does not use an additive constant ("c" term) has been employed, but only for the purpose of convenience. For specific radial intervals of interest, values of a and b can be defined to give the same values of X/Q as CRAC2 at the two specific radial distances that define the interval. This is what has been done in this methodology specification. The 0.5-mile site boundary and2-mile LPZ were chosen simply as typical radial distances.	

Stability	a	b
A	2.47E-4	2.118
B	0.078	1.085
C	0.144	0.911
D	0.368	0.6764
E	0.2517	0.6720
F	0.184	0.6546

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	The time base for plume meander for long duration releases shall be the fixed value in CRAC2, three minutes.	• For long release times (greater than a few minutes), plume meander becomes an important factor in determining peak centerline doses. In CRAC2, the time base for plume meander was fixed at 3 minutes; in MACCS, it is a user input with 10 minutes having been used in NUREG-1150 and appearing in the standard problem input file. The data base supporting the modeling of plume meander includes averaging times (i.e., the time base) of approximately 3 to 10 minutes. Since the important parameter for plume meander is the ratio of release duration to the time base and since the release duration being used in the PAG assessment is 10 hours, per 2.6.5.14, duration to time base is better approximated by using the low end of the averaging range (i.e., the fixed CRAC2 value of 3 minutes) than the high end.	

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.12	**Release Height and Energy of Release**	**Release Height and Energy of Release**	8
	The release height and energy of release assigned to the physically-based source term shall correspond to a cold, ground-level release for the purpose of calculating the dose.	Current severe accident analysis practice is to use release height and energy values that are consistent with the containment failure size/location or leak rate and associated thermodynamic conditions. However, for the ALWR physically-based source term, containment is intact, releases are not credited through a stack, and best estimate meteorology is used. Thus a cold, ground level release is appropriate.	8
2.6.5.13	**Duration of Exposure to Ground Contamination**	**Duration of Exposure to Ground Contamination**	8
	The duration of exposure to ground contamination shall be 24 hours from the start of release of fission products from the fuel.	The 24-hour period provides margin for ALWR accident detection, notification, and evacuation The 24-hour period is also consistent with the existing emergency planning basis.	8
2.6.5.14	**Duration of Release and Number of Plume Segments**	**Duration of Release and Number of Plume Segments**	8
	The release duration to be used in calculating doses for the ALWR physically-based source term shall be 10 hours if a single plume segment is used or 24 hours if multiple plume segments are used.	The CRAC2 code has a limit on release duration of 10 hours and can employ only a single plume. The MACCS code will accept a release duration greater than 10 hours and can employ multiple plumes (i.e., different source terms in succession), this capability being most useful when the character of the release to the environment abruptly changes in the course of an accident. This is not the case for the ALWR physically-based source term, where the difference in dose between a 10-hour release duration and a 24-hour release duration is only a few percent.	8

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.15	**Shielding Factors**	**Shielding Factors**	8
	Shielding factors shall be 0.75 for plume exposure and 0.33 for exposure to ground contamination.	The values given are those from NUREG-0396, Section F, "no immediate protective actions" and are consistent with the "normal activity" requirement of the PRA KAG.	8
2.6.5.16	**Breathing Rate and Inhalation Protection Factors**	**Breathing Rate and Inhalation Protection Factors**	8
	The breathing rate shall be 3.3 x 10-4m3/sec. For codes with provision for an inhalation protection factor, this value shall be set at 0.4. For codes without an inhalation protection factor, the breathing rate shall be reduced by a factor of 2.5.	The breathing rate identified in the May 1992 revision of the PAG Manual is the value specified. In the MACCS code, there is provision to reduce the inhalation dose by a factor to account for differences between the plume concentration and the concentration actually being breathed. NUREG/CR-4551 (one of the supporting documents for NUREG-1150) suggests an annual average value of 0.4 for normal activity (0.2 for active sheltering). The use of a "normal activity" inhalation protection factor is consistent with the requirements of the PRA KAG.	8
2.6.5.17	**Dry Deposition Velocity**	**Dry Deposition Velocity**	8
	The dry deposition velocity shall be 1.0 cm/sec for iodine and 0.1 cm/sec for other particulates.	These values are those of the May, 1992 revision of the PAG Manual. Current severe accident analysis practice is to use values of 1.0 cm/sec (NUREG-0396/CRAC2) to 0.3 cm/sec (NUREG-1150/MACCS); the PRA KAG does not establish a requirement for dry deposition velocity.	8
2.6.6	**Methodology for Performing Supplemental PRA**	**Methodology for Performing Supplemental PRA**	8
	The supplemental PRA shall be performed in accordance with the Volume II, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules (KAG) with the exception that the off-site dose exceedance limit is 1 rem, per Section 2.6.3 above. As specified in the KAG, both internal events	The KAG is the ALWR methodology for PRA evaluations. The KAG specifies that the PRA address internal events plus external events with the exception of seismic risk which is to be addressed by the seismic margin approach per Chapter 1, Section 2.5.3.4, of the	8

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	and external events shall be addressed as well as a comprehensive set of uncertainty and sensitivity studies.	URD. The KAG specifies quantitative assessments of uncertainties including propagation of distributions for Level 1 analyses,consideration of phenomenological uncertainties in Level 2 and 3 analyses, and sensitivity studies to assess the quantitative effect of particularly important uncertainties.	
	The required demonstration on the NRC Safety Goal Policy shall use the following methodology: • The ALWR reference site parameters in Annex B to the KAG shall be used. • No evacuation shall be assumed prior to 24 hours. Subsequent to 24 hours, the evacuation parametersof the KAG, Annex B, shall be used. • To demonstrate the NRC Safety Goal Policy quantitative objective for risk to an average individual (less than 0.1% of the risk from all other accidents), ALWR accident risk shall be less than 4x10-7 per person per year.	The numbers specified for risk comparisons are based upon recent data from the National Safety Council (Accident Facts, National Safety Council, 1988). The quantitative objective for latent cancer risks, which is also part of the NRC Safety Goal Policy, is not included in this required demonstration of Safety Goal compliance because, as noted in NUREG-1150, emergency response in close-in regions does not contribute substantially to differences in latent cancer risk. It is expected, however, that ALWRs would have no difficulty in meeting the latent cancer risk quantitative objective.	

Section A.2
ALWR URD Emergency Planning Criteria
(Volume III, Chapter 1, Section 2.6)

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Paragraph No.	Requirement	Rationale	Rev.
2.6	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING** ALWRs shall be designed to allow simplification and standardization of emergency planning. The Plant Designer shall perform an evaluation of the plant design against two ALWR emergency planning technical criteria prescribed below for containment performance and offsite dose. The methodology which is specified for demonstrating the criteria shall be utilized in this evaluation. The Plant Designer shall also perform a supplemental PRA evaluation in support of the evaluation against the two ALWR emergency planning criteria.	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING** Technical criteria and methodology are provided so as to specify what a Plant Designer seeking approval of ALWR emergency planning for a particular plant design must demonstrate. It is intended that these criteria and methodology form the technical basis for any necessary regulatory action on emergency planning for both the plume exposure pathway and the ingestion pathway. The criteria and methodology are intended to be used in an integrated manner and the criteria should not be applied without utilizing the methodology specified in this section. The criteria and methodology for containment performance and dose evaluation are primarily deterministic. The PRA evaluation is not a criterion itself but rather is intended to complement the two criteria. This is consistent with the NRC Severe Accident Policy which states that safety acceptability should be based on an approach which stresses deterministic engineering analysis, complemented by PRA. The requirements in this section are generally unique to emergency planning although the containment performance criterion draws heavily on containment performance requirements in other locations of the Utility Requirements Document.	5 8

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Paragraph No.	Requirement	Rationale	Rev.
2.6.1	**Containment Performance Criterion**	**Containment Performance Criterion**	8
	For ALWR emergency planning, the plant shall be provided with the capability to address severe accident containment challenges, including design features and characteristics to preclude core damage sequences which could bypass containment, and to withstand loads representing those associated with core damage sequences. The methodology in Section 2.6.4 below shall be used to evaluate that capability. The LOCA plus hydrogen load from Chapter 5, Section 6.6.2.2 should not be exceeded for a period of approximately 24 hours after the start of release of fission products from the fuel. Beyond approximately 24 hours, means for preventing uncontrolled fission product release from containment shall be provided in accordance with Chapter 5, Section 6.6.2.6.	While ALWR accident prevention design features make the possibility of core damage extremely remote, specifying the capability to address severe accident containment challenges, including avoiding containment bypass and withstanding loads which are expected to envelope best estimate pressure and temperatures associated with severe accident conditions, provides confidence that the containment can withstand a severe accident. Not exceeding LOCA plus hydrogen loads for approximately 24 hours provides low leakage for the period corresponding to the site boundary dose criterion. Even if a core damage event should occur, the ALWR Program considers that it is very likely that the ALWR containment would be able to meet appropriate limits for an indefinite time period, i.e., no containment overpressure would occur. This is based on LWR accident management capabilities and the TMI-2 accident experience which suggest that it is likely that core damage events will be recovered in-vessel, and on ALWR reactor cavity design features (e.g., debris spreading area, flooding of debris) which are designed to quench the ex-vessel debris. Nevertheless, for defense-in-depth purposes, a requirement has been specified for no uncontrolled release beyond approximately 24 hours to provide protection against long-term containment overpressure failure. Radioactive decay and removal of fission products in containment is such that a release at 16 hours, or even earlier depending on the plant design, would result in no acute health effects at the site boundary. Thus, the approximately 24-hour period provides significant margin	8

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		to that time at which the acute health effects dose threshold could be exceeded.	
2.6.2	**Dose Criterion**	**Dose Criterion**	8
	Dose at 0.5 mile shall be evaluated per the methodology in Section 2.6.5 below and shall be shown not to exceed the Protective Action Guides (PAGs) for a period of approximately 24 hours from the start of release of fission products from the fuel.	The PAGs are projected dose levels for evacuation (1 to 5 rem) which are specified by the Environmental Protection Agency (EPA) in a 1992 report as guidance for actions to protect the public in the early phase of a nuclear incident (i.e., the plume exposure pathway). As noted in NUREG-1338, based on experience for non-radiological emergencies, evacuations take from two to eight hours, including time to notify the public. Not exceeding the PAG for approximately 24 hours would provide significant margin for ALWR accident detection, notification, and evacuation, if necessary. A separate ingestion pathway requirement has not been specified since the ingestion exposure planning distance will be determined on a generic basis for all ALWRs as noted in Section 2.6.5 below.	8
2.6.3	**Supplemental PRA Evaluation**	**Supplemental PRA Evaluation**	5
	A PRA evaluation shall be performed per the methodology in Section 2.6.6 below to demonstrate that the following goals are met: • A mean core damage frequency ≤ 10-5/yr; • A cumulative frequency <10-6/yr for sequences resulting in greater than 1 rem total effective dose equivalent (TEDE) over 24 hours at the site boundary.	The requirement to perform the supplemental PRA evaluation and the associated goals are intended to demonstrate the integrated effectiveness of the two emergency planning criteria (Sections 2.6.1 and 2.6.2 above). The supplemental PRA also serves as a tool for the Plant Designer for refining and optimizing the design. Finally, the supplemental PRA will provide confidence to the NRC in the overall safety of the plant and in the margin to NRC guidelines on core damage frequency and large release. Given the guidance in the NRC Severe Accident	8

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Paragraph No.	Requirement	Rationale	Rev.
		Policy Statement, it is not intended that the PRA goals be made part of design certification or of any rulemaking on emergency planning.	
	In addition, it shall be demonstrated that ALWR designs are consistent with the prompt accident quantitative health objective of the NRC Safety Goal Policy with no credit for evacuation prior to 24 hours.	This requirement demonstrates that an acceptable level of radiological risk to the public, as defined by the prompt accident quantitative health objective of the NRC Safety Goal Policy, can be achieved with evacuation, which as noted in Section 2.6.2, can be accomplished with significant margin within 24 hours.	
2.6.4	**Methodology for Demonstrating Containment Performance Criterion**	**Methodology for Demonstrating Containment Performance Criterion**	5
	The Plant Designer shall demonstrate that the pressure and temperature loads associated with core damage sequences are no more limiting than the peak LOCA plus hydrogen loads of Chapter 5, Section 6.6.2.2. For plant designs meeting the requirements of Chapter 5, Section 6.6.2.1, the characteristics of the core damage sequences shall be as follows:	Chapter 5, Section 6.6.2.2, requires that the peak LOCA plus hydrogen loads not exceed applicable ASME limits. The loads associated with core damage sequences must therefore be no more limiting than the LOCA plus hydrogen loads.	8
	• Containment is isolated and otherwise intact (i.e., no bypass has occurred); • Reactor coolant system is depressurized to <100 psig; • Ample water is in the reactor cavity/lower drywell prior to or immediately upon vessel penetration for cooling ex-vessel core debris; • Containment heat removal is adequate; • BWR containments are inerted, and hydrogen control	Consistent with Chapter 5, Section 6.6.2, design characteristics and features are to be provided which address severe accident challenges, including bypass and loads from core damage sequences. An exhaustive set of severe accident challenges, regardless of the probability of occurrence of the challenge, have been addressed based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. The conclusion from the technical work in support of this requirement is that if core damage should occur, it will be into an intact containment with the RCS at low pressure and with containment systems	

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Paragraph No.	Requirement	Rationale	Rev.
	system is functioning.	functioning as designed.	
	Best estimate severe accident methods shall be utilized in evaluating the loads. Accepted industry computer codes such as MAAP shall be applied.	Best estimate methods are appropriate for the severe accident evaluation since the evaluation relates to matters beyond the design basis, i.e., the ALWR Safety Margin Basis, and since the ALWR plant features for addressing severe accident challenges significantly reduce the uncertainty in severe accident phenomena.	
	In accordance with Chapter 5, Section 6.6.2.3, functional sequence types from the PRA with mean frequencies greater than 10-7/yr shall be analyzed for containment response. Functional sequence types with mean frequency less than 10-7/yr shall be reported for discussion.	Evaluation of core damage sequence loads with sequence selection based on PRA provides additional assurance that risk-significant sequences are being considered.	
2.6.5	**Methodology for Demonstrating Dose Criterion**	**Methodology for Demonstrating Dose Criterion**	5
	The demonstration that the dose criterion is met shall utilize a physically-based source term release into an intact containment as defined in Chapter 5, Section 2.4.1. The methodology for the PAG dose evaluation shall consist of the following.	The physically-based source term is based on release and removal phenomena from actual core damage sequences and should be reasonably bounded for source terms from the probabilistically significant sequences. The intact containment is based on ALWR containment performance requirements which have been specified such that severe accident challenges to containment are effectively precluded or can be accommodated, thus providing integrity of the containment.	8
2.6.5.1	**Approach**	**Approach**	5
	A probabilistic dose (PD) method (e.g., CRAC2 or MACCS) shall be used.	A PD method is chosen for consistency with the basis for existing emergency planning and the fact that PD methods have provision for the particulate component of the source term and thus are an appropriate method for calculating PAG comparison doses. The use of CRAC2,	5

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Paragraph No.	Requirement	Rationale	Rev.
		MACCS, or another similar code is consistent with current level 3 PRA evaluations and ALWR PRA Key Assumptions and Groundrules (KAG).	
2.6.5.2	**Meteorological Database**	**Meteorological Database**	5
	The meteorological database shall be that provided in Annex B to Appendix A to Chapter 1 of the URD.	This meteorological database is that provided in the PRA KAG. It is an actual site meteorological database for which the RG 1.145 two-hour Exclusion Area Boundary X/Q is estimated to be greater than the X/Q for 80 to 90 percent of U.S. operating sites.	5
2.6.5.3	**Direction-Dependent vs. Direction-Independent**	**Direction-Dependent vs. Direction-Independent**	5
	The dose calculation shall be direction-independent.	The calculations supporting existing emergency planning are direction-independent, i.e., the frequency of exceeding given dose levels is provided independent of direction. The NRC safety goals use a direction-independent approach as well. The use of a direction-independent approach is also consistent with the methods to be used in preparing the complementary cumulative distribution function (CCDF) for the exceedance frequency of off-site doses at the site boundary required by the PRA KAG.	5
2.6.5.4	**Statistical Measure of Dose to be Compared to 1 Rem PAGs**	**Statistical Measure of Dose to be Compared to 1 Rem PAGs**	5
	The dose to be compared to the 1 rem PAG for ALWR emergency planning shall be the median dose.	Existing emergency planning was based in part on establishing that "most" core melt accidents would not exceed the PAG. There are two sources of variability in determining the meaning of "most" in the situation for existing emergency planning (i.e., NUREG 0396): the	8

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		magnitude of the source term, and the meteorology. A similar approach is used here for ALWR emergency planning. Median dose (i.e., 50th percentile meteorology) together with the physically-based source term, which tends to bound the source term expected for nearly all core melt accidents in an ALWR, assures that the dose from most core melt accidents will not exceed 1 rem.	
2.6.5.5	**Statistical Measure of Dose to Compared to 5 Rem PAG**	**Statistical Measure of Dose to Compared to 5 Rem PAG**	8
	The dose to be compared to the 5 rem PAG shall be the 90th percentile dose.	More extreme (e.g., very stable atmospheric conditions, low wind speed) meteorology could cause higher doses for a given source term. While doses exceeding 1 rem would not be expected as noted above, a 5 rem limit has been specified for 90th percentile meteorology in order to address more extreme meteorological conditions. A 5 rem limit for such conditions is considered reasonable on several grounds. First, ICRP 63 recommends a dose limit for evacuation no lower than 50 mSv (i.e., 5 rem). Second, under stable, low wind speed conditions, the plume is concentrated (only about 100 feet wide at 0.5 mile) and is moving slowly, so the need for rapid evacuation would be quite limited. Finally, 5 rem is the upper end of the 1 to 5 rem range recommended by EPA and thus is a reasonable limit for emergency planning purposes under low probability weather conditions.	8
2.6.5.6	**Whole Body Dose vs. Effective Dose Equivalent**	**Whole Body Dose vs. Effective Dose Equivalent**	8
	The dose to be calculated is the sum of the effective dose equivalent (EDE) resulting from exposure to external sources (cloud shine and ground shine) and the committed effective dose equivalent (CEDE) from plume inhalation.	The May 1992 revision to Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (PAG Manual) calls for the use of EDE as the basis for determining off-site doses in relation to the 1 to 5 rem	8

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		PAG. MACCS already employs this concept, as does the current 10CFR20. A separate thyroid dose limit is unnecessary since the EDE includes the organ weighted contribution from thyroid exposure. Not specifying a separate thyroid dose limit is also consistent with the recent NRC 10CFR 50/100 rule change which specifies EDE as defined in Section 2.6.5.6. A separate ingestion exposure pathway requirement has not been specified since the ingestion exposure planning distance will be determined, using the May, 1992 PAG Manual guidelines, on a generic basis for all ALWRs. This will be accomplished by assuming that the 0.5 mile dose is equal to the PAG (i.e., the EDE limit for plume exposure), determining the maximum iodine contribution to this dose, and using this maximum iodine release as the basis for calculating the distance at which the ingestion dose equals the controlling ingestion pathway PAG (i.e., a projected infant thyroid dose from cow's milk of 1.5 rem on a preventative basis and 15 rem as a basis for emergency contamination).	
2.6.5.7	**Inclusion of Organic Iodide in the PAG Calculation**	**Inclusion of Organic Iodide in the PAG Calculation**	8
	In calculating doses for comparison with the PAG values to justify ALWR emergency planning, the contribution from organic iodide can be neglected.	The I and HI are quite reactive and are likely to undergo natural deposition as rapidly (or more rapidly) than the particulate. Given that pH is controlled as specified in the Utility Requirements Document, the actual dose contribution from organic iodide is expected to be very small (a few percent of thyroid dose) and thus can be omitted from the dose calculation.	8

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2.6.5.8	**Dose Commitment**	**Dose Commitment**	5
	A dose commitment of 50 years shall be used for TEDE from plume inhalation.	In the May 1992 revision of the PAG Manual, plume inhalation dose commitment is assumed to be the "lifetime". It is judged that a 50-year commitment is adequate on a generic basis to fulfill that requirement; it is also the duration used in the current 10CFR20. This differs from the PRA as specified in the KAG where the intent is to compare calculated doses to the 25 rem threshold for acute health effects (based on the current 25 rem whole body requirement in 10CFR100). It also differs from NUREG 0396 which utilizes one year commitment for inhalation.	8
2.6.5.9	**Radionuclides to be Included**	**Radionuclides to be Included**	5
	The radionuclides identified in Table II-2 of the CRAC2 User's Guide (NUREG/CR-2326) shall be the minimum list of radionuclides included in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning.	There are 54 radionuclides identified in this list. In MACCS there are six additional radionuclides: Sr-92, Y-92, Y-93, Ba-139, La-141, and La-142. These are not critical for the PAG comparison calculation; the impact of the Sr, Y, Ba and La isotopes already included in the CRAC2 list is much greater, given their relative quantities, half-lives and dose conversion factors; therefore, the CRAC2 list is acceptable.	5
2.6.5.10	**Dose Conversion Factors**	**Dose Conversion Factors**	5
	External dose conversion factors (plume and ground exposure) shall be based on Kocher, D.C., "Dose Rate Conversion Factors for External Exposure to Photons, and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," Health Phys., Volume 38, pp. 543-621 (1980). Inhalation dose	Federal Guidance Report No. 11 is the document referenced by the May, 1992 revision of the PAG Manual. However, in this guide, external dose conversion factors are provided only for noble gases. The external dose conversion factors used in MACCS for NUREG-1150 calculations are referenced in NUREG/CR-4551 to the	8

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	conversion factors shall be based on Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," Office of Radiation Programs, USEPA (1988).	specified Health Physics article. These are judged to be acceptable for the use described herein. The inhalation dose conversion factors provided in the guide are for a 50-year "lifetime" commitment, consistent with 2.6.5.8 above.	
2.6.5.11	**Plume Modeling**	**Plume Modeling**	5
	The model used to treat dispersion in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning shall be a straightline Gaussian plume. Plume centerline doses shall be reported. The values of σ_y and σ_z that are used to characterize the Gaussian plume expansion shall be based on Pasquill-Gifford curves. If the analytical model used in the analysis employs a uniform approximation of the expansion in the crosswind (y)direction (e.g., CRAC2), the final result shall be increased by an appropriate factor to provide centerline doses. In the case of CRAC2 (which employs a 3- σ_y "top hat" approximation of the cross-wind Gaussian distribution), the factor shall be 1.2. The initial σ_y shall be the building width divided by 4.3 if some other factor is used to determine the initial σ_y (e.g., a factor of 3 in CRAC2), and the building width specifiction shall be changed at the input level to compensate (e.g., the building width for CRAC2 shall be input as 70% of its actual value).	The plume modeling in MACCS differs somewhat from that in CRAC2. The differences have been resolved as follows: • To demonstrate that the PAGs will not be exceeded within the exclusion area boundary (EAB) radius, the peak centerline value is the value that should be reported. To obtain this value, the CRAC2 results must be multiplied by a factor of 1.2. In addition, to compensate for the initially more disperse plume in CRAC2 (which results from setting the initial σ_y equal to building width/3 instead of building width/4.3), it is necessary to set the CRAC2 building width at the input level to 70% of its actual value.	5
	The correlation for dispersion in the vertical direction (z) shall be the form $\sigma_z = ax^b + c$ where x is the distance the plume has traveled. The values for a, b and c shall be the fixed values in CRAC2. In the event a simpler form has been	• In CRAC2, the expansion in the z-direction (vertical) is controlled by an expression for σ_z as a function of plume travel, x. The expression has the form	

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	employed for calculational ease (e.g., $\sigma_Z = ax^b$ in MACCS), the coefficients shall be set to provide the same value of σ_Z at a site boundary of 0.5 mile and at a low population zone (LPZ) radius of two miles as would be calculated using the fixed values for a, b and c in CRAC2. Those values are as follows:	$\sigma_Z = ax^b + c$ with the constants fixed in the coding. In MACCS, a different correlation which does not use an additive constant ("c" term) has been employed, but only for the purpose of convenience. For specific radial intervals of interest, values of a and b can be defined to give the same values of X/Q as CRAC2 at the two specific radial distances that define the interval. This is what has been done in this methodology specification. The 0.5-mile site boundary and2-mile LPZ were chosen simply as typical radial distances.	

Stability	a	b
A	2.47E-4	2.118
B	0.078	1.085
C	0.144	0.911
D	0.368	0.6764
E	0.2517	0.6720
F	0.184	0.6546

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	The time base for plume meander for long duration releases shall be the fixed value in CRAC2, three minutes.	• For long release times (greater than a few minutes), plume meander becomes an important factor in determining peak centerline doses. In CRAC2, the time base for plume meander was fixed at 3 minutes; in MACCS, it is a user input with 10 minutes having been used in NUREG-1150 and appearing in the standard problem input file. The data base supporting the modeling of plume meander includes averaging times (i.e., the time base) of approximately 3 to 10 minutes. Since the important parameter for plume meander is the ratio of release duration to the time base and since the release duration being used in the PAG assessment is 10 hours, per 2.6.5.14, duration to time base is better approximated by using the low end of the averaging range (i.e., the fixed CRAC2 value of 3 minutes) than the high end.	

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.12	**Release Height and Energy of Release**	**Release Height and Energy of Release**	5
	The release height and energy of release assigned to the physically-based source term shall correspond to a cold, ground-level release for the purpose of calculating the dose.	Current severe accident analysis practice is to use release height and energy values that are consistent with the containment failure size/location or leak rate and associated thermodynamic conditions. However, for the ALWR physically-based source term, containment is intact, releases are not credited through a stack, and best estimate meteorology is used. Thus a cold, ground level release is appropriate.	5
2.6.5.13	**Duration of Exposure to Ground Contamination**	**Duration of Exposure to Ground Contamination**	5
	The duration of exposure to ground contamination shall be 24 hours from the start of release of fission products from the fuel.	The 24-hour period provides margin for ALWR accident detection, notification, and evacuation The 24-hour period is also consistent with the existing emergency planning basis.	5
2.6.5.14	**Duration of Release and Number of Plume Segments**	**Duration of Release and Number of Plume Segments**	5
	The release duration to be used in calculating doses for the ALWR physically-based source term shall be 10 hours if a single plume segment is used or 24 hours if multiple plume segments are used.	The CRAC2 code has a limit on release duration of 10 hours and can employ only a single plume. The MACCS code will accept a release duration greater than 10 hours and can employ multiple plumes (i.e., different source terms in succession), this capability being most useful when the character of the release to the environment abruptly changes in the course of an accident. This is not the case for the ALWR physically-based source term, where the difference in dose between a 10-hour release duration and a 24-hour release duration is only a few percent.	5

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Paragraph No.	Requirement	Rationale	Rev.
2.6.5.15	**Shielding Factors**	**Shielding Factors**	5
	Shielding factors shall be 0.75 for plume exposure and 0.33 for exposure to ground contamination.	The values given are those from NUREG-0396, Section F, "no immediate protective actions" and are consistent with the "normal activity" requirement of the PRA KAG.	5
2.6.5.16	**Breathing Rate and Inhalation Protection Factors**	**Breathing Rate and Inhalation Protection Factors**	8
	The breathing rate shall be 3.3 x 10-4m3/sec. For codes with provision for an inhalation protection factor, this value shall be set at 0.4. For codes without an inhalation protection factor, the breathing rate shall be reduced by a factor of 2.5.	The breathing rate identified in the May 1992 revision of the PAG Manual is the value specified. In the MACCS code, there is provision to reduce the inhalation dose by a factor to account for differences between the plume concentration and the concentration actually being breathed. NUREG/CR-4551 (one of the supporting documents for NUREG-1150) suggests an annual average value of 0.4 for normal activity (0.2 for active sheltering). The use of a "normal activity" inhalation protection factor is consistent with the requirements of the PRA KAG.	8
2.6.5.17	**Dry Deposition Velocity**	**Dry Deposition Velocity**	8
	The dry deposition velocity shall be 1.0 cm/sec for iodine and 0.1 cm/sec for other particulates.	These values are those of the May, 1992 revision of the PAG Manual. Current severe accident analysis practice is to use values of 1.0 cm/sec (NUREG-0396/CRAC2) to 0.3 cm/sec (NUREG-1150/MACCS); the PRA KAG does not establish a requirement for dry deposition velocity.	8
2.6.6	**Methodology for Performing Supplemental PRA**	**Methodology for Performing Supplemental PRA**	8
	The supplemental PRA shall be performed in accordance with the Volume II, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules (KAG) with the exception that the off-site dose exceedance limit is 1 rem, per Section 2.6.3 above. As specified in the KAG, both internal events	The KAG is the ALWR methodology for PRA evaluations. The KAG specifies that the PRA address internal events plus external events with the exception of seismic risk which is to be addressed by the seismic margin approach per Chapter 1, Section 2.5.3.4, of the	8

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	and external events shall be addressed as well as a comprehensive set of uncertainty and sensitivity studies.	URD. The KAG specifies quantitative assessments of uncertainties including propagation of distributions for Level 1 analyses,consideration of phenomenological uncertainties in Level 2 and 3 analyses, and sensitivity studies to assess the quantitative effect of particularly important uncertainties.	
	The required demonstration on the NRC Safety Goal Policy shall use the following methodology: • The ALWR reference site parameters in Annex B to the KAG shall be used. • No evacuation shall be assumed prior to 24 hours. Subsequent to 24 hours, the evacuation parametersof the KAG, Annex B, shall be used. • To demonstrate the NRC Safety Goal Policy quantitative objective for risk to an average individual (less than 0.1% of the risk from all other accidents), ALWR accident risk shall be less than 4x10-7 per person per year.	The numbers specified for risk comparisons are based upon recent data from the National Safety Council (Accident Facts, National Safety Council, 1988). The quantitative objective for latent cancer risks, which is also part of the NRC Safety Goal Policy, is not included in this required demonstration of Safety Goal compliance because, as noted in NUREG-1150, emergency response in close-in regions does not contribute substantially to differences in latent cancer risk. It is expected, however, that ALWRs would have no difficulty in meeting the latent cancer risk quantitative objective.	

Section A.3
ALWR URD Containment Performance Requirements
(Volume II, Chapter 5, Section 6.6.2)

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Paragraph No.	Requirement	Rationale	Rev.
6.6	**SEVERE ACCIDENT REQUIREMENTS**	**SEVERE ACCIDENT REQUIREMENTS**	0
6.6.1	**Definition**	**Definition**	0
6.6.1.1	**Scope**	**Scope**	1
	Severe accident requirements for all mitigation functions common to the BWR and PWR are specified in this section. This section consists of four main subsections: Containment Performance, Cavity/Pedestal-Drywell Configuration, Containment Heat Removal/Fission Product Control, and Severe Accident Equipment Requirements.		1
6.6.2	**Containment Performance**	**Containment Performance**	4
	The ALWR containment performance requirement shall consist of a number of elements as specified below. The initial element shall include a matrix of plant features to address a comprehensive set of containment challenges from severe accidents. This matrix approach, together with the other elements of containment performance, provide high assurance of containment integrity and low off-site dose in the event of a severe accident.	The elements below comprise a deterministic approach to containment performance. The deterministic approach is supplemented by the PRA requirement to meet the ALWR goal of 10-6, 25 rem. This is consistent with NRC Severe Accident Policy Statement guidance and provides the complete set of containment performance requirements that are necessary to address severe accidents. The combined set of deterministic/PRA requirements satisfies the Commission response to SECY-90-016 for a deterministic alternative which provides comparable mitigation capability to the conditional containment failure probability (CCFP) of 0.1 but does not discourage improvements in core damage prevention.	4

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Paragraph No.	Requirement	Rationale	Rev.
6.6.2.1	**Plant Features to Minimize Containment Challenges**	**Plant Features to Minimize Containment Challenges**	8
	The plant shall include design characteristics and features to address a comprehensive set of severe accident challenges to the containment. Design characteristics and features shall include:	Design characteristics and features to address a comprehensive set of severe accident containment challenges are necessary to provide severe accident protection for the ALWR consistent with the NRC Severe Accident Policy, ALWR safety policy, and to meet the ALWR requirements.	8
	• Features to provide reliable shutdown of the reactor by rod insertion, e.g., Chapter 4, Section 5.3 (BWR) and Chapter 4, Section 8.2 (PWR) as well as diverse reactivity control capability in the form of SLC, Section 4.6 (BWR) and SIS, Section 5.4 (PWR).	• Reliable reactivity control, through rod insertion and the capability to accommodate failure to scram in the form of diverse means of reactivity insertion, limits the challenges associated with ATWS.	
	• Features to reliably depressurize the reactor pressure vessel as described in Sections 4.5.3 (BWR) and 5.5 (PWR) or limit the magnitude of the pressure rise in containment resulting from ejection of core debris at elevated RPV pressure, as described in Section 6.6.3.1.	• A reliable depressurization system minimizes the probability of high pressure core ~~melts~~ accidents with subsequent potential for direct ontainment heating. Cavity configuration can also limit the magnitude of containment pressure rise from high pressure melt ejection.	
	• Features to handle the pressure and temperature resulting from generation of combustible gases as covered in Section 6.5.	• Features that control combustion and/or prevent detonation of hydrogen, eliminate a major threat to containment integrity following a severe accident.	
	• Features that provide reliable containment cooling for decay heat removal as described in sections 4.5.3 (BWR) and 8.3 (PWR).	• Long term containment cooling is required to maintain containment pressure within design limits.	
	• Features to limit the generation of non-condensible gases as a result of corium-concrete interaction as covered in Section 6.6.3.	• Containment integrity is challenged as a result of pressure buildup from production of non-condensible gases following corium-concrete interaction. Preventing or limiting this event enhances containment performance.	

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	• Features to assure containment integrity including isolation and precluding steam generator tube rupture and other containment bypass scenarios, e.g., Chapter 3, Section 2, and Chapter 5, Sections 4.2, 4.3, 4.4, 4.5, 6.2, and 7.2, for the BWR and Chapter 3, Sections 2 and 4, and Chapter 5, Sections 5.2, 5.3, 5.5, 6.2, and 8.2 for the PWR.	• Challenges to containment integrity which result from failures which occur independent of or coincident with core damage (e.g., containment bypass events) must be avoided.	
6.6.2.2	**Containment Performance Structural Evaluation**	**Containment Performance Structural Evaluation**	8
	The Plant Designer shall demonstrate that the containment system's pressure boundary, when subjected to the pressure and temperature loads from LOCA plus hydrogen described below,combined with the appropriate dead loads, meets the following ASME Code, Section III criteria: • For Class MC free standing steel vessels and for the steel portions of Class CC reinforced concrete vessels which are not backed by concrete, the following requirements shall apply: – Paragraph NE-3221, Service Level C Limits on stress intensity values. – For regions of ellipsoidal or torispherical shell surfaces of containment, the allowable compressive stress due to internal pressure shall not exceed 60 percent of the value of critical buckling stress determined by one of the methods given in ASME Subparagraph 3222.1 (a).	The ASME Section III Code referenced structural integrity criteria satisfies the intended minimum requirements of 10CFR50.34(f)(3)(v). Also, any gross distortions and subsequent large strains in pressure boundary material due to potential shell buckling modes is precluded. The LDB requirements of Section 2.4.1 are expected to be limiting for inerted containments while the SMB requirements are expected to be limiting for containments which are not inerted.	8

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	• Compressive stress in ellipsoidal or torispherical shell heads due to internal pressure loading is a localized stress field which does not represent a challenge to overall containment stability; thus a lower factor of safety against buckling than otherwise permitted by Subparagraph 3222.2 is appropriate in these regions. The value of 60 percent of the critical buckling stress results in a safety factor of 1.67, which is consistent with the requirements of Code Case N-284 for local buckling. • For the steel liner portions of Class CC vessels which are backed by concrete, the factored load limits on liner strains established in Subsubarticle CC-3720 shall apply. • For those portions of other ASME Code class components which also constitute a portion of the containment systems pressure boundary, the corresponding ASME Section III Service Level C Limits shall apply.		
6.6.2.2.1	**Inerted Pressure Suppression Containments**	**Inerted Pressure Suppression Containments**	8
	The analysis of LOCA plus hydrogen loads shall assume: • Pool temperature equal to the peak temperature associated with the DBA LOCA within 24 hours from the accident initiation. • All nitrogen in the drywell is located in the wetwell airspace.	The assumptions maximize the pressure and temperature loads in the containment in the performance of the 10CFR50.34(f)(3)(v) analysis.	8

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	• The total hydrogen equivalent to 100% active fuel cladding metal water reaction is located in the wetwell airspace.		
6.6.2.2.2	**Non-inerted Containments**	**Non-inerted Containments**	8
	The analysis of LOCA plus hydrogen loads shall assume: • Peak pressure associated with the DBA LOCA; • Accumulation of hydrogen associated with 75% active fuel cladding metal water reaction; • Adiabatic isochoric complete combustion of this accumulated quantity of hydrogen. If containment is found to be steam inerted at the peak DBA pressure, then combustion shall be assumed to occur at the time steam condensation reduces the mole fraction of steam to combustible levels (~50% mole fraction steam).	The Licensing Design Basis analysis required by 10CFR50.34(f)(3)(v) would credit a hydrogen control system as hydrogen is generated. The Safety Margin Basis analysis requirement contained in this section postulates the peak DBA pressure and a realistic upper bound to total hydrogen concentration, i.e., that associated with 75% active clad oxidation, before crediting a hydrogen control system or ignition sources. This yields a higher peak pressure than that required by 10CFR50.34(f)(3)(v). Burning is assumed to occur at the highest potential containment pressure if inerting initially precludes combustion.	8
6.6.2.3	**Severe Accident Sequence Selection for Reporting Containment Response**	**Severe Accident Sequence Selection for Reporting Containment Response**	8
	The Plant Designer shall report containment performance during severe accidents. Analysis of severe accident sequences shall be performed to	The primary means of addressing severe accident containment challenges is the deterministic matrix of design characteristics and features of Section 6.6.2.1 and the deterministic analyses of	8

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	confirm that the containment provides substantial margin with respect to severe accident challenges. Accident sequences from the PRA shall be selected for analysis of containment performance. PRA sequences shall be grouped into functional sequence types for the purpose of determining the mean total frequency of all accident sequences with approximately the same type of challenge. The sequence types could be those resulting from the failure of any one of the following functions:	Section 6.6.2.2. This deterministic approach addresses an exhaustive list of containment challenges, regardless of probability. The probabilistic requirement of Section 6.6.2.3 complements the deterministic approach as required in the NRC Severe Accident Policy. The difficulty of assigning accurate numerical estimates notwithstanding, use of PRA in this manner provides valuable design insights and added confidence that containment margin exists for severe accidents and that important risk contributors have been addressed.	
	• Reactivity insertion; • RCS depressurization; • Core or core debris coolant inventory control; • Containment pressure/temperature control; • Combustible gas control; • Containment isolation and containment bypass control; • Other functions, the failure of which could lead to containment challenge.	This set of functions is considered necessary to assure containment integrity using a similar approach as described in the report, "Passive ALWR Severe Accident Containment Performance Requirements," January, 1992. This report concludes that the only potentially significant severe accident challenges to a standard ALWR plant design which implements the provisions in the Requirements Document are those associated with core damage events that occur into an intact containment with the RCS at low pressure with containment systems functioning as designed.	
	Functional sequence types with mean frequency greater than approximately 10-7/yr shall be analyzed for containment response. The determination of sequence types with mean frequency greater than 10-7/yr shall utilize sequence information from the respective plant specific PRAs.	The approximately 10-7/yr threshold for functional sequence types to be analyzed for containment response is consistent with the NUREG-1420 10-7/yr limit for insignificant risks and is consistent with Standard Review Plan guidance to evaluate potential accidents from hazards in the plant vicinity which exceed approximately 10-7/yr. Also, NUREG-1150 uses a cutoff of 10-7/yr for accident progression analysis. NUREG-	

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	The determination shall utilize propagation of uncertainty distributions to determine the mean frequency and shall utilize sensitivity studies as appropriate, shall include internal events, and shall either include, or provide a justification based on design, siting, or low frequency of occurrence, for exclusion of external events.	1338 stated that any sequence appearing to have a frequency down to about 10-7/yr will be examined from the standpoint of residual risk. Finally, consideration of functional sequence types greater than approximately 10-7/yr provides assurance that the cumulative effects of such sequence types will not exceed the 10-6/yr probability goal for off-site consequences.	
	Functional sequence types with frequency less than 10-7 per year shall be reported for discussion: • Identifying the design features and operating characteristics credited to reach this low frequency; • Singling out the frequency of those sequence types which may result in early containment failure.	The plant specific PRA is the most complete and accurate source of accident sequence information. The specified treatment of uncertainties and external events is consistent with the KAG. The purpose of this requirement is to assure that there is understanding of those features designed to preclude containment failure resulting from a severe accident. It is also expected to show that those phenomena which could lead to exceeding the capacity of containment early in a postulated severe accident event are a small fraction of the ALWR PRA goals for core damage frequency and consequences.	
	The loads resulting from any analyzed functional sequence types shall be no more limiting than the peak LOCA plus hydrogen loads of Section 6.6.2.2 for approximately 24 hours after the start of fission product release from the fuel.	If the loads resulting from the analyzed severe accident sequence types are enveloped by the conditions determined for LOCA plus hydrogen in accordance with Section 6.6.2.2, the comparison of these severe accident loads may be made directly with the LOCA plus hydrogen loads. In the event the loads exceed those determined in accordance with Section 6.2.2.2, it is expected the Plant Designer will be able to demonstrate that the containment still meets the functional criteria for Service Level C or Unity Factored Load as permitted by 10CFR50.34(f)(3)(v) and provide confidence that the structural integrity and leak tightness of the passive plant containment will be maintained following a severe accident. Should any functional sequence type selected for analysis result	

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		in loads which exceed the functional criteria for Service Level C or Unity Load permitted by 10CFR50.34(f)(3)(v) or result in containment bypass, the Plant Designer should identify the reasons for the high loads or the bypass and explain why the accident sequence frequencies cannot be further reduced, and provide recommendations for an alternate basis on which confirmation of acceptable containment performance can be justified.	
6.6.2.4	**Containment Ultimate Capacity Analysis**	**Containment Ultimate Capacity Analysis**	1
	The Plant Designer shall perform an analysis to determine the ultimate structural capability of the containment. For steel containments, the ultimate capacity shall be defined as the pressure and temperature loadings which correspond to the collapse load defined by the method detailed in paragraph II-1430 of the ASME Code, Section III, Appendix II. For concrete containments, the ultimate structural capacity shall be defined as the pressure and temperature loading which produces liner plate strains equal to the liner strain limits of the ASME Code Section III, Subarticle CC-3720 for the Factored Load Category. The analysis shall consider the penetrations and their interaction with the containment, the shield building, and other structures internal or external to the containment, which might cause localized failure prior to the limit load for the overall pressure boundary. Results from testing of prototype details or models of prototype details may be used to augment such analyses. The failure mode associated with the ultimate structural capacity shall be identified.	An analysis of containment ultimate capacity is required by Standard Review Plans 3.8.1 and 3.8.2, including the determination of pressure retaining capacity of localized areas. The failure analysis criteria included here are identical to or more conservative than those developed during NRC/IDCOR issue resolution (see ARSAP Technical Task 2.3 report) or are more realistically based on recent experimental tests for concrete containments by Sandia National Laboratories. These tests have indicated that concrete containment capability may be limited by leakage resulting from liner plate tears. EPRI report NP-6261 describes computer modeling techniques used to predict the failure mode of the scale model concrete containment tested by Sandia. Interaction of the containment penetrations with the shield building or other structures may produce leakage paths.	1

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Paragraph No.	Requirement	Rationale	Rev.
6.6.2.5	**Cavity/Lower Drywell Structural Capability**	**Cavity/Lower Drywell Structural Capability**	1
	The Plant Designer shall evaluate the reactor cavity/lower drywell for loads rsulting from localized pressure due to a low pressure core melt ejection and effects from core debris material/water interaction to demonstrate that the reactor vessel will not lose overall structural support.	This requirement assures structural adequacy of the reactor cavity/lower drywell for local steam pressure loads resulting from low pressure core melt ejection and core debris water interaction.	1
6.6.2.6	**Long-term Containment Overpressure Protection**	**Long-term Containment Overpressure Protection**	8
	Protection of the containment for long-term overpressurization shall be provided. If significant time for recovery of core cooling and containment heat removal is available (e.g., 48 hours) prior to reaching Service Level C (steel containment structures) or Unity Factored Load (concrete containment structures) then overpressure protection may be provided simply by the size and strength of the containment. As an alternative, a system for pressure relief may be provided. In pressure suppression containments, pressure relief shall be provided preferentially from the suppression chamber air space to take advantage of scrubbing through the suppression pool. For other containment designs, pressure relief may be provided, utilizing available systems and equipment, provided that a significant decrease in the residual public risk results.	Containment overpressure protection provides additional defense-in-depth to protect the containment from catastrophic failure. The representative time for repair and recovery (48 hours) is drawn from the deterministic criteria of SECY-90-016 with respect to containment performance and takes advantage of the size of the containment in providing significant time for recovery of core cooling and decay heat removal systems and equipment. Compliance with this time criterion is demonstrated using "best estimate" analysis methodologies, including realistic assumptions. Controlled pressure relief is consistent with current BWR EPGs and provides a simple means of additional protection against low probability end-of-spectrum events such as loss of decay heat removal and non-condensible gas generation from core concrete interaction beyond that already provided in the requirements. For larger containments, the risk addressed by overpressure design features is sufficiently low that additional plant complexity is not warranted. Nevertheless, any simple overpressure protection provisions should be justified on a risk basis.	8

Section A.4
ALWR URD Containment Performance Requirements
(Volume III, Chapter 5, Section 6.6.2)

VOLUME III, CHAPTER 5: ENGINEERED SAFETY SYSTEMS

Paragraph No.	Requirement	Rationale	Rev.
6.6	**SEVERE ACCIDENT REQUIREMENTS**		0
6.6.1	**Definition**		0
6.6.1.1	**Scope**		0
	Severe accident requirements for all mitigation functions common to the BWR and PWR are specified in this section. This section consists of four main subsections: Containment Performance, Cavity/Pedestal-Drywell Configuration, Containment Heat Removal/Fission Product Control, and Severe Accident Equipment Requirements.		0
6.6.2	**Containment Performance**	**Containment Performance**	0
	The ALWR containment performance requirement shall consist of a number of elements as specified below. The initial element shall include a matrix of plant features to address a comprehensive set of containment challenges from severe accidents. This matrix approach, together with the other elements of containment performance, provide high assurance of containment integrity and low off-site dose in the event of a severe accident.	The elements below comprise a deterministic approach to containment performance. The deterministic approach is supplemented by the PRA requirement to meet the ALWR goal of 10-6, 25 rem. This is consistent with NRC Severe Accident Policy Statement guidance and provides the complete set of containment performance requirements that are necessary to address severe accidents. The combined set of deterministic/PRA requirements satisfies the Commission response to SECY-90-016 for a deterministic alternative which provides comparable mitigation capability to the conditional containment failure probability (CCFP) of 0.1 but does not discourage improvements in core damage prevention.	8

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Paragraph No.	Requirement	Rationale	Rev.
6.6.2.1	**Plant Features to Minimize Containment Challenges**	**Plant Features to Minimize Containment Challenges**	8
	The plant shall include design characteristics and features to address a comprehensive set of severe accident challenges to the containment. Design characteristics and features shall include:	Design characteristics and features to address a comprehensive set of severe accident containment challenges are necessary to provide severe accident protection for the ALWR consistent with the NRC Severe Accident Policy, ALWR safety policy, and to meet the ALWR requirements.	8
	• Features to provide reliable shutdown of the reactor by rod insertion, e.g., Chapter 4, Section 5.3 (BWR) and Chapter 4, Section 8.2 (PWR) as well as diverse reactivity control capability in the form of SLC, Section 4.6 (BWR) and SIS, Section 5.4 (PWR).	• Reliable reactivity control, through rod insertion and the capability to accommodate failure to scram in the form of diverse means of reactivity insertion, limits the challenges associated with ATWS.	
	• Features to reliably depressurize the reactor pressure vessel as described in Sections 4.5.3 (BWR) and 5.5 (PWR) or limit the magnitude of the pressure rise in containment resulting from ejection of core debris at elevated RPV pressure, as described in Section 6.6.3.1.	• A reliable depressurization system minimizes the probability of high pressure core ~~melts~~ accidents with subsequent potential for direct ontainment heating. Cavity configuration can also limit the magnitude of containment pressure rise from high pressure melt ejection.	
	• Features to handle the pressure and temperature resulting from generation of combustible gases as covered in Section 6.5.	• Features that control combustion and/or prevent detonation of hydrogen, eliminate a major threat to containment integrity following a severe accident.	
	• Features that provide reliable containment cooling for decay heat removal as described in sections 4.5.3 (BWR) and 8.3 (PWR).	• Long term containment cooling is required to maintain containment pressure within design limits.	
	• Features to limit the generation of non-condensible gases as a result of corium-concrete interaction as covered in Section 6.6.3.	• Containment integrity is challenged as a result of pressure buildup from production of non-condensible gases following corium-concrete interaction. Preventing or limiting this event enhances containment performance.	

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Paragraph No.	Requirement	Rationale	Rev.
	• Features to assure containment integrity including isolation and precluding steam generator tube rupture and other containment bypass scenarios, e.g., Chapter 3, Section 2, and Chapter 5, Sections 4.3, 6.2, and 7.2, for the BWR and Chapter 3, Sections 2 and 4, and Chapter 5, Sections 5.3 and 6.2 for the PWR.	• Challenges to containment integrity which result from failures which occur independent of or coincident with core damage (e.g., containment bypass events) must be avoided.	
6.6.2.2	**Containment Performance Structural Evaluation**	**Containment Performance Structural Evaluation**	5
	The Plant Designer shall demonstrate that the containment system's pressure boundary, when subjected to the pressure and temperature loads from LOCA plus hydrogen described below,combined with the appropriate dead loads, meets the following ASME Code, Section III criteria: • For Class MC free standing steel vessels and for the steel portions of Class CC reinforced concrete vessels which are not backed by concrete, the following requirements shall apply: – Paragraph NE-3221, Service Level C Limits on stress intensity values. – For regions of ellipsoidal or torispherical shell surfaces of containment, the allowable compressive stress due to internal pressure shall not exceed 60 percent of the value of critical buckling stress determined by one of the methods given in ASME Subparagraph 3222.1 (a).	The ASME Section III Code referenced structural integrity criteria satisfies the intended minimum requirements of 10CFR50.34(f)(3)(v). Also, any gross distortions and subsequent large strains in pressure boundary material due to potential shell buckling modes is precluded. The LDB requirements of Section 2.4.1 are expected to be limiting for inerted containments while the SMB requirements are expected to be limiting for containments which are not inerted.	5

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Paragraph No.	Requirement	Rationale	Rev.
	• Compressive stress in ellipsoidal or torispherical shell heads due to internal pressure loading is a localized stress field which does not represent a challenge to overall containment stability; thus a lower factor of safety against buckling than otherwise permitted by Subparagraph 3222.2 is appropriate in these regions. The value of 60 percent of the critical buckling stress results in a safety factor of 1.67, which is consistent with the requirements of Code Case N-284 for local buckling. • For the steel liner portions of Class CC vessels which are backed by concrete, the factored load limits on liner strains established in Subsubarticle CC-3720 shall apply. • For those portions of other ASME Code class components which also constitute a portion of the containment systems pressure boundary, the corresponding ASME Section III Service Level C Limits shall apply.		
6.6.2.2.1	**Inerted Pressure Suppression Containments**	**Inerted Pressure Suppression Containments**	5
	The analysis of LOCA plus hydrogen loads shall assume: • Pool temperature equal to the peak temperature associated with the DBA LOCA within 24 hours from the accident initiation. • All nitrogen in the drywell is located in the wetwell airspace.	The assumptions maximize the pressure and temperature loads in the containment in the performance of the 10CFR50.34(f)(3)(v) analysis.	5

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Paragraph No.	Requirement	Rationale	Rev.
	• The total hydrogen equivalent to 100% active fuel cladding metal water reaction is located in the wetwell airspace.		
6.6.2.2.2	**Non-inerted Containments**	**Non-inerted Containments**	5
	The analysis of LOCA plus hydrogen loads shall assume: • Peak pressure associated with the DBA LOCA; • Accumulation of hydrogen associated with 75% active fuel cladding metal water reaction; • Adiabatic isochoric complete combustion of this accumulated quantity of hydrogen. If containment is found to be steam inerted at the peak DBA pressure, then combustion shall be assumed to occur at the time steam condensation reduces the mole fraction of steam to combustible levels (~50% mole fraction steam).	The Licensing Design Basis analysis required by 10CFR50.34(f)(3)(v) would credit a hydrogen control system as hydrogen is generated. The Safety Margin Basis analysis requirement contained in this section postulates the peak DBA pressure and a realistic upper bound to total hydrogen concentration, i.e., that associated with 75% active clad oxidation, before crediting a hydrogen control system or ignition sources. This yields a higher peak pressure than that required by 10CFR50.34(f)(3)(v). Burning is assumed to occur at the highest potential containment pressure if inerting initially precludes combustion.	5
6.6.2.3	**Severe Accident Sequence Selection for Reporting Containment Response**	**Severe Accident Sequence Selection for Reporting Containment Response**	8
	The Plant Designer shall report containment performance during severe accidents. Analysis of severe accident sequences shall be performed to	The primary means of addressing severe accident containment challenges is the deterministic matrix of design characteristics and features of Section 6.6.2.1 and the deterministic analyses of	8

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	confirm that the containment provides substantial margin with respect to severe accident challenges. Accident sequences from the PRA shall be selected for analysis of containment performance. PRA sequences shall be grouped into functional sequence types for the purpose of determining the mean total frequency of all accident sequences with approximately the same type of challenge. The sequence types could be those resulting from the failure of any one of the following functions:	Section 6.6.2.2. This deterministic approach addresses an exhaustive list of containment challenges, regardless of probability. The probabilistic requirement of Section 6.6.2.3 complements the deterministic approach as required in the NRC Severe Accident Policy. The difficulty of assigning accurate numerical estimates notwithstanding, use of PRA in this manner provides valuable design insights and added confidence that containment margin exists for severe accidents and that important risk contributors have been addressed.	
	• Reactivity insertion; • RCS depressurization; • Core or core debris coolant inventory control; • Containment pressure/temperature control; • Combustible gas control; • Containment isolation and containment bypass control; • Other functions, the failure of which could lead to containment challenge.	This set of functions is considered necessary to assure containment integrity using a similar approach as described in the report, "Passive ALWR Severe Accident Containment Performance Requirements," January, 1992. This report concludes that the only potentially significant severe accident challenges to a standard ALWR plant design which implements the provisions in the Requirements Document are those associated with core damage events that occur into an intact containment with the RCS at low pressure with containment systems functioning as designed.	
	Functional sequence types with mean frequency greater than approximately 10-7/yr shall be analyzed for containment response. The determination of sequence types with mean frequency greater than 10-7/yr shall utilize	The approximately 10-7/yr threshold for functional sequence types to be analyzed for containment response is consistent with the NUREG-1420 10-7/yr limit for insignificant risks and is consistent with Standard Review Plan guidance to evaluate potential accidents from hazards in the plant vicinity which exceed approximately 10-7/yr. Also, NUREG-1150 uses a	

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	sequence information from the respective plant specific PRAs. The determination shall utilize propagation of uncertainty distributions to determine the mean frequency and shall utilize sensitivity studies as appropriate, shall include internal events, and shall either include, or provide a justification based on design, siting, or low frequency of occurrence, for exclusion of external events.	cutoff of 10-7/yr for accident progression analysis. NUREG-1338 stated that any sequence appearing to have a frequency down to about 10-7/yr will be examined from the standpoint of residual risk. Finally, consideration of functional sequence types greater than approximately 10-7/yr provides assurance that the cumulative effects of such sequence types will not exceed the 10-6/yr probability goal for off-site consequences.	
	Functional sequence types with frequency less than 10-7 per year shall be reported for discussion:	The plant specific PRA is the most complete and accurate source of accident sequence information. The specified treatment of uncertainties and external events is consistent with the KAG.	
	• Identifying the design features and operating characteristics credited to reach this low frequency; • Singling out the frequency of those sequence types which may result in early containment failure.	The purpose of this requirement is to assure that there is understanding of those features designed to preclude containment failure resulting from a severe accident. It is also expected to show that those phenomena which could lead to exceeding the capacity of containment early in a postulated severe accident event are a small fraction of the ALWR PRA goals for core damage frequency and consequences.	
	The loads resulting from any analyzed functional sequence types shall be no more limiting than the peak LOCA plus hydrogen loads of Section 6.6.2.2 for approximately 24 hours after the start of fission product release from the fuel.	If the loads resulting from the analyzed severe accident sequence types are enveloped by the conditions determined for LOCA plus hydrogen in accordance with Section 6.6.2.2, the comparison of these severe accident loads may be made directly with the LOCA plus hydrogen loads. In the event the loads exceed those determined in accordance with Section 6.2.2.2, it is expected the Plant Designer will be able to demonstrate that the containment still meets the functional criteria for Service Level C or Unity Factored Load as permitted by 10CFR50.34(f)(3)(v) and provide confidence that the structural integrity and leak tightness of the passive plant containment will be maintained following a severe accident.	

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Paragraph No.	Requirement	Rationale	Rev.
		Should any functional sequence type selected for analysis result in loads which exceed the functional criteria for Service Level C or Unity Load permitted by 10CFR50.34(f)(3)(v) or result in containment bypass, the Plant Designer should identify the reasons for the high loads or the bypass and explain why the accident sequence frequencies cannot be further reduced, and provide recommendations for an alternate basis on which confirmation of acceptable containment performance can be justified.	
6.6.2.4	**Containment Ultimate Capacity Analysis**	**Containment Ultimate Capacity Analysis**	0
	The Plant Designer shall perform an analysis to determine the ultimate structural capability of the containment. For steel containments, the ultimate capacity shall be defined as the pressure and temperature loadings which correspond to the collapse load defined by the method detailed in paragraph II-1430 of the ASME Code, Section III, Appendix II. For concrete containments, the ultimate structural capacity shall be defined as the pressure and temperature loading which produces liner plate strains equal to the liner strain limits of the ASME Code Section III, Subarticle CC-3720 for the Factored Load Category. The analysis shall consider the penetrations and their interaction with the containment, the shield building, and other structures internal or external to the containment, which might cause localized failure prior to the limit load for the overall pressure boundary. Results from testing of prototype details or models of prototype details may be used to augment such analyses. The failure mode associated with the ultimate structural capacity shall be identified.	An analysis of containment ultimate capacity is required by Standard Review Plans 3.8.1 and 3.8.2, including the determination of pressure retaining capacity of localized areas. The failure analysis criteria included here are identical to or more conservative than those developed during NRC/IDCOR issue resolution (see ARSAP Technical Task 2.3 report) or are more realistically based on recent experimental tests for concrete containments by Sandia National Laboratories. These tests have indicated that concrete containment capability may be limited by leakage resulting from liner plate tears. EPRI report NP-6261 describes computer modeling techniques used to predict the failure mode of the scale model concrete containment tested by Sandia. Interaction of the containment penetrations with the shield building or other structures may produce leakage paths.	0

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Paragraph No.	Requirement	Rationale	Rev.
6.6.2.5	**Cavity/Lower Drywell Structural Capability**	**Cavity/Lower Drywell Structural Capability**	7
	The Plant Designer shall evaluate the reactor cavity/lower drywell for loads rsulting from localized pressure due to a low pressure core melt ejection and effects from core debris material/water interaction to demonstrate that the reactor vessel will not lose overall structural support.	This requirement assures structural adequacy of the reactor cavity/lower drywell for local steam pressure loads resulting from low pressure core melt ejection and core debris water interaction.	7
6.6.2.6	**Long-term Containment Overpressure Protection**	**Long-term Containment Overpressure Protection**	5
	Protection of the containment for long-term overpressurization shall be provided. If significant time for recovery of core cooling and containment heat removal is available (e.g., 48 hours) prior to reaching Service Level C (steel containment structures) or Unity Factored Load (concrete containment structures) then overpressure protection may be provided simply by the size and strength of the containment. As an alternative, a system for pressure relief may be provided. In pressure suppression containments, pressure relief shall be provided preferentially from the suppression chamber air space to take advantage of scrubbing through the suppression pool. For other containment designs, pressure relief may be provided, utilizing available systems and equipment, provided that a significant decrease in the residual public risk results.	Containment overpressure protection provides additional defense-in-depth to protect the containment from catastrophic failure. The representative time for repair and recovery (48 hours) is drawn from the deterministic criteria of SECY-90-016 with respect to containment performance and takes advantage of the size of the containment in providing significant time for recovery of core cooling and decay heat removal systems and equipment. Compliance with this time criterion is demonstrated using "best estimate" analysis methodologies, including realistic assumptions. Controlled pressure relief is consistent with current BWR EPGs and provides a simple means of additional protection against low probability end-of-spectrum events such as loss of decay heat removal and non-condensible gas generation from core concrete interaction beyond that already provided in the requirements. For larger containments, the risk addressed by overpressure design features is sufficiently low that additional plant complexity is not warranted. Nevertheless, any simple overpressure protection provisions should be justified on a risk basis.	5

Section A.5
Summary of Evaluation of ALWR URD Requirements to Address Severe Accident Challenges

Table A.5-A-1
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE		KEY ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*		KEY ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
1	Containment Isolation	Isolation	PWR/BWR	P	Reduced fluid line penetration. Isolation provisions and leakage rate testing per standards. Valves capable of closure with possible flow and full containment pressure. Control room position indication for automatic and remote manual valves. Manual valve configuration permits locking only in closed position. Closed systems penetrating containment evaluated for ex-vessel severe accidents. Fail closed or DC powered isolation valves. Capability for periodic gross check of containment integrity.	P	Passive Residual Heat Removal minimizes core damage risk given isolation failure (with RHR on-line even without DC power).
2	Interfacing System LOCA	Bypass	PWR/BWR		Reduced interfaces between the Reactor Coolant System (RCS) and low pressure systems High to low pressure interfaces provided with isolation valve leak testing capability, isolation valve position indicator in control room, high pressure alarm. Interlocks prevent isolation valve opening when RCS pressure exceeds RSDC system design pressure (PWR). RDSC designed for full RCS pressure (BWR) Double valve isolation		Pressure Relief Design pressure such that full RCS pressure is below rupture pressure and no leaks will occur which exceed RCS makeup capacity.

* **Design features which are passive plant only are identified with P.**

Table A.5-1 (Continued)
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	KEY ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*		KEY ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
3	Blowdown Forces	Containment Pressure Control	PWR/BWR	Design and ISI in accordance with ASME BPV Code. Leak Before Break.		Design containment for double ended guillotine break of largest pipe.
4	Pipe Whip and Jet Impingement	Bypass	PWR/BWR	Design and ISI in accordance with ASME BPV Code. Leak Before Break. Use of only proven materials and fabrication processes. Use of EPRI water chemistry guidelines.		Protection from jet/pipe whip where leak before break is not demonstrated.
5	Steam Generator Tube Rupture	Bypass	PWR	Improved water chemistry. Proven materials. Mechanical design of tubes, tube supports, and tube sheets reduce likelihood of SGTR. Improved design features facilitate SG cleaning and replacement.	P P	Operator actions can terminate leakage prior to ADS actuation for design basis leak. Depressurization system operation terminates tube leakage automatically. Passive RHR plus additional features prevent secondary side relief following SGTR.
6	ATWS	Reactivity Control	BWR	Diverse Reactor Protection System (RPS). Diverse means of rod insertion.		Standby Liquid Control (SLC) Checkerboard pattern of scram group rods maximizes group worth.
			PWR	Diverse RPS (or capability to ride out ATWS).		Borated Safety Injection (SI) Negative moderator temperature coefficient over entire fuel cycle improves ATWS response.

* **Design features which are passive plant only are identified with P.**

Table A.5-1 (Continued)
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	KEY ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*		KEY ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
7	Suppression Pool Bypass	Containment Pressure Control	BWR	Vacuum Breakers: potential loads accounted for, position indication, minimal leakage.		ADS use of SRVs which discharge to suppression pool and thus ensure vapor suppression despite leakage.
8	Catastrophic RPV Failure	Internal Containment Loading	PWR/BWR	$RT_{NDT} < 10^{o}F$; Initial $RT_{NDT} < 20^{o}$ F for PWR core belt line; low fluence at vessel wall. No welds in belt line region. Relief valves prevent over pressure, backed up by depressurization system and low head injection. Design in accordance with ASME code. Design features to avoid relief valve opening for expected transients		
9	Internal Vacuum	Containment Pressure Control	PWR/BWR			Vacuum Breakers Design for external pressure loads.
10	Internal (Plant) Missiles	External Containment Loading	PWR/BWR	Turbine overspeed protection. Improved turbine integrity/one-piece rotors.		Turbine orientation avoids missile contact with containment. Missile protection for any safety related components in missile path (SRP 3.5.1.3)
11	Tornado and Tornado Missiles	External Containment Loading	PWR/BWR	Conformance with ANSI 2.12 and ANSI 51.5.	P	Passive core cooling systems located within containment.

* **Design features which are passive plant only are identified with P.**

Table A.5-1 (Continued)
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

				KEY ALWR REQUIREMENTS		
	CHALLENGE	**AFFECTED SAFETY FUNCTION**	**PLANT TYPE**	**LIMIT POTENTIAL FOR CHALLENGE***		**ACCOMMODATE CHALLENGES***
12	Man-Made Site Proximity Hazards	External Containment Loading	PWR/BWR	Conformance with ANSI 2.12.	P	Passive core cooling systems located within containment.
13	Seismic	External Containment Loading	PWR/BWR	Siting requirements exclude the most vulnerable sites.		SSE at 0.3g. Evaluation at > SSE with PRA or margins assessment as part of design process. Address vulnerabilities from past experience, e. g., provide common basemat.

* **Design features which are passive plant only are identified with P.**

Table A.5-2
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE		KEY ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
1	High Pressure Melt Ejection (HPME)	Reactor Pressure Control	BWR		Depressurization systems.	Suppression pool cools heated gases. Inerted containment (no combustion heat addition).
			PWR		Depressurization systems.	
				P	Passive RHR can aid depressurization.	Cavity configuration to limit transport of fragmented core debris.
2	Hydrogen Generation to Detonable Limits	Combustible Gas Control	BWR		Inerted.	Evaluation required if local detonation is possible.
			PWR		Limited H_2 generation with design features, such as ADS and cavity flooding. Hydrogen control system (e.g., non-safety related igniters) designed to keep hydrogen concentration below 10% for 100% active clad equivalent reaction. Containment size prevents global detonable H_2 concentration (< 13%) for generation up to 75% active clad equivalent reaction. Design provides convective mixing and minimizes DDT-prone geometry.	Evaluation required if local detonation is possible.

* **Design features which are passive plant only are identified with P.**

Table A.5-2
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

				KEY ALWR REQUIREMENTS	
	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	LIMIT POTENTIAL FOR CHALLENGE*	ACCOMMODATE CHALLENGES*
3	Hydrogen Deflagration	Combustible Gas Control	BWR	Inerted.	Demonstrated accommodation of generation equivalent to 100% active clad reaction. Structural evaluation for LOCA plus hydrogen loads (75% active clad reaction).
			PWR	Deflagration likely at low concentrations (< 10%) given hydrogen control system (IRWST and PCCS limit steam inerting potential).	Demonstrated accommodation of generation equivalent to 100% active clad reaction with multiple burns. Structural evaluation for LOCA plus hydrogen loads, including global burn of hydrogen equivalent to 75% active clad reaction.
4	In-Vessel Debris-Water Interaction	Internal Containment Loading	BWR/PWR	Large-scale phenomena limited in probability. In-vessel geometry limits interacting quantities and size of any interaction.	Rugged reactor vessel contains forces; as backup, rugged lower drywell/reactor cavity contains lower head failure.
5	Ex-Vessel Debris-Water Interaction	Internal Containment Loading	BWR/PWR	Large-scale phenomena limited in probability. Ex-vessel geometry limits interacting quantities and size of any interaction.	Rugged lower drywell/reactor cavity confirmed by evaluation. Containment design accommodates steam generation.

* **Design features which are passive plant only are identified with P.**

Table A.5-2
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	KEY ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
6	Noncondensible Gas Generation	Fuel/Debris Cooling	BWR/PWR	Features limiting concrete erosion (see item 7) limit noncondensible gas generation as well. Sacrificial concrete specified as low gas generation type. Overlying pool cools gases from core concrete interaction.	Containment size and pressure retention capability.
7	Basemat Erosion and Vessel Support Degradation	Fuel-Debris Cooling	BWR/PWR	Reactor cavity/lower drywell spreading area promotes core debris cooling. Lower drywell/cavity flooding. Lower drywell flooding thermally actuated directly from suppression pool. Overflow from containment reflux via PWR IRWST prefloods reactor cavity. Backup capability for water addition from sources external to containment.	Sacrificial concrete where debris on floor contacts boundary structures
8	Core Debris in Sump	Fuel-Debris Cooling	BWR/PWR	Special cavity sump design prevents localized, unterminated core-concrete interaction. Sump drain line configuration precludes gravity transport of debris ex-containment. Reactor cavity/lower drywell flooding.	

* **Design features which are passive plant only are identified with P.**

Table A.5-2
SUMMARY OF ALWR URD REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

	CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	KEY ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*		KEY ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
9	Core Debris Contact With Liner	Fuel/Debris Cooling	BWR/PWR	Liner protected by concrete. Lower drywell/cavity flooding. Design features to limit debris dispersal including ADS		
10	Decay Heat Generation	Containment Pressure Control	BWR	Main Condenser. Reactor Water Cleanup System.	P	Passive Containment Cooling.
			PWR	Steam Generator/Main Feedwater /Emergency or Backup Feedwater. Reactor Shutdown Cooling	P P	Passive Containment Cooling. Passive Heat Removal through containment shell without PCCS water limits containment pressure.
11	Tube Rupture from Hot Gases	Bypass	PWR	Steam Generator/Main Feedwater/Emergency or Backup Feedwater. Depressurization System.		

* **Design features which are passive plant only are identified with P.**

Section A.6
ALWR URD Requirements to Address Dominant WASH 1400 and Subsequent PRA Accident Sequences and Failure Modes

LOCA

- No recirculation piping in BWR; minimal number of welds in RCS piping in PWR
- RCS depressurization system allows low pressure systems to be effective regardless of the break size.
- It is unnecessary to switch to recirculation in PWRs since in passive plants, passive containment heat removal condenses steam released into containment and returns it to the vessel by gravity, and in evolutionary plants, the IRWST is located at a low elevation in containment so that water pumped from the tank returns via spillways.
- In passive plants, safety system dependencies essentially eliminated (include only dc power for the purpose of depressurization).
- In evolutionary plants, active safety and support systems have been designed to achieve high functional reliability based on worldwide nuclear experience.

Vessel Rupture

- Reduced RCS peak pressure for plant transients.
- Improved materials:
  - Less than .012% phosphorus, weld and base metal
  - Less than .03% copper, PWR base metal
  - Less than .05% copper, BWR base metal
  - Less than .08% copper, weld metal
  - Less than .05% vanadium, weld metal
- Initial ductility transition reference temperature less than 10°F (less than -10°F for PWR core belt region), reference temperature shift less than 30°F over plant life.
- Low fluence at vessel wall.
- No welds in beltline region.

Interfacing System LOCA

- Low pressure systems normally isolated from the RCS are provided with interlocks to prevent their exposure to RCS pressure and are enunciated should high pressure conditions occur.
- The ultimate rupture strength of potential interfacing systems is capable of withstanding full RCS pressure.

Transient (loss of injection)

- In passive plants, core passive residual heat removal system automatically actuates on loss of ac power. Passive system is fail safe and can operate independent of any support system.

- In passive plants, automatic depressurization and gravity injection are capable of providing adequate core cooling independent of normal makeup systems and passive residual heat removal system.
- In evolutionary plants, depressurization systems provide the capability to reduce primary system pressure to initiate feed and bleed and RHR cooling in PWRs and inject water from alternate sources in BWRs.

Transient (station blackout)

- In passive plants, core passive residual heat removal system automatically actuates on loss of ac power. Passive system is fail safe and can operate independently from any support system.
- In evolutionary plants, a non-safety related, alternate ac on-site power source is provided in addition to safety related ac on-site power sources for each division.
- In evolutionary plants, a four train (2 electric and 2 steam turbine driven pumps) and automatic dump valves permit operation at hot standby for a blackout period of up to 8 hours.
- In passive plants, battery capacity in excess of 72 hours.
- In passive plants, canned rotor reactor coolant pumps are provided in the PWR, eliminating the potential for seal LOCA (the BWR is natural circulation and has no recirculation pumps).
- In evolutionary plants, the core shall be capable of withstanding a station blackout for at least 8 hours without fuel damage.

ATWS

- PWR capability to ride out an ATWS.
- PWR negative moderator temperature coefficient over entire operating cycle.
- PWR borated safety injection.
- BWR capability to mitigate short term ATWS effects and shutdown automatically by diverse means:
  - Safety relief valve capacity > 100% power
  - Motor drives diverse from hydraulic drive mechanisms
  - Auxiliary Rod Insertion system diverse from reactor protective system
  - Automatic Standby Liquid Control independent of all support systems except dc power

Shutdown Risk

- Permanent, operable, redundant water level instrumentation designed for use during shutdown conditions.
- Antisiphon provisions in refueling pool cooling and cleanup system piping to prevent pool drain down.
- Features to prevent or mitigate the effects of losing suction to decay heat removal pumps during shutdown condition (e.g., piping design to minimize vortexing and air entrainment).
- Features to assure required net positive suction head is always available to decay heat removal pumps.
- In passive plants, passive decay heat removal systems are capable of removing decay heat and preventing RCS overpressure.
- Detailed requirements for analyses of mid-loop operation (PWRs) and low-level operation (BWRs) to provide assurance that known loss of shutdown cooling problems have been addressed and that information to operate the plant safely during shutdown has been developed.
- Provision of a separate power supply circuit to the plant permanent non-safety leads for use in the event of extended unavailability of the normal power supply such as may occur during shutdown.
- Capability of closing valves for draining the reactor vessel or RCS without reliance on ac power.
- Limitations on boron dilution flow in PWRs such that the operator has at least 30 minutes after indication of dilution to terminate the incident prior to any recriticality.

Overpressure (steam)

- In passive PWR, passive containment cooling systems transfer heat directly through the containment steel shell without dependence on support systems.
- In evolutionary plants, dedicated safety related systems are provided to remove heat from the containment and BWR suppression pool.

Overpressure (noncondensables) and Basemat Penetration

- Reactor cavity/lower drywell configured to promote spreading of core debris to increase coolability.
- Ample water is available to cool debris in the reactor cavity/lower drywell passively, by means independent of potential causes for core damage.

In-Vessel Steam Explosion

- Containment failure due to in-vessel steam explosion was unlikely in WASH 1400, and has been reexamined several times since and is now considered to be extremely unlikely. This is due to improved understanding of steam explosion phenomena, particularly the extent to which water depletion in the debris-water interaction zone (due to high heat transfer rates from debris fragments to the water and to the dispersive effect of the subsequent high steaming rates on the surrounding water pool) limits molten debris premixing and mechanical energy yield.

Hydrogen Combustion

- The BWR containment is inerted.
- The PWR containment is required to have a hydrogen control system. Even without crediting this system, the PWR containment is capable of withstanding a burn associated with hydrogen generated from oxidation of as much as 75% of the active fuel cladding without exceeding ASME Service Level C limits.

Containment Isolation

- Passive plants have fewer penetrations as a result of safety systems being located inside containment and other changes to reduce the number of penetrations.
- Most penetrations are isolated during power operation.
- Penetrations which may be open during power operation are fail safe or dc powered making them effectively independent of support systems.
- A periodic, on line leakage monitor is specified to avoid large openings.

Liner Melt-through

- Reactor cavity and lower drywell are configured to protect the containment boundary from direct contact by core debris.

Ex-Vessel Steam Explosion

- Similar to in-vessel steam explosions, water depletion in the debris-water interaction zone limits ex-vessel molten debris premixing and mechanical energy yield; also, explosion venting due to open geometry and shallow pool limits pressure pulse propagation to structures.
- A rugged BWR reactor vessel foundation design is provided together with a URD requirement to demonstrate that ex-vessel debris water interactions will not cause loss of reactor vessel structural support.
- A shield is provided in the BWR lower drywell to protect the containment boundary from the effects of debris-water interactions.
- Ex-vessel steam explosion loads evaluated for PWRs and predicted not to cause containment failure

<u>Direct Containment Heating</u>

- A safety related RCS depressurization system. The depressurization systems require only dc power for operation. Evolutionary PWR system is manually actuated while other ALWR plants are automatically actuated.
- Decay heat removal systems are capable of reducing and maintaining the RCS at low pressures. In passive plants, these systems are passive.
- Cavity/lower drywell configuration is such that much of the debris will be trapped as opposed to being entrained in the steam flow. Also, recent work suggests that any debris which is entrained is exposed to only a small fraction of the steam flow from the RCS, thus greatly limiting the potential for thermal/chemical interactions.

<u>Overtemperature</u>

- RCS depressurization system and RCS passive decay heat removal system minimize high pressure melt ejection and resulting core debris transport into upper drywell.
- Ample water available in lower drywell to cool debris and avoid high temperatures.
- BWR drywell spray to reduce temperatures.

<u>Steam Generator Tube Rupture (SGTR)</u>

- Reduced primary coolant temperatures to reduce corrosion.
- Improved water chemistry and tube materials (i.e., NiCrFe alloy 690 TT).
- Improved mechanical design of tubes and tube bundles.
- In passive plants, passive RHR prevents need for secondary side relief and steam generator overfill.
- RCS depressurization system terminates tube leakage.
- RCS depressurization minimizes convection of hot gases and limits the differential pressure which could cause tube rupture.
- Accident management equipment and procedures designed to provide injection of water into steam generator secondary side in the event of a high pressure, unterminated core damage sequence

B

ASSESSMENT OF AP600 DESIGN CONFORMANCE WITH ALWR EMERGENCY PLANNING DESIGN CRITERIA

The Westinghouse AP600 design has been submitted to NRC for design certification under 10 CFR 52. The following assessment of this standard passive advanced light water reactor (ALWR) plant design is provided to describe how the design meets the ALWR emergency planning design criteria. The assessment is based on information contained in the AP600 Standard Safety Analysis Report (SSAR) [B-1], the AP600 Probabilistic Risk Assessment [B-2], and other documentation referenced by the SSAR. As described in the ALWR Utility Requirements Document (URD) [B-3], Volume III, Chapter 1, Section 2.6, the ALWR emergency planning design criteria include a containment performance criterion and site boundary dose criterion. It is necessary to assess conformance of the design to these criteria. In addition, the URD specifies that a supplementary PRA evaluation be performed in support of the assessment.

B.1 CONTAINMENT PERFORMANCE CRITERION

The containment performance criterion for emergency planning appears in URD Chapter 1, Section 2.6.1 and is repeated in Appendix A. A summary of the containment performance criterion is as follows:

> Plant design characteristics and features shall be provided to preclude core damage sequences which could bypass containment and to withstand containment loads from core damage sequences. Containment loads from core damage sequences should be evaluated and should not exceed the combination of peak loads from a DBA LOCA and a hydrogen burn from oxidation of 75% of the active fuel cladding, or not exceed ASME Service Level C limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours, there shall be no uncontrolled release.

The methodology which is specified in the URD for demonstrating the containment performance criterion includes the following:

- Incorporate the design characteristics and features specified in the URD to address severe accident challenges.
- Evaluate containment response for a reference severe accident sequence. This sequence shall be a low pressure core melt into an intact containment with containment systems functioning

as designed to confirm that containment loads do not exceed peak LOCA plus hydrogen loads or ASME Service Level C limits.

- Evaluate containment response for functional sequence types with mean frequency $>10^{-7}$ per year to confirm that containment leaktightness and structural integrity are maintained, and that containment loads do not exceed peak LOCA plus hydrogen loads or Service Level C limits. Functional sequence types with mean frequency $<10^{-7}$ per year should be reported for discussion including a description of the plant features credited to reach this low frequency.
- Provide protection of the containment for overpressurization beyond 24 hours. Overpressure protection may be provided by the size and strength of the containment. On the order of two to three days is judged to be adequate time for actions by the plant staff to bring the accident under control.

The steps used for the assessment of AP600 compliance with the containment performance criterion were as follows:

1. Confirm that the design meets the requirements of the URD, Volume III, Chapter 5, Section 6.6.2.1 by performing a comparison between the AP600 design characteristics and features and the requirements.

2. Confirm that containment loads from an AP600 reference core damage sequence do not exceed limits as specified in the URD, Volume III, Chapter 5, Section 6.6.2.2 for approximately 24 hours under realistic severe accident assumptions.

3. Confirm as specified in the URD, Volume III, Chapter 5, Section 6.6.2.3 that containment leaktightness and structural integrity are maintained in response to AP600 functional severe accident sequences with frequency $> 10^{-7}$ per year and that containment loads do not exceed limits specified in the URD, Volume III, Chapter 5, Section 6.6.2.2. In addition, functional severe accident sequences with frequency $< 10^{-7}$ per year are reported with a description of plant features credited to reach this low frequency.

4. Confirm that the AP600 design meets the URD, Volume III, Chapter 5, Section 6.6.2.5 and that no uncontrolled release will occur beyond approximately 24 hours.

B.1.1 Plant Design Characteristics and Features to Address Containment Challenges

The first step in the assessment was performed by reviewing the AP600 SSAR to confirm, for each containment challenge, the existence of specific design features or characteristics to fulfill the key URD requirements associated with the challenge. The list of challenges and associated URD requirements summarized in Appendix A, Tables A.5-1 and A.5-2 was used for this review. A requirement was considered met when an explicit reference to the system, feature, or characteristic was made in the AP600 SSAR.

Table B-1 summarizes the results of the assessment for AP600. This table lists the challenges and associated requirements, and identifies in brackets the sections of the AP600 SARwhich address each requirement. Specific AP600 design features or capabilities were identified in response to all of the requirements. Additional information was requested from Westinghouse for the items listed in Table B-2. The reference [B-4] Westinghouse letter provides this additional information.

On the basis of the assessment, the AP600 design meets the requirements of Volume III, Chapter 5, Section 6.6.2.1 of the URD.

B.1.2 Comparison of Loads From A Reference Severe Accident With URD Limits

Since the AP600 design meets the URD provisions related to containment challenges, the reference severe accident sequence for which containment loads should be compared with URD limits is a low pressure core melt accident into an intact containment with the reactor coolant system (RCS) at low pressure and containment systems functioning as designed.

A comparison of AP600 containment loads from a reference severe accident sequence with URD limits has been performed by evaluating a low pressure core melt accident sequence presented in Chapter 34 of the AP600 PRA. This base case accident sequence is initiated by a break which occurs in one of the two direct vessel injection (DVI) lines. Operation of the automatic depressurization system (ADS) is successful, but it is assumed that there is failure of gravity injection from the internal refueling water storage tank (IRWST) to the reactor coolant system. The base case sequence is representative of accident class 3BE which is the largest contributor to the AP600 core damage frequency, comprising about 46% of the total core damage frequency. Table B-3 shows the accident classes defined for AP600.

Table B-1
Assessment of AP600 Design Conformance with ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS: LIMIT POTENTIAL FOR CHALLENGE	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS: ACCOMMODATE CHALLENGE
1. Containment Isolation	Isolation	• Reduced fluid line penetrations [6.2.3.2.1 & 6.2.3.1.3-A]. • Isolation provisions and leakage rate testing per standards [6.2.5.2.2]. • Valves capable of closure with maximum flow and full containment pressure [6.2.3.1.3-F]. • Control room position indication for automatic and remote manual valves [6.2.3.1.3-H,I]. • Manual valve configuration permits locking only in fully closed position [6.2.3.1.3-J]. • Closed systems penetrating containment evaluated for ex-vessel severe accidents [6.2.3.1.1-H]. • Fail closed or DC powered isolation valves [6.2.3.1.3-J]. • Capability for periodic gross check of containment integrity [see Ref. [B-4]].	• Passive Residual Heat Removal (RHR) minimizes core damage risk given isolation failure (with RHR on-line even without DC power) [6.3].
2. Interfacing System LOCA	Bypass	• Reduced interfaces between the Reactor Coolant System (RCS) and low pressure systems [PRA App. A.3.2]. • High to low pressure interfaces provided with isolation valve leak testing capability [6.2.5.2.2 & for RHR, Fig. 5.4-7], isolation valve position indicator in control room [6.2.3.1.3-H, I & for RHR see Ref. [B-4], and high pressure alarm [RHR, 5.4.7.2.2 & 7.6.1.1.1, see Ref. [B-4]]. • Interlocks prevent isolation valve opening when RCS pressure exceeds RSDC system design pressure [5.4.7.2.2 & 7.6.1.1]. • Double isolation [5.4.7.2.2].	• Pressure Relief [5.2.2 & 5.4.7.2.2]. • Design pressure such that full RCS pressure is below rupture pressure and no leaks will occur which exceed RCS makeup capacity [5.4.7.2.2 & PRA App. A.3.2].
3. Blowdown Forces	Containment Pressure Control	• Design and ISI in accordance with ASME BPV Code [5.2.1.1]. • Leak Before Break [5.1.3.4 & 3.6.1.1-P].	• Design containment for double-ended guillotine break of largest pipe [6.2.1.1.1].
4. Pipe Whip and Jet Impingement	Bypass	• Design and ISI in accordance with ASME BPV Code [5.2.1.1]. • Leak Before Break [5.1.3.4 & 3.6.1.1-P]. • Use of only proven materials and fabrication processes [5.2.3.1]. • Use of EPRI water chemistry guidelines [5.4.2.4.3 & 10.3.5].	• Protection from jet/pipe whip where leak before break is not demonstrated [3.6.1.1-C; 3.6.2.3.4.2 & 3.6.2.4.1].

Table B-1 (Continued)
Assessment of AP600 Design Conformance with ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
5. Steam Generator Tube Rupture	Bypass	• Improved water chemistry [5.4.2.4.3 & 10.3.5]. • Proven materials [5.4.2.4.1]. • Mechanical design of tubes, tube supports, and tube sheets reduce likelihood of SGTR [5.4.2.3.3, 5.4.2.3.4 & 5.4.2.4.2]. • Improved design features facilitate SG cleaning and replacement [5.4.2.2 & 5.4.2.5].	• Operator actions can terminate leakage prior to ADS actuation for design basis leak [15.6.3]. • Automatic Depressurization System (ADS) operation terminates tube leakage [15.6.3]. • Passive RHR prevent secondary side relief following SGTR [15.6.3].
6. ATWS	Reactivity Control	• Diverse RPS (or capability to ride out ATWS [4.3.1.7]) [PRA App. C12].	• Borated Safety Injection (SI) [5.4.13]. • Negative moderator temperature coefficient over entire fuel cycle improves ATWS response [4.2.2.3].
7. Suppression Pool Bypass	Containment Pressure Control	NOT APPLICABLE	
8. Catastrophic RPV Failure	Internal Containment Loading	• $RT_{NDT} \leq 10°F$; initial $RT_{NDT} \leq -10°F$ for PWR core beltline; low fluence at vessel wall [5.3.3.1]. • Phosphorous, copper, vanadium, and sulfur material limits in high fluency regions [5.3.2.1] • No welds in beltline region [5.3.1.2 & 5.3.4.1]. • Relief valves prevent overpressure, backed up by depressurization system, low-head injection [5.4.9] • Design in accordance with ASME code [5.3.1.1]. • Design features to avoid relief valve opening for expected plant transients [6.3.1.1.1 & 15.2.8.3].	
9. Internal Vacuum	Containment Pressure Control		• Design for external pressure loads [3.8].
10. Internal (Plant) Missiles	External Containment Loading	• Turbine overspeed protection [10.2.2.3.6]. • Improved turbine integrity/one-piece rotors [10.2.3].	• Turbine orientation avoids missile contact with containment [3.5.1.3]. • Missile protection for any safety related components in missile path (SRP 3.5.1.3) [3.5].
11. Tornado and Tornado Missiles	External Containment Loading	• Conformance with ANSI 2.12 and ANSI 51.5 (in accordance with ASCE 7-88, "Minimum Design Loads for Buildings and other Structures," formerly ANSI A58.1-82) [3.3.1; 3.5.2 & 3.5.3].	• Passive core cooling systems located within containment [Fig. 6.3-5, 6.3-6, 6.3-7].
12. Man-Made Site Proximity Hazards	External Containment Loading	• Conformance with ANSI 2.12 [see Ref. [B-4]].	• Passive Core cooling systems located within containment [Fig. 6.3-5, 6.3-6, 6.3-7].

Table B-1 (Continued)
Assessment of AP600 Design Conformance with ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
13. Seismic	External Containment Loading	• Siting requirements exclude the most vulnerable sites [no effect on design].	• SSE at 0.3g [3.7.1]. • Evaluation at > SSE with margins assessment as part of design process [PRA App. H]. • Address vulnerabilities from past experience, e.g., provide common basemat [3.8.5.1].
14. High Pressure Melt Ejection (HPME)	Reactor Pressure Control	• Diverse depressurization systems [5.1.2, 5.4.6 & 6.3.2.2.7.5]. • Passive RHR can aid depressurization [6.3]. [See also PRA 10.2.2 & App. L.2.5].	• Cavity configuration to limit transport of fragmented core debris [PRA 10.2.3].
15a. Hydrogen Generation to Detonable Limits	Combustible Gas Control	• Limit H_2 generation with design features, such as ADS and cavity flooding [5.4.6 & 3.8.3.1.5]. • Hydrogen control system designed to keep hydrogen concentration below 10% for 100% active clad equivalent reaction [6.2.4]. • Containment size prevent global detonable H_2 concentration (<13%) for generation up to 75% active clad equivalent reaction [see Ref. [B-4]]. • Design ensures convective mixing and minimizes DDT-prone geometry [6.2.4.1.1; PRA 10.2.5 & App. O].	• Evaluation required if local detonation is possible [PRA 14.0; 15.0; App. N & App. O].
15b. Hydrogen Deflagration	Combustible Gas Control	• Recombination or deflagration likely at low concentrations (<10%) given hydrogen control system (IRWST and PCCS limit steam inerting potential) [PRA App. N & App. O].	• Demonstrated accommodation of generation equivalent to 100% active clad reaction with multiple burns [PRA App. N]. • Structural evaluation for LOCA plus hydrogen loads, including global burn of hydrogen equivalent to 75% active clad reaction [PRA App. N.4.8].
16. In-Vessel Debris-Water Interaction	Internal Containment Loading	• Large-scale phenomena limited in probability [PRA 10.2.1]. • In-vessel geometry limits interacting quantities and size of any interaction [PRA 10.2.1].	• Rugged reactor vessel contains forces [PRA 10.2.1]; as backup, rugged reactor cavity contains lower head failure [PRA 10.2.1].
17. Ex-Vessel Debris-Water Interaction	External Containment Loading	• Large-scale phenomena limited in probability [PRA 10.2.1]. • Ex-vessel geometry limits interacting quantities and size of any interaction [PRA 10.2.1].	• Rugged reactor cavity confirmed by evaluation [PRA 10.2.1]. • Containment design accommodates steam generation [PRA 10.2.1].

Table B-1 (Continued)
Assessment of AP600 Design Conformance with ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
18. Noncondensible Gas Generation	Fuel/Debris Cooling	• Features limiting concrete erosion (see item 19) limit noncondensible gas generation as well. • Overlying pool cools gases from core-concrete interaction [PRA 10.2.4].	• Containment size and pressure retention capability [PRA 10.2.4 & App. I].
19. Basemat Erosion and Vessel Support Degradation	Fuel/Debris Cooling	• Reactor cavity/lower drywell spreading area of $0.02m^2$/MWt promotes core debris cooling [PRA 10.2.4]. • Reactor cavity flooding [PRA 10.2.2]. • Overflow from containment reflux via PWR IRWST prefloods reactor cavity [PRA 10.2.2]. • Backup capability for water addition from sources external to containment [PRA App. C.4.4.1].	• Sacrificial concrete where debris on floor contacts boundary structures [3.8.2.1.2].
20. Core Debris in Sump	Fuel/Debris Cooling	• Special cavity sump design prevents localized unterminated core-concrete interaction [10.2.4]. • Sump drainline configuration precludes gravity transport of debris ex-containment [PRA 10.2.4]. • Reactor cavity flooding [PRA 10.2.2].	
21. Core Debris Contact with Liner	Fuel/Debris Cooling	• Liner protected by concrete [3.8.2.12]. • Reactor cavity flooding [PRA 10.2.2]. • Design features to limit debris dispersal including ADS [5.4.6 & 3.8.3.1.5].	
22. Decay Heat Generation		• Steam Generators/Main Feedwater (MFW)/Startup Feedwater [10.4.9]. • Normal Residual Heat Removal System [5.4.7].	• Passive Containment Cooling [6.2.2]. • Passive Heat Removal through containment shell without PCCS water limits containment pressure [PRA App. L.3.1 & L.3.2].
23. Tube Rupture from Hot Gases		• Steam Generators/MFW/Startup Feedwater [10.4.9]. • Depressurization System [5.1.2]	

Table B-2
AP600 SSAR Items for Which Additional Information was Requested

1. No SSAR provision exists for periodic gross check of containment integrity. Westinghouse has committed to the ALWR Program to provide this capability in AP600.

2. No SSAR requirement exists for a high pressure alarm on the high-to-low pressure interface for the Primary Sampling System and the Chemical Volume and Control System (CVCS). Westinghouse has committed to the ALWR Program to provide this capability in AP600.

3. Isolation valve position indication for the RHR System is not mentioned in the SSAR. Westinghouse confirmed that this capability is provided in the design.

4. No SSAR commitment to ANSI 2.12 [B-13] exists for man-made site proximity hazards. Westinghouse has committed that the AP600 will conform to ANSI 2.12.

5. No explicit statement is made in the SSAR regarding containment size being large enough to limit dry hydrogen concentration to less than 13% given 75% active clad oxidation. However, the AP600 containment design has been evaluated based on the AP600 zircaloy mass and containment volume, and the 13% requirement is met.

In the base case sequence for accident class 3BE, the DVI line break causes the reactor coolant system pressure to drop. Low pressure causes the reactor to scram, and leads to core makeup tank actuation. In addition, the accumulators inject into the reactor coolant system, and after about 14 minutes, the automatic depressurization system is actuated on low core makeup tank level. The failure of gravity injection into the reactor coolant system causes the reactor coolant system water level to drop, and the top of the core is uncovered. After 1.2 hours, the operator is assumed to open the valve to drain the IRWST into the reactor cavity. The onset of core melting occurs at about 1.4 hours. In about 3.5 hours the water level in the reactor cavity increases to the point that it begins entering the reactor coolant system through the break. The core is reflooded both from the inside and outside, and the reactor vessel is predicted not to fail. Hydrogen produced from metal oxidation is controlled by igniters.

The containment peak pressure and temperature are 36 psia (2.5 bar) and 370 °F (463 °K) respectively. The peak LOCA plus hydrogen burn pressure has been determined in Chapter 41, Section 41.11 of the AP600 SSAR to be 89 psia (6.14 bar). This section also indicates that this gives a 15 psi margin to the Service Level C limit which is 104.7 psia at 400° F based on the capacity of the 16-foot diameter equipment hatch cover which is the limiting part of the containment.. Thus the containment design provides substantial margin for severe accident loads compared to the URD limits.

B.1.3 Evaluation of Containment Response for Functional Sequence Types

The URD requires in Volume III, Chapter 5, Section 6.6.2.3 that containment leaktightness and structural integrity be analyzed in response to functional severe accident sequences with frequency > 10^{-7} per year, and that containment loads not exceed the URD limits Volume III, Chapter 5, Section 6.6.2.2. In addition, functional accident sequences with frequency <10^{-7} per year shall be reported with a description of plant features credited to reach this low frequency. Finally, the frequency of those sequences types which may result in early containment failure are to be identified.

Table B-3
Summary of AP600 Severe Accident Sequences for Level 2 PRA Analysis Base Case Sequences

ACCIDENT CLASS	LEVEL 1 DEFINITION	LEVEL 2 BASE CASE DEFINITION	CORE DAMAGE FREQUENCY per year	CONTAINMENT MAXIMUM PRESSURE AND COMPARTMENT TEMPERATURE	CONTAINMENT FAILURE	FEATURES AND CHARACTERISTICS LIMITING FREQUENCY AND CONTAINMENT LOADS
3BE	RCS fully depressurized and IRWST gravity injection failure	PCCS, wet cavity, H_2 igniters success	7.8×10^{-8}	2.5 bar (36 psia) 463°K (370°F)	no	Leak before break; only proven RCS materials; reliable, redundant passive injection; PCCS; H_2 igniters
3BL	RCS fully depressurized, IRWST recirculation failure	PCCS, wet cavity, H_2 igniters success	4.4×10^{-8}	2.8 bar (41 psia) 740°K (820°F)	no	Leak before break; only proven RCS materials; reliable, redundant injection and recirculation; PCCS; H_2 igniters
3BR	RCS fully depressurized and accumulators fail	PCCS, wet cavity, H_2 igniters success	7.7×10^{-9}	3.1 bar (44 psia) 660°K (710°F)	no	Leak before break; only proven RCS materials; reliable, redundant accumulators; PCCS; H_2 igniters
3C	Vessel rupture	PCCS, wet cavity, H_2 igniters success	1.0×10^{-8}	3.3 bar (48 psia) 780 °K (950 °F)	no	Vessel material and fabrication requirements Table B-1, # 8
3D & 1D	LOCA (except large) or transient and partial depressurization	PCCS, wet cavity, H_2 igniters success	6.2×10^{-9}	2.4 bar (34 psia) 390°K (240°F)	no	Leak before break; only proven RCS materials; reliable, redundant, diverse ADS; reliable, redundant accumulators; PCCS; H_2 igniters
6E & 6L	SG tube rupture, IRWST injection or recirculation failure	PCCS, wet cavity, H_2 ig-niter success; MSSV fails open	8.7×10^{-9}	1.6 bar (23 psia) 370 °K (210°F)	bypass	Proven tube materials and water chemistry, hot leg temperature less than 400°F, reliable, redundant, diverse ADS, reliable recirculation
1AP	LOCA (except large) and ADS failure	PCCS, wet cavity, H_2 ig-niter success; induced SGTR	3.2×10^{-9}	2.1 bar (30 psia) 500 °K (440°F)	bypass	Leak before break; only proven RCS and tube materials; reliable , redundant, diverse ADS: reliable, redundant CMTs
1A	Transients with ADS failure	PCCS, wet cavity, H_2 ig-niter success; induced SGTR	1.8×10^{-9}	2.4 bar (35 psia) 590 °K (600°F)	bypass	Low transient frequency; reliable, redundant, diverse ADS; reliable PRHR
3A	ATWS	PCCS, wet cavity, H_2 igniters success	1.0×10^{-8}	N/A	N/A	Reliable scram system; RCS coping capability, minimal adverse MTC

ADS - automatic depressurization system
ATWS - anticipated transient w/o scram
CMT - core makeup tank
DVI- direct vessel injection
LOCA - loss of coolant accident
MSSV - main steam safety valve
MTC - moderator temperature coefficient
PCCS - passive containment cooling
PRHR - passive residual heat removal system
PORV - power operated relief valve
RCS - reactor coolant system

There are 11 functional sequences types or accident classes described in Chapter 34 of the SSAR. These accident classes are shown in Table B-3, and encompass the numerous accident sequences defined in the Level 1 PRA. The specific scenario chosen for Level 2 analysis for an accident class (Level 2 base case) is based on the most probable sequence in the accident class. In addition, sensitivity cases were analyzed in the PRA to ensure that all major aspects of the AP600 design's response to core damage scenarios are addressed. The base case Level 2 analysis and the sensitivity studies are used to determine representative sequences for each release category to estimate fission product releases that are input for the offsite dose analysis. AP600 sensitivity studies are summarized in Tables B-4 and B-5.

As shown in Table B-3, there are no accident classes with core damage frequency estimated to be greater than 10^{-7} per year. Two of the accident classes (3BE and 3BL) have frequencies somewhat above 10^{-8} per year. For these accident classes, containment leaktightness and structural integrity are maintained and containment loads are well below URD limits. For all accident classes, the plant features which limit frequency are noted in the right-hand column of Table B-3.

Three accident classes in Table B-3 are assumed to lead to containment failure. The 6E & 6L class consists of unisolated, spontaneous SGTR sequences. The 1A and 1AP classes involve induced SGTR sequences. In addition, sensitivity studies were performed which in some cases were assumed to lead to containment failures as shown in Tables B-4 and B-5. Sequences in these tables assume failure of most features that mitigate severe accidents in the AP600 design, and combine these failure assumptions with pessimistic phenomenological assumptions. Containment failure is predicted to occur in these sensitivity cases in which hydrogen detonation, high pressure core melt and ejection, or creep rupture of the steam generator tubes is assumed. The frequencies associated with these sequences is far less than the frequencies reported for the base case studies since they include the unavailability of mitigating features and the probabilities of incurring the pessimistic phenomenological assumptions incorporated in these studies. As indicated in Tables B-4 and B-5, the frequencies of these sensitivity study sequences are estimated to be below 10^{-9} per year based on the large combinations of independent events that are necessary in order for these sequences to occur.

Table B-4
Summary AP600 Severe Accident Sequences for Level 2 PRA Analysis Accident Class 3BE Sensitivity Cases

LEVEL 2 SENSITIVITY CASE	SENSITIVITY CASE DEFINITION	FREQUENCY per year	CONTAINMENT MAXIMUM PRESSURE AND COMPARTMENT TEMPERATURE	CONTAINMENT FAILURE	FEATURES AND CHARACTERISTICS LIMITING FREQUENCY AND CONTAINMENT LOADS
3BE-2	partial ADS; 1/2 CMTs and 0/2 accumulators; 0/2 IRWST gravity injection lines	$< 10^{-8}$	3 bar (43 psia) 530°K (490°F)	no	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection; PCCS; H_2 igniters
3BE-3	9-inch hot leg break; partial ADS; 1/2 CMTs and 1/2 accumulators; 0/2 IRWST gravity injection lines	$< 10^{-8}$	2.9 bar (42 psia) 500°K (420°F)	no	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection; PCCS; H_2 igniters
3BE-4	6-inch hot leg break; partial ADS, 1/2 CMTs and 0/2 accumulators; 0/2 IRWST injection & recirculation lines; 1/2 cavity flooding lines	$< 10^{-8}$	2.9 bar (42 psia) 500°K (234 °F)	no	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection and recirculation; PCCS; H_2 igniters
3BE-5	6-inch hot leg break; partial ADS, 1/2 CMTs and 0/2 accumulators;0/2 IRWST injection & recirculation lines; 1/2 cavity flooding lines;CI failure in valve vault	$< 10^{-9}$	2.3 bar (33 psia) 490°K (420°F)	CI failure assumed in valve vault	Leak before break; only proven RCS materials; reliable, redundant ADS, passive ADS, injection and recirculation; PCCS; H_2 igniters, reliable, redundant CI
3BE-7	6-inch hot leg break; partial ADS, 1/2 CMTs and 0/2 accumulators; 0/2 IRWST injection & recirculation lines; 0/2 cavity flooding lines	$< 10^{-9}$	2.9 bar (42 psia) 570°K (570°F)	Containment failure assumed in valve vault at time of vessel failure	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection and recirculation; PCCS; H_2 igniters
3BE-8	4-inch DVI line break; partial ADS, 1/2 CMTs and 1/2 accumulators;0/2 IRWST injection & recirculation lines; 1/2 cavity flooding lines, H_2 igniters failed; containment failure assumed in valve vault	$< 10^{-9}$	2.5 bar (36 psia) 475°K (390°F)	Early H_2 detonation assumed in valve vault	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection and recirculation; PCCS; H_2 igniters
3BE-9	4-inch DVI line break; partial ADS, 1/2 CMTs and 1/2 accumulators; 0/2 IRWST injection & recirculation lines; 1/2 cavity flooding lines, H_2 igniters failed; containment failure assumed in valve vault	$< 10^{-9}$	2.7 bar (39 psia) 650°K (710°F)	H_2 detonation assumed in valve vault after core relocation	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection and recirculation; PCCS; H_2 igniters
3BE-10	6-inch hot leg break; partial ADS, 1/2 CMTs and 0/2 accumulators; 0/2 IRWST injection & recirculation lines; 0/2 cavity flooding lines, PCCS fails	$< 10^{-9}$	7.1 bar (105 psia) 540°K (510°F)	Overpressure failure assumed at 90 psig (~30 hrs after accident initiation)	Leak before break; only proven RCS materials; reliable, redundant ADS, passive injection and recirculation; PCCS; H_2 igniters

Table B-5
SUMMARY OF AP600 SEVERE ACCIDENT SEQUENCES FOR LEVEL 2 PRA ANALYSIS ACCIDENT CLASS 3BL, 3D, AND 6E SENSITIVITY CASES

LEVEL 2 SENSITIVITY CASE	SENSITIVITY CASE DEFINITION	FREQUENCY per year	CONTAINMENT MAXIMUM PRESSURE AND COMPARTMENT TEMPERATURE	CONTAINMENT FAILURE	FEATURES AND CHARACTERISTICS LIMITING FREQUENCY AND CONTAINMENT LOADS
3BL-2	partial ADS 1/2 CMTs and 0/2 accumulators 1/2 IRWST gravity injection lines	$< 10^{-8}$	2.8 bar (41 psia) 740°K (875°F)	no	Leak before break; only proven RCS materials; reliable, redundant ADS, accumulators, CMT, passive injection; PCCS; H_2 igniters
3BL-3	6-inch hot leg break full ADS, PRHR failure 1/2 CMTs and 2/2 accumulators 2/2 IRWST gravity injection lines	$< 10^{-9}$	2.8 bar (41 psia) 650°K (710°F)	Containment failure due to assumed H_2 detonation just after core relocation	Leak before break; only proven RCS materials; reliable PRHR; PCCS; H_2 igniters
3D-2	DVI line break partial ADS, 1/2 CMTs and 0/2 accumulators 0/2 IRWST injection & recirculation lines 1/2 cavity flooding lines, H_2 igniters fail	$< 10^{-9}$	2.3 bar (34 psia) 403°K (265 °F)	Containment failure in CMT room due to assumed diffusion flame prior to core relocation	Leak before break; only proven RCS materials; reliable, redundant CMTs, accumulators, passive injection and recirculation; PCCS; H_2 igniters
6E-2	rupture of 5 SG tubes, SG safety valve fails open at time=0, ADS failure, 0/2 IRWST injection & recirculation lines, 1/2 cavity flooding lines	$< 10^{-9}$	1.7 bar (25 psia) 380°K (225°F)	bypass	Proven tube materials and water chemistry, hot leg temperature less than 400°F, reliable, redundant, diverse ADS, reliable, redundant injection & recirculation
6E-3	rupture of 3 SG tubes, SG safety valve fails to reseat upon atuo opening, ADS failure, 1/2 CMTs and 0/2 accumulators, 0/2 IRWST injection & recirculation lines 0/2 cavity flooding lines	$< 10^{-9}$	2.9 bar (23 psia) 370°K (205°F)	bypass	Proven tube materials and water chemistry; hot leg temperature less than 400°F; reliable, redundant, diverse ADS; reliable, redundant CMTs, accumulators, injection & recirculation, cavity flooding lines

Based on the above described results, the AP600 design addresses severe accident containment challenges as required in Chapter 1, Section 2.6.1 of the URD with significant margin. Functional sequence frequencies are all below 10^{-7} per year and for all sequences, except those employing the most pessimistic phenomenological assumptions, the loads generated are below the URD limits.

B.1.4 Assessment of Uncontrolled Release

For AP600 core damage sequences with adequate cavity flooding and debris coolability, containment overpressure is not expected. Even for sensitivity sequences that are assumed to lead to overpressurization by noncondensable gases, failure is predicted to occur more than 30 hours after the onset of core damage. Thus based on the review of core damage sequences and their potential for uncontrolled releases, it is concluded that the AP600 design has sufficient margin to assure no uncontrolled releases will occur beyond approximately 24 hours.

B.2 Dose Criterion

The dose criterion for emergency planning appears in Chapter 1, Section 2.6.2 of the URD and is repeated in Appendix A. A discussion of the dose criterion appears in the Main Report. A summary of the dose criterion and associated methodology is provided here for completeness.

The criterion is that the dose at 0.5 mile from the reactor due to fission product source term release from a damaged core shall not exceed the Protective Action Guides (PAGs) for approximately 24 hours.

The methodology for demonstrating the dose criterion includes the use of a physically-based source term using release and timing parameters from NUREG 1465 [B-5], a probabilistic dose method (i.e., MACCS 1.5), use of a range of meteorological conditions, and use of effective dose equivalent with a 50 year commitment. The PAGs are projected dose levels for evacuation (1 to 5 rem effective dose) which are specified by the Environmental Protection Agency in a 1992 report [B-6] as guidance for actions to protect the public in the early phase of a nuclear incident.

A physically-based source term, based on NUREG 1465 releases, is used for both DBA applications and for emergency planning applications in the ALWR. The physically-based source term specifies fission product release timing and release magnitude to containment, chemical form of the fission products, fission product removal from containment, and fission product holdup in secondary buildings. The main differences between the DBA application and the emergency planning application of NUREG 1465 are that for emergency planning, the full NUREG 1465 release (i.e., ex-vessel release and late in-vessel release as well as gap and early in-vessel release) is considered, and fission product removal is based on more realistic assumptions (e.g., reasonable credit for non-safety systems).

The emergency planning application utilizing the full NUREG 1465 release is intended to be at the limiting end of the spectrum for PRA source terms from core melt accidents with intact containment. Thus this emergency planning source term should generally envelope potential source terms from PRA intact containment sequences. To confirm this, comparisons of the emergency planning source term have been performed with the MAAP generated source terms from the AP600 PRA for release categories with intact containment or with frequency greater than ~10^{-7} per year.

B.2.1 AP600 NUREG 1465 Source Term Dose Evaluation Methodology

The ALWR emergency planning dose evaluation against the PAGs used NUREG 1465 release fractions. Table B-6 depicts the release fractions from NUREG 1465 which were used here. As noted above, ex-vessel and late in-vessel releases are addressed even though the AP600 includes a cavity flooding system which is designed to flood up around the reactor vessel and which should prevent reactor vessel lower head meltthrough. It is also noted that for the AP600 radiological DBA calculation, slightly different low volatile release fractions were used based upon ALWR Program work [B-7, B-8].

For AP600, the following methodology was used for the PAG dose calculation:

Table B-6
NUREG 1465 PWR Release Fractions to Primary Containment*

Nuclide	Gap Release	Early In-Vessel	Ex-Vessel**	Late In-Vessel***	Total
Duration (hr)	0.5	1.3	2.0	10.0	-
Nobles	0.05	0.95	0	0	1.0
I	0.05	0.35	0.25	0.1	0.75
Cs	0.05	0.25	0.35	0.1	0.75
Te		0.05	0.25	0.005	0.305
Sr, Ba		0.02	0.1	0	0.12
Ru		0.0025	0.0025	0	0.005
Cerium		0.0005	0.005	0	0.0055
Lanthanum		0.0002	0.005	0	0.0052

Notes:
* All numbers are fraction of original core fission product inventory released into the containment.
** The ex-vessel release would be from the ex-vessel debris either to the containment gas space in the volume below the reactor vessel or into a water pool overlying the ex-vessel debris in a flooded cavity design
*** The late in-vessel release is from the fuel remaining in the reactor vessel after lower vessel head meltthrough.

- The requirements of Chapter 1, Section 2.6.4 of the URD were followed. This includes calculating doses at 0.5 miles from the reactor, 24 hour exposure, 50 year inhalation dose commitment, a median effective dose calculation for comparison
- against 1 rem, and a 90 percentile effective dose calculation for comparison against 5 rem.
- The inside containment fission product removal calculation was divided into four intervals: (1) the gap and in-vessel release interval (0 to 1.8 hours); (2) the ex-vessel release inteval (1.8 to 3.8 hours); (3) the late in-vessel release interval (3.8 hours to 13.8 hours); and (3) the remaining 10.2 hours (13.8 hours to 24 hours).
- The AP600 accident management spray system was credited using spray system design information from reference [B-9].
- Auxiliary building holdup was credited, as was done in the AP600 PRA. The decontamination factor (DF) is about 3 based on the AP600 PRA [B-2].
- Cavity water pool scrubbing (affecting ex-vessel release) was credited. The reference [B-10] report was used as the basis for estimating scrubbing DF. A pool depth of about 30 feet would exist in AP600 based on the height of RCS piping relative to the cavity floor together with the fact that cavity flooding is designed to flood up to the RCS piping. For conservatism, it was assumed that the cavity pool depth is 5 meters, and that the pool water is saturated. From reference [B-10], a best estimate DF for a 500 cm, saturated pool is calculated as the natural log of 4.4. Thus, the DF is approximately 80.

Regarding interval (1), the lambda from the AP600 spray system was estimated using the Standard Review Plan aerosol spray removal model [B-11] which is generally found to be conservative compared to more mechanistic models. Using parameters from reference [B-9] (flow 1000 gpm, fall height 25 m, capacity 200,000 gal), and using containment volume 1.6E6 ft^3, the spray lambda was estimated to be about 1.9 hr^{-1}. This lambda is about a factor of 3 larger than that from natural removal calculated for the DBA [B-12] and would result in release from containment about a factor of 3 smaller than calculated for the DBA. From reference [B-12], the core fraction of iodine from the in-vessel source term which was released from containment was about 2.6E-5. Reducing this by 1/3 for the spray lambda and an additional 1/3 for auxiliary building DF, the release to the environment during interval (1) is about 2.9E-6.

In interval (2), the ex-vessel source term is released to containment. From Table B-6, this source term is seen to be about 60% of the in-vessel source term. Reducing this source term by a factor of 80 due to pool DF, and noting that the sprays would continue for most of this 2 hour period, the ex-vessel source term would result in a release from containment of the order of one percent of the in-vessel release. Thus it may be neglected.

In interval (3), the late in-vessel source term is released to containment. The spray system may no longer be assumed to be available. The core fraction of iodine leaked from containment is estimated as follows. The suspended concentration (units of core fraction of iodine) may be expressed as

$$dn / dt = s - \lambda n \qquad (1)$$

where n is suspended concentration of iodine, s is the iodine source (taken to be 0.01 per hour, i.e., 10% release over 10 hours), and λ is removal rate, taken to be 0.6 hr^{-1} for the interval after ~4 hours based on reference [B-12]. Solving equation (1) for n(t), integrating from 0 to 10 hours, and dividing by 10, we obtain an average suspended concentration, $\bar{n}$, over the 10 hours of interval (3) of about 0.014 (fraction of iodine core inventory). The fraction of core inventory released from containment during interval (3) is thus

$$\bar{n}(0.001)(10/24) = 5.8E-6$$

Reducing this by 1/3 for the auxiliary building DF, we obtain ~1.9E-6 for the release to the environment during interval (3).

In interval (4), the aerosol which remains suspended in containment at the end of the late in-vessel release can leak from containment. The initial aerosol at the beginning of interval (4) is obtained by solving equation (1):

$$n(t) = 0.01/0.6 - 0.01/0.6e^{-0.6t}$$

Setting t=10, we obtain n(10) = 0.017. Noting that there is no aerosol source, we may write an equation for suspended core inventory fraction for interval (4) as

$$n(t) = 0.017e^{-\lambda t} \qquad (2)$$

where from reference [B-12], λ is approximately 0.55 for interval (4).

Integrating from 0 to 10.2 hours, and dividing by 10.2, we obtain an average suspended concentration, $\bar{n}$,over the 10 hours of interval (4) of about 0.003 (fraction of iodine core inventory). The fraction of core inventory released from containment during interval (4) is thus

$$\bar{n}(0.001)(10.2/24) = 1.3E-6$$

Reducing this by 1/3 for the auxiliary building DF, we obtain ~0.4E-6 for the release to the environment during interval (4).

Summing the release to the environment for intervals (1) to (4), we obtain the total release (fraction of core inventory of iodine) as

2.9E-6 + 1.9E-6 + 0.4E-6 = 5.2E-6

It is noted that the above result is conservative from the standpoint that the manual containment spray will reduce containment pressure significantly beginning early in the accident, which will reduce containment leakage below the 0.1% per day.

B.2.2 AP600 NUREG 1465 Source Term Dose Evaluation Results

The doses were obtained by extrapolating from MACCS calculated doses in the AP600 PRA [B-2]. With the above release of iodine (and proportional releases of the other fission product groups) a median 24 hour dose of 0.72 rem TEDE and a 90th percentile 24 hour dose of 3.52 rem TEDE is obtained. Thus the doses are less than the URD limits.

B.2.3 AP600 PRA Intact Containment Dose Results

Comparisons of the doses from the NUREG 1465 source term, including early and late in-vessel and ex-vessel releases, have been performed with the source terms from the AP600 PRA for functional sequences with intact containment or with frequency greater than ~10^{-7} per year.

As noted above, none of the AP600 accident classes have frequency greater than 10^{-7} per year. For release category IC (intact containment), the AP600 SSAR considered accident classes 3BE, 3BL, 3BR, 3C, and 3D of which 3BE dominates in frequency at 7.8E-8. Accident Class 3BE also was chosen in the PRA to be the base case for release category IC. The source term for release category IC closely approximates the URD physically based source term with about 45% of the core inventory of iodine (and corresponding amounts of other radionuclides) released to containment. The containment is intact with the containment leak rate taken as the AP600 design leakage. The containment leaks from the penetration area to the middle annulus between the primary and secondary containment shell which results in holdup of fission products and a reduction in offsite dose of a factor of about 3. The dose evaluation was performed as part of the AP600 PRA using the MACCS code assuming that the release occurs at ground level and that 5% of the iodine release to containment is volatile and does not deposit. The median dose after 24 hours from the start of release of fission products from the fuel is about 0.65 rem effective dose. The 90th percentile dose is about 3 rem effective dose.

As indicated in the AP600 PRA, variations on the base case and sensitivity sequences with intact containment have fission product releases that are in some cases slightly higher than the releases for the base case release category, but are still below 1 rem median effective dose and in any case are below 10^{-7} per year in frequency.

On the basis of these PRA results, it is concluded that the NUREG 1465 source term is reasonably bounding for intact containment sequences and thus is an appropriate source term for the PAG comparison.

B.3 SUPPORTING PRA EVALUATION

The requirement for a supporting PRA evaluation for emergency planning appears in Chapter 1, Section 2.6.3 of the URD and is repeated in Appendix A. A summary of the requirement is provided here for completeness.

The requirement is to: (1) demonstrate that the core damage frequency is less than 10^{-5} per year; (2) demonstrate that cumulative frequency for sequences resulting in a dose at 0.5 mile greater than 1 rem for 24 hours is less than 10^{-6} per year, and (3) demonstrate that the prompt accident qualitative health objective of the NRC Safety Goal Policy is met with no credit for offsite evacuation prior to 24 hours.

The supporting PRA evaluations are to be performed in accordance with URD Volume III, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules (KAG) with the exception that the off-site dose exceedance limit is 1 rem. The KAG specifies that the PRA address internal events plus external events with seismic risk to be addressed by the seismic margin approach per Chapter 1, Section 2.5.3.4, of Volume III of the URD. In addition, the KAG specifies quantitative assessment of assessment of uncertainties, including propagation of distributions for Level 1 analysis, consideration of phenomenological uncertainties in Level 2 and 3 analysis, and sensitivity studies to assess the effect of particularly important uncertainties.

Westinghouse has performed a PRA for the AP600 in accordance with the KAG. The total mean frequency of core damage was estimated to be 1.7×10^{-7} per year for internal events at power. For external events the core damage frequency for fires and internal floods was estimated to be about 6.5×10^{-7} per year. Other external events are site specific, but on the basis of design characteristics and features provided to address such events the contribution of these events to core damage frequency is expected to be negligible. For shutdown conditions the core damage frequency was estimated to be 4.1×10^{-7} per year. Thus, the total core damage frequency is 1.2×10^{-6} and has significant margin to the 10^{-5} per year URD goal.

The AP600 complementary cumulative distribution function (CCDF) for offsite dose for 24 hours has been developed in the PRA. The cumulative frequency for sequences resulting in greater than 1 rem at 0.5 mile is approximately 2×10^{-8} per year, thus providing significant margin to the URD 10^{-6}, 1 rem goal.

Based on the PRA performed by Westinghouse for the AP600 design, the AP600 design meets the supplementary PRA requirements with considerable margin.

B.4 CONCLUSIONS REGARDING PASSIVE PLANT CONFORMANCE TO ALWR REQUIREMENTS

Based on this assessment, the AP600 design meets the emergency planning design criteria and supporting PRA evaluation. It is recognized that plant specific designs will continue to evolve during remaining activities required to complete the design and construct the plant. These design evolutions are not expected to impact the conclusions of this assessment. Westinghouse is responsible to demonstrate that their certified designs continue to meet the emergency planning design criteria through the remaining design activities.

B.5 REFERENCES

B-1 Westinghouse Electric Corporation, "AP600 Standard Safety Analysis Report," DE-AC03-90SF18495, June 26, 1992 (Revision 8 dated June 28, 1996).

B-2 Westinghouse Electric Corporation, "AP600 Probabilistic Risk Assessment," DE-AC03-90SF18495, June 26, 1992 (Revision 9 dated April 11, 1997).

B-3 Electric Power Research Institute, "ALWR Utility Requirements Document, " NP-6780-L, Revision 7 dated May, 1996.

B-4 Westinghouse letter to EPRI, August 3, 1999

B-5 L. Soffer, et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG 1465, Final Report, February, 1995.

B-6 "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C., May 1992.

B-7 D. E. Leaver, et al., "Passive ALWR Source Term," DOE/ID-10321, U.S. Department of Energy, Idaho Falls, ID, February, 1991.

B-8 R.R. Hobbins, et al., "Releases to Containment of Low Volatility Fission Products from In-Vessel Processes," DOE Advanced Reactor Severe Accident Program memorandum in support of AP600, February 14, 1995.

B-9 Memo from J. Scobel (Westinghouse) to D. Leaver (Polestar), March, 18, 1998.

B-10 "A Simplified Model of Aerosol Scrubbing by a Water Pool Overlying Core Debris Interacting with Concrete," NUREG/CR-5901, November, 1993.

B-11 NUREG 0800, NRC Standard Review Plan, Section 6.5.2.

B-12 "AP600 Containment Aerosol Calculation Results," Polestar calculation PSAT0902H.03, Rev. 1, April 10, 1997

B-13 "Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites," ANSI/ANS Standard 2.12.

C
ASSESSMENT OF SYSTEM 80+ DESIGN CONFORMANCE WITH ALWR EMERGENCY PLANNING DESIGN CRITERIA

The ABB-Combustion Engineering System 80+ has received design certification under 10 CFR 52. The following assessment of this standard evolutionary advanced light water reactor (ALWR) plant design is provided to describe how the design meets the ALWR emergency planning design criteria. The assessment is based on information contained in the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC) [C-1] and other documentation referenced by the CESSAR-DC. As described in the ALWR Utility Requirements Document (URD) [C-2], Volume II, Chapter 1, Section 2.6, the ALWR emergency planning design criteria include a containment performance criterion and site boundary dose criterion. It is necessary to assess conformance of the design to these criteria. In addition, the URD specifies that a supplementary PRA evaluation be performed in support of the assessment.

C.1 CONTAINMENT PERFORMANCE CRITERION

The containment performance criterion for emergency planning appears in URD Volume I, Chapter 1, Section 2.6.1 and is repeated in Appendix A. A summary of the containment performance criterion is as follows:

> Plant design characteristics and features shall be provided to preclude core damage sequences which could bypass containment and to withstand containment loads from core damage sequences. Containment loads from core damage sequences should be evaluated and should not exceed the combination of peak loads from a DBA LOCA and a hydrogen burn from oxidation of 75% of the active fuel cladding, or not exceed ASME Service Level C limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours, there shall be no uncontrolled release.

The methodology which is specified in the URD for demonstrating the containment performance criterion includes the following:

- Incorporate the design characteristics and features specified in the URD to address severe accident challenges.
- Evaluate containment response for a reference severe accident sequence. This sequence shall be a low pressure core melt into an intact containment with containment systems functioning

as designed to confirm that containment loads do not exceed peak LOCA plus hydrogen loads or ASME Service Level C limits.

- Evaluate containment response for functional sequence types with mean frequency $>10^{-7}$ per year to confirm that containment leaktightness and structural integrity are maintained, and that containment loads do not exceed peak LOCA plus hydrogen loads or ASME Service Level C limits. Functional sequence types with mean frequency $< 10^{-7}$ per year should be reported for discussion including a description of the plant features credited to reach this low frequency.
- Provide protection of the containment for overpressurization beyond 24 hours. Overpressure protection may be provided by the size and strength of the containment. On the order of two to three days is judged to be adequate time for actions by the plant staff to bring the accident under control.

The steps used for the assessment of System 80+ compliance with the containment performance criterion were as follows:

1. Confirm that the design meets the requirements of the URD, Volume II, Chapter 5, Section 6.6.2.1 by performing a comparison between the System 80+ design characteristics and features and the requirements.

2. Confirm that containment loads from a System 80+ reference core damage sequence do not exceed peak limits as specified in the URD, Volume II, Chapter 5, Section 6.6.2.2 for approximately 24 hours under realistic severe accident assumptions.

3. Confirm as specified in the URD, Volume II, Chapter 5, Section 6.6.2.3 that containment leaktightness and structural integrity are maintained in response to System 80+ functional severe accident sequences with frequency $> 10^{-7}$ per year and that containment loads do not exceed limits specified in the URD, Volume II, Chapter 5, Section 6.6.2.2. In addition, functional severe accident sequences with frequency $< 10^{-7}$ per year are reported with a description of plant features credited to reach this low frequency.

4. Confirm that the System 80+ design meets the URD, Volume II, Chapter 5, Section 6.6.2.5 and that no uncontrolled release will occur beyond approximately 24 hours.

C.1.1 Plant Design Characteristics and Features to Address Containment Challenges

The first step in the assessment was performed by reviewing the System 80+ CESSAR-DC to confirm, for each containment challenge, the existence of specific design features or characteristics to fulfill the key URD requirements associated with the challenge. The list of challenges and associated URD requirements summarized in Appendix A, Tables A.5-1 and A.5-2 was used for this review. A requirement was considered met when an explicit reference to the system, feature, or characteristic was made in the CESSAR-DC.

Table C-1 summarizes the results of the assessment for System 80+. This table lists the challenges and associated requirements, and identifies in brackets the sections of the System 80+ CESSAR-DC which address each requirement. Specific CESSAR-DC design features or capabilities were identified in response to all of the requirements. Additional information was requested from ABB for the items listed in Table C-2. The reference [C-3] ABB-CE letter provides this additional information.

On the basis of the assessment, the System 80+ design meets the requirements of Volume II, Chapter 5, Section 6.6.2.1 of the URD.

C.1.2 Comparison of Loads from a Reference Severe Accident against URD Limits

Since the System 80+ meets the URD provisions related to containment challenges, the reference severe accident sequence for which containment loads should be compared with URD limits is a low pressure core melt accident into an intact containment with the reactor coolant system (RCS) at low pressure and containment systems functioning as designed.

The System 80+ PRA (Chapter 19, Section 19.11.5.4 of the CESSAR-DC) includes MAAP evaluations for station blackout, a smaller large break LOCA, small break LOCA, loss of feedwater, steam generator tube rupture, and V sequence. The smaller large break LOCA has been selected as the reference low pressure sequence. This is sequence LL-3E which is discussed in 3/VR grouping of plant accident sequences. The accident is initiated by a 0.5 ft^2 break with subsequent failure of safety injection. This plant accident sequence grouping (or “Functional Sequence”, as defined for the purpose of this assessment) represented by this LL-3E case is the largest contributor to the System 80+ core damage frequency, representing about 30% of the total core damage frequency. Table C-3 presents all of the 16 plant accident sequence groupings (or “Functional Sequences”) for the System 80+ PRA.

Table C-1
Assessment of System 80+ Design Conformance With ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
1. Containment Isolation	Isolation	• Isolation provisions and leakage rate testing per standards [6.2.4.1.2-A,G,H & 6.2.4.4]. • Valves capable of closure with maximum flow and full containment pressure [6.2.4.1.2-B,L & 6.2.4.3 - see Table C-2]. • Control room position indication for automatic and remote manual valves [Table 7.5-2]. • Manual valve configuration permits locking only in fully closed position [See Table C-2]. • Closed systems penetrating containment evaluated for ex-vessel severe accidents [19.12.2.2.6.1.1.2.1, .2][1]. • Fail closed or DC powered isolation valves [Table 6.2.4-1 - see Table C-2]. • Capability for periodic gross check of containment integrity [6.2.6.1].	
2. Interfacing System LOCA	Bypass	• Reduced interfaces between the Reactor Coolant System (RCS) and low pressure systems [App. 5E]. • High to low pressure interfaces provided with isolation valve leak testing capability, isolation valve position indicator in control room, and high pressure alarm [App. 5E - see Table C-2]. • Interlocks prevent isolation valve opening when RCS pressure exceeds RSDC system design pressure [App. 5E - see Table C-2]. • Double isolation [App. 5E - see Table C-2].	• Pressure Relief [App. 5E - see Table C-2]. •Design pressure such that full RCS pressure is below rupture pressure and no leaks will occur which exceed RCS makeup capacity [App. 5E - see Table C-2].
3. Blowdown Forces	Containment Pressure Control	• Design and ISI in accordance with ASME BPV Code [5.2.1.2]. • Leak Before Break [3.6.3].	• Design containment for double-ended guillotine break of largest pipe [Table 6.2.1-1].

[1] For SG tube integrity (i.e., SG and associated piping is only closed system inside containment credited for containment isolation).

Table C-1 (Continued)
Assessment of System 80+ Design Conformance With ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
4. Pipe Whip and Jet Impingement	Bypass	• Design and ISI in accordance with ASME BPV Code [5.2.1.2]. • Leak Before Break [3.6.3]. • Use of only proven materials and fabrication processes [5.2.3.2.2 & 5.2.3.4]. • Use of EPRI water chemistry guidelines [9.3.4].	• Protection from jet/pipe whip where leak before break is not demonstrated [3.6.2].
5. Steam Generator Tube Rupture	Bypass	• Improved water chemistry [9.3.4 & 10.3.5]. • Proven materials [5.2.3.2.2]. • Mechanical design of tubes, tube supports, and tube sheets reduce likelihood of SGTR [5.4.2.4.1]. • Improved design features facilitate SG cleaning and replacement [5.4.2.2 & 10.4.8 - see Table C-2].	• Depressurization system operation terminates tube leakage [App. 5F - see Table C-2].
6. ATWS	Reactivity Control	• Diverse RPS (or capability to ride out ATWS) [7.7.1.1.11 & 19.4.13].	• Borated Safety Injection (SI) [6.3.3.1]. • Negative moderator temperature coefficient over entire fuel cycle improves ATWS response [Figure 19.5-2].
7. Suppression Pool Bypass	Containment Pressure Control	NOT APPLICABLE	
8. Catastrophic RPV Failure	Internal Containment Loading	• $RT_{NDT} \leq 10°F$; initial $RT_{NDT} \leq -10°F$ for PWR core beltline; low fluence at vessel wall [5.2.2.11, 5.3.2.1 & 5.3.3 - see Table C-2]. •Phosphorous, copper, vanadium, and sulfur limits on material in high fluency region [5.2.3.1] • No welds in beltline region [5.2.3.1] • Primary and secondary safety valves prevent overpressure, depressurization system and low-head injection backup [App 5A] • Design in accordance with ASME code [5.3.1.1]. • Design features to avoid relief valve opening for expected plant transients [5.4.10.1]	
9. Internal Vacuum	Containment Pressure Control		Design for external pressure loads [6.2.1.1.3.6].

Table C-1 (Continued)
Assessment of System 80+ Design Conformance With ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
10. Internal (Plant) Missiles	External Containment Loading	• Turbine overspeed protection [10.2.1]. • Improved turbine integrity/one-piece rotors [10.2.3- see Table C-2].	•Turbine orientation avoids missile contact with containment [10.2.1]. • Missile protection for any safety related components in missile path (SRP 3.5.1.3) [3.5.3].
11. Tornado and Tornado Missiles	External Containment Loading	• Conformance with ANSI 2.12 and ANSI 51.5 (in accordance with ASCE 7-88, "Minimum Design Loads for Buildings and other Structures," formerly ANSI A58.1-82) [19.7.1.12].	
12. Man-Made Site Proximity Hazards	External Containment Loading	• Plant site outside the radius of influence of potential hazards [19.7.1.13].	
13. Seismic	External Containment Loading	• Siting requirements exclude the most vulnerable sites [no effect on design].	• SSE at 0.3g [19.7.1.18]. • Evaluation at > SSE with margins assessment as part of design process [19.7.5]. • Address vulnerabilities from past experience, e.g., provide common basemat [19.7.5].
14. High Pressure Melt Ejection (HPME)	Reactor Pressure Control	• Safety depressurization system [19.11.3.5].	• Cavity configuration to limit transport of fragmented core debris [19.11.3.6.2 and 19.11.4.1.1.2].
15a. Hydrogen Generation to Detonable Limits	Combustible Gas Control	• Limit H_2 generation with design features, such as ADS and cavity flooding [6.7, 6.8.2.2.3, 19.11.3.3, & 19.11.3.5]. • Hydrogen control system designed to keep hydrogen concentration below 10% for 100% active clad equivalent reaction [6.2.5.1.2]. • Containment size prevent global detonable H_2 concentration (<13%) for generation up to 75% active clad equivalent reaction [19.11.4.1.3.1.4.1]. • Design ensures convective mixing and minimizes DDT-prone geometry [19.11.4.1.3.1.4.2, App. 19.11K, & Table 19.11.4.1.3-6].	• Evaluation required if local detonation is possible [19.11.4.1.3.2.5].

Table C-1 (Continued)
Assessment of System 80+ Design Conformance With ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS: LIMIT POTENTIAL FOR CHALLENGE	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS: ACCOMMODATE CHALLENGE
15b. Hydrogen Deflagration	Combustible Gas Control	• Recombination or deflagration likely at low concentrations (<10%) given hydrogen control system (IRWST and PCCS limit steam inerting potential) [App. 19.11K].	•Demonstrated accommodation of generation equivalent to 100% active clad reaction with multiple burns [App 19.11K]. • Structural evaluation for LOCA plus hydrogen loads, including global burn of hydrogen equivalent to 75% active clad reaction [3.8.2.4.3 & 19.11.4.1.3.1.4.1].
16. In-Vessel Debris-Water Interaction	Internal Containment Loading	• Large-scale phenomena limited in probability [19.11.4.1.2.1]. • In-vessel geometry limits interacting quantities and size of any interaction [19.11.4.1.2.1].	• Rugged reactor vessel contains forces [19.11.4.1.2.1]; as backup, rugged reactor cavity contains lower head failure [19.11.4.1.2.2].
17. Ex-Vessel Debris-Water Interaction	External Containment Loading	• Large-scale phenomena limited in probability [19.11.4.1.2.2]. • Ex-vessel geometry limits interacting quantities and size of any interaction [19.11.4.1.2.2].	• Rugged reactor cavity confirmed by evaluation [19.11.3.6.2.7 & 19.11.4.1.2.2.2.5]. • Containment design accommodates steam generation [19.11.4.1.2.3 & Table 19.11.4.1.2-4].
18. Noncondensible Gas Generation	Fuel/Debris Cooling	• Features limiting concrete erosion (see item 19) limit noncondensible gas generation as well. • Overlying pool cools gases from core-concrete interaction [19.11.4.2.2].	• Containment size and pressure retention capability [19.11.3.6.2.10 & 19.11.4.2.1.2.3].
19. Basemat Erosion and Vessel Support Degradation	Fuel/Debris Cooling	• Reactor cavity/lower drywell spreading area of $0.02 m^2$/MWt promotes core debris cooling [19.11.3.6.2.5 & 19.11.4.2.2.4 - see Table C-2]. • Reactor cavity flooding [6.8.2.2.4 & 19.11.3.3]. • Overflow from containment reflux via PWR IRWST prefloods reactor cavity [19.11.3.3.2]. • Backup capability for water addition from sources external to containment [6.5.5].	• Sacrificial concrete where debris on floor contacts boundary structures [19.11.3.6.2.6 & 19.11.4.2.2.4].

Table C-1 (Continued)
Assessment of System 80+ Design Conformance With ALWR Requirements Which Address Containment Challenges

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
20. Core Debris in Sump	Fuel/Debris Cooling	• Special cavity sump design prevents localized unterminated core-concrete interaction [19.11.3.6 & 19.11.4.2.2.4]. • Sump drainline configuration precludes gravity transport of debris ex-containment [19.11.3.6.2.6 & 19.11.4.2]. • Reactor cavity flooding [6.8.2.2.4 & 19.11.3.3].	
21. Core Debris Contact with Liner	Fuel/Debris Cooling	• Liner protected by concrete [19.11.3.6.2.6 & 19.11.4.2.2.4]. • Reactor cavity flooding [6.8.2.2.4 & 19.11.3.3]. • Design features to limit debris dispersal including ADS [6.7, 19.11.3.5, & 19.11.3.6.2].	
22. Decay Heat Generation		• Steam Generators/Main Feedwater (MFW)/Startup Feedwater [10.4.7 & 10.4.9]. • Normal Residual Heat Removal System [5.4.7].	
23. Tube Rupture from Hot Gases		• Steam Generators/MFW/Startup Feedwater [10.4.7 & 10.4.9]. • Depressurization System [6.7]	

Table C-2
CESSAR-DC for Which Additional Information was Rquested

Containment Isolation:

1. Closure against possible flow discussed only for containment purge valves.
2. Locking only in closed position not addressed.
3. Several normally-open, fail-as-is valves (Table 6.2.4-1, Items 23 - 26, 30 - 35, 91-92) are listed.

Interfacing System LOCA:

1. Incomplete interface control (see matrix below, based on Appendix 5E and Section 7.6.1.1):

Interface	Leak test capability?	Isolation valve PI in CR?	High pressure alarm?	Interlock on pressure?	Double valve isolation?
SCS supply	Y	Y	Y	Y	Y
SCS return	Y	Y	Y	Y	Y
SIS delivery line	Y	Y	Y	N	Y
Letdown line	Y	Y	Y	Y	Y
Charging line	Y	Y	Y	Y	Y
Seal injection filter vent/drain (a)	N	N	Y	N	N
RCP controlled bleedoff (b)	Y	Y	Y	N	Y
Hot leg sampling (c)	Y	Y	Y	N	Y
Pressurizer surge line sampling (c)	Y	Y	Y	N	Y
Pressurizer steam space sampling (c)	Y	Y	Y	N	Y

Notes: (a) reduced line size prevents coolant loss > makeup and overpressure in equipment drain tank (also, CR pressure alarm on tank)
(b) normally open line with sufficient "relief" (i.e., flow) capability installed (also, CR pressure alarm on volume control tank)
(c) orificed line with relief to equipment drain tank (with pressure alarm in CR)

2. Positive pressure relief provided only for last five entries in table above (to protect portions of piping not designed for 40% of RCS pressure).
3. Minimum design pressure of 900 psig (40% of RCS pressure, sufficient to preclude rupture) provided only for first five entries in table above (in lieu of positive pressure relief).

Steam Generator Tube Rupture

1. Improved design features for SG replacement not addressed.

Catastrophic RPV Failure

1. Initial RT_{NTD} < -10 F for core beltline not committed to; goal stated for initial RT_{NTD} < -20 F.
2. Low fluence at vessel wall not commented upon.

Internal (Plant) Missiles

1. Welded rotor (not one-piece).

There are two important points to be made regarding the selection of sequence LL-3E as the reference sequence. First, while the station blackout, small LOCA, and loss of feedwater sequences in CESSAR-DC Section 19.11.5.4 have RCS pressure above 250 psig at the time of vessel failure (and thus as evaluated in the PRA would not be low pressure core melt accidents), no credit was taken in these sequences for accident management action to actuate the safety depressurization system (SDS) which would be expected to depressurize the RCS to below 250 psig by the time of vessel failure. Thus, use of a low pressure sequence as the dominant sequence for System 80+ is a reasonable assumption. Furthermore, the presence of the offset core debris chamber in System 80+ (see CESSAR-DC Section 19.11.3.6.2.4) would limit transport of fragmented core debris from a high pressure melt ejection, thus reducing the effect of RCS pressures above 250 psig on containment integrity.

Second, for conservatism in evaluating containment loads and fission product releases, ABB assumed containment spray (and thus containment heat removal) failure in the MAAP evaluation of the smaller large break LOCA sequence (i.e., LL-3E is sequence LL-3 with containment safeguard end state E which is containment spray unavailable - see CESSAR-DC Table 19.12.1-4). This is not consistent with the reference sequence definition for plants which meet the URD requirements for containment challenges (i.e., containment systems functioning as designed). However, since sequence LL-3E is the dominant low pressure sequence for which containment loads have been calculated in the System 80+ PRA and since the assumption of containment spray unavailable is conservative, the containment loads for LL-3E will be used for purposes of the comparison to URD limits.

In the reference LL-3E sequence the break causes the reactor coolant system pressure to drop. Low pressure causes the reactor to scram and the safety injection tanks to discharge. The safety injection tank discharge maintains core cooling initially; but the safety injection system fails to inject coolant, and the core begins to uncover at about 30 minutes. The containment spray system fails to start on high containment pressure, and all containment heat removal is lost. Flooding of the reactor cavity is successful, and hydrogen produced from metal oxidation is controlled by igniters.

Table C-3
Summary of System 80+ Plant Accident Sequence Groupings (Functional Sequences) for Leve 2 PRA Analysis

PLANT ACCIDENT SEQUENCE GROUPINGS (see Table 19.12.1-6 of CESSAR-DC for sequences with frequences $\geq 10^{-9}$ / year)	LEVEL 1 DEFINITION	LEVEL 2 DEFINITION	COMBINED CORE DAMAGE FREQUENCY per year	CONTAINMENT MAXIMUM 48 HR PRESSURE AND SUSTAINED TEMPERATURE	CONTAIN-MENT FAILURE	DESIGN FEATURES RE-SULTING IN FREQUENCY $<10^{-7}$
LL-4A	Large LOCA, SIT fails	Sprays and CHR function, Wet Cavity	4.4×10^{-9}	N/A	N/A	• Leak before break • Use only proven materials in RCS • Reliable, redundant SITs
LL-3A, LL-3B, ML1-3A, ML1-3B, ML2-3A, ML2-3B, VR-A	Med/Large LOCA, SI fails or Vessel Rupture	Sprays and CHR function, Wet or Dry Cavity	5.0×10^{-7}	9.3 bar (136 psia) 450 K (350 F) Case: LL-3E	~ 60 hours for case without sprays/CHR(1,2 .3)	N/A
SL-4A	Small LOCA, LTDHR fails, Feed/Bleed fails	Sprays and CHR function, Wet Cavity	8.5×10^{-9}	N/A	N/A	• Use only proven materials in RCS • Reliable RCS pump seal design [LATER]
SL-10A, SL-11A, SL-11B	Small LOCA, SI and ASC or SCS Injection fail	Sprays and CHR function, Wet or Dry Cavity	1.7×10^{-7}	~1.4 bar (20 psia) ~340 K (150 F) Case SL-11F	None(2,3)	N/A
SL-11E	Small LOCA, SI and ASC fail	Sprays fail, Wet Cavity	2.6×10^{-9}	N/A	N/A	• Use only proven materials in RCS • Reliable RCS pump seal design • Reliable, redundant SI system • Reliable, redundant CSS • CSS independent from SI
SGTR-8A	SGTR, Unisolable Leak, MSHR fails (leading to loss of decay heat removal from core)	Sprays and CHR function, Wet Cavity	3.0×10^{-9}	N/A	Bypass	• Proven tube materials and water chemistry; improved tube design • Block valves on ADV lines • Auto actuation and improved reliability of turbine bypass after SIAS • High capacity SG blowdown to MC

Table C-3 (Continued)
Summary of System 80+ Plant Accident Sequence Groupings (Functional Sequences) for Leve 2 PRA Analysis

PLANT ACCIDENT SEQUENCE GROUPINGS (see Table 19.12.1-6 of CESSAR-DC for sequences with frequences $\geq 10^{9}$ / year)	LEVEL 1 DEFINITION	LEVEL 2 DEFINITION	COMBINED CORE DAMAGE FREQUENCY per year	CONTAINMENT MAXIMUM 48 HR PRESSURE AND SUSTAINED TEMPERATURE	CONTAINMENT FAILURE	DESIGN FEATURES RE-SULTING IN FREQUENCY $<10^{-7}$
SGTR-9F	SGTR, Unisolable Leak, RCS Pressure Control and IRWST Refill fail	Sprays fail, Dry Cavity	3.1×10^{-8}	N/A	Bypass	• Proven tube materials and water chemistry; improved tube design • Block valves on ADV lines • RDS • High bypass SI pumps • CVCS capability to refill IRWST from BAST
SGTR-12A, SGTR-16A, SGTR-17A, SGTR-17B	SGTR, EFW and Feed/Bleed fails or SI and ASC or SCS Injection fail	Sprays and CHR function, Wet or Dry Cavity	2.8×10^{-7}	~1.4 bar (20 psia) ~340 K (150 F) Case SGTR-12A /SGTR-15A	None(2,3)	N/A
SGTR-17E	SGTR, SI and ASC fail	Sprays fail, Wet Cavity	1.2×10^{-8}	N/A	~ 70 hours	• Proven tube materials and water chemistry; improved tube design • Reliable, redundant SI system • Reliable, redundant CSS • CSS independent from SI
SGTR-17F	SGTR, SI and ASC fail	Sprays fail, Dry Cavity	1.0×10^{-9}	N/A	~70 hours	• Proven tube materials and water chemistry; improved tube design • Reliable, redundant SI system • Reliable, redundant CSS • CSS independent from SI
LSSB-9A, LOFW-8A, LOFW-8E, LOFW-9A, LOFW-9B, LOFW-9E, LOFW-9F, TOTH-9A, LOOP-9A, ATWS-9A	Transient, EFW or SCRAM/Boron Delivery fail and Feed/Bleed fails	Sprays and CHR function or Sprays fail, Wet or Dry Cavity	4.7×10^{-7}	7.6 bar (112 psia) 460 K (370 F) Cases LOFW-9E /LOOP-9E	~ 70 hours for case without sprays/CHR(2,3)	N/A

Table C-3 (Continued)
Summary of System 80+ Plant Accident Sequence Groupings (Functional Sequences) for Leve 2 PRA Analysis

PLANT ACCIDENT SEQUENCE GROUPINGS (see Table 19.12.1-6 of CESSAR-DC for sequences with frequences ≥ 10^{-9} / year)	LEVEL 1 DEFINITION	LEVEL 2 DEFINITION	COMBINED CORE DAMAGE FREQUENCY per year	CONTAINMENT MAXIMUM 48 HR PRESSURE AND SUSTAINED TEMPERATURE	CONTAINMENT FAILURE	DESIGN FEATURES RESULTING IN FREQUENCY <10^{-7}
LOFW-4A, LOFW-4B, LOFW-4E, LOFW-5A, TOTH-4A, TOTH-4B, TOTH-4E, TOTH-5A, SBOBD-E, SBOBD-F, LHV-5A	Transient, LTDHR fails and Feed/Bleed fails or EFW lost after Battery Depletion	Sprays and CHR function or Sprays fail, Wet or Dry Cavity	1.4×10^{-7}	Similar to LSSB-9A ex-cept slower progression - 5.7 bar (84 psia) 430 K (315 F) Case SBOBD-E	Similar to LSSB-9A ex-cept slower progression - ~83 hours <u>without</u> sprays/CHR[(2,3)]	N/A
TOTH-12A, TOTH-12B	Transient/LOCA due to PSV failure, SI fails	Sprays and CHR func-tion, Wet or Dry Cavity	2.6×10^{-8}	N/A	N/A	• PSV designed and tested for high reliability • Reliable, redundant SI system
ATWS-29A, ATWS-29B	Transient/LOCA due to SCRAM failure and Adverse MTC	Sprays and CHR func-tion, Wet or Dry Cavity	4.6×10^{-8}	N/A	N/A	• Extremely reliable scram system • Diverse boron injection shutdown system • Core design for negative MTC during all operating conditions
ATWS-29E	Transient/LOCA due to SCRAM failure and Adverse MTC	Sprays fail, Wet Cavity	1.4×10^{-9}	N/A	N/A	• Extremely reliable scram system • Diverse boron injection shutdown system • Core design for negative MTC during all operating conditions
ISL-F	Interfacing System LOCA Outside Containment	Sprays fail, Dry Cavity	1.0×10^{-9}	N/A	Bypass	• Interfacing system designed for RCS pressure • RCS interfaces have isolation valve leak test capability. position indication in CR, and high pressure alarm • Interlocks prevent isolation valve opening at high RCS pressure

Table C-3 (Continued)
Summary of System 80+ Plant Accident Sequence Groupings (Functional Sequences) for Leve 2 PRA Analysis

Note 1 - For LL-3, ML1-3, and ML2-3 Core Damage Sequences there are no failed Spray or CHR Containment Safeguard End States associated with Plant Accident Sequences with frequencies greater than 10^{-9} per year. Therefore, it is very conservative to even partially characterize this group's containment response using such a sequence.

Note 2 - For cases with Dry Cavity (e.g., LL-3F, LOOP-9F) basemat melt-through would be expected in about 8 days, although debris contact with the embedded liner could occur in approximately one day. Dry Cavity sequences are approximately one order of magnitude less likely than corresponding Wet Cavity sequences.

Note 3 - Inclusion of containment isolation failure would reduce functional sequence frequency to of the order of 10^{-9} /yr.

ADV - atmospheric dump valves
ASC - alternate shutdown cooling
BAST - boric acid storage tank
CHR - containment heat removal
CI - containment isolation
CSS - containment spray system
CVCS - chemical/volume control system
IRWST - in-containment refueling water storage tank
LOCA - loss of coolant accident
LTDHR - long-term decay heat removal
MC - main condenser
MSHR - main-steam heat removal
MSSV - main-steam safety valve
MTC - moderator temperature coefficient
PSV - primary safety-valve
RDS - rapid depressurization system
SCS - shutdown cooling system
SG - steam generator
SGTR - steam generator tube rupture
SI - safety injection
SIAS - safety injection actuation signal
SIT - safety injection tank

The containment pressure and temperature after 24 hours for LL-3E are 100 psia (6.8 bar) and 320°F (433 °K) respectively, considerably less than the ASME Service Level C limits of about 140 psia at 360 F (see CESSAR-DC Table 19.11.5.4.2.1-2, Figure 19.11.5.4.2.1-5, and Figure 19.11.3.1-2). In actuality, these LL-3E containment pressure and temperature results greatly exceed the expected results for a reference sequence because the reference sequence definition includes sprays and containment heat removal as noted above. In fact, the frequency for LL-3E (without sprays or heat removal) is less than the 10^{-9} per year cutoff frequency for reporting plant accident sequences in CESSAR-DC Table 19.12.1-6 from which Table C-3 was prepared. If sprays and containment heat removal were properly included, the peak pressure and temperature would be much lower. Thus the containment design provides substantial margin to the ASME Service Level C limits.

From this assessment, it is concluded that the System 80+ design meets the reference severe accident sequence portion of the URD, Volume II, Chapter 5, Section 6.6.2.2 containment performance criterion. The containment load from the reference sequence does not exceed ASME Service Level C limits for the System 80+ containment for approximately 24 hours under realistic severe accident assumptions.

C.1.3 Evaluation of Containment Response for Functional Sequence Types

The URD requires in Volume II, Chapter 5, Section 6.6.2.3 that containment leaktightness and structural integrity be analyzed in response to functional severe accident sequences with frequency $>10^{-7}$ per year, and that containment loads not exceed the limits described in the URD, Volume II, Chapter 5, Section 6.6.2.2. In addition, functional severe accident sequences with frequency $<10^{-7}$ per year are to be reported with a description of plant features credited to reach this low frequency. Finally, the frequencies of those sequence types which may result in early containment failure are to be identified.

There are 16 functional sequences derived from Table 19.12.1-6 of the CESSAR-DC and these are shown in Table C-3. These 16 functional sequences encompass the numerous accident sequences found in the Level 1 PRA.

As shown in Table C-3, there are five functional sequences with core damage frequencies estimated to be greater than 10^{-7} per year. Table C-3 shows that two of these five functional sequences (SL-11F and SGTR-12A/SGTR-15A) have representative analysis cases with very low containment loads relative to ASME Service Level C limits. The remaining three representative analysis cases for functional sequences with frequencies greater than 10^{-7} per year (LL-3E, LOFW-9E/LOOP-9E, and SBOBD-E) are for containment spray and containment heat removal unavailable (i.e., LL-3E) or only making up a very small part of the sequence frequency (i.e., LOFW-9E/LOOP-9E and SBOBD-E). Thus, as explained above, these three functional sequences would exhibit containment pressures and temperatures much less than that shown in Table C-3 if the representative analysis cases were less conservative. However, even with the conservative analysis, the pressures at 24 hours are well below ASME Service Level C limits (100 psia and 320 F for LL-3E, 64 psia and 286 F for LOFW-9E/LOOP-9E, and 36 psia and 230 F for SBOBD-E - see CESSAR-DC Tables 19.11.5.4.2.1-2, 19.11.5.4.1.1-2, and 19.11.5.4.1.3-2, respectively). In addition, Table C-3 provides a list of features credited in the PRA analysis which are responsible for the very low probabilities of the functional sequences with frequencies $<10^{-7}$ per year.

Note that containment failure (within a time frame of two to three days) is not predicted for any functional sequence with a frequency greater than 10^{-7} per year. Indeed, there is only one functional sequence with a frequency greater than 10^{-8} per year which involves a containment failure, and that is the SGTR sequence SGTR-9F. In the PRA this sequence is mapped into Plant Damage State 194 and Release Category RC4.36L, and while the S80SOR-predicted source term for this release category is relatively large, the start of release is not until 25 hours.

Based on the results described above the System 80+ design addresses severe accident containment challenges as required in Chapter 1, Section 2.6.1 of the URD with significant margin. Functional sequence frequencies exceeding 10^{-7} per year do not exhibit containment loading greater than ASME Service Level C and do not result in containment failure. Only one sequence with a frequency of the order of 10^{-8} per year exhibits containment failure (a SGTR sequence), and all other sequences exhibiting containment failure have frequencies of the order of 10^{-9} per year or less.

C.1.4 Assessment of Uncontrolled Release

As shown in Table C-3, for System 80+ core damage sequences with adequate cavity flooding and containment heat removal, no containment overpressure is expected. Even for sensitivity sequences that are assumed to lead to overpressurization due to loss of containment heat removal or to basemat penetration, failure is predicted to occur much later than 48-60 hours after the onset of core damage. Thus based on the review of core damage sequences and their potential for uncontrolled releases, it is concluded that the System 80+ design has sufficient margin to assure no uncontrolled releases will occur beyond approximately 24 hours.

C.2 Dose Criterion

The dose criterion for emergency planning appears in Chapter 1, Section 2.6.2 of the URD and is repeated in Appendix A. A discussion of the dose criterion appears in the Main Report. A summary of the dose criterion and associated methodology is provided here for completeness.

The criterion is that the dose at 0.5 mile from the reactor due to fission product source term release form a degraded core shall not exceed the Protective Action Guides (PAGs) for approximately 24 hours.

The methodology for demonstrating the dose criterion includes the use of a physically-based source term using release and timing parameters from NUREG 1465 [C-4], a probabilistic dose method (i.e., MACCS 1.5), use of a range of meteorological conditions, and use of effective dose equivalent with a 50 year commitment. The PAGs are projected dose levels for evacuation (1 to 5 rem effective dose) which are specified by the Environmental Protection Agency in a 1992 report [C-5] as guidance for actions to protect the public in the early phase of a nuclear incident.

A physically-based source term, based on NUREG 1465 releases, is used for both DBA applications and for emergency planning applications in the ALWR. The physically-based source term specifies fission product release timing and release magnitude to containment, chemical form of the fission products, fission product removal from containment, and fission product holdup in secondary buildings. The main differences between the DBA application and the emergency planning application of NUREG 1465 are that for emergency planning, the full NUREG 1465 release (i.e., ex-vessel release and late in-vessel release as well as gap and early in-vessel release) is considered, and fission product removal is based on more realistic assumptions (e.g., reasonable credit for non-safety systems).

The emergency planning application utilizing the full NUREG 1465 release is intended to be at the limiting end of the spectrum for PRA source terms from core melt accidents with intact containment. Thus this emergency planning source term should generally envelope potential source terms from PRA intact containment sequences. To confirm this, comparisons of the emergency planning source term have been performed with the S80SOR-generated source terms from the System 80+ PRA for release categories with intact containment or with frequency greater than ~10^{-7} per year.

C.2.1 SYSTEM 80+ NUREG 1465 Source Term Dose Evaluation Methodology

Table C-4 depicts the release fractions from the final NUREG 1465. Table C-5 depicts release fractions based upon an earlier draft version of NUREG 1465 [C-6] which were used in the CESSAR-DC for the System 80+ emergency planning dose evaluation against the PAGs. The differences are in the low volatile releases. As discussed in the Main Report, the final NUREG 1465 releases are intended for use in the PAG dose calculation; what System 80+ has done (and presented for NRC review in Section 15.6.5 of CESSAR-DC) is conservative.

For System 80+, CESSAR-DC reported the following for the PAG dose calculation:

- Requirements of Chapter 1, Section 2.6.4 of the URD were followed. This included calculating the median effective doses at 0.5 miles from the reactor, 24 hour exposure, 50 year inhalation dose commitment, for comparison against 1 rem.
- Aerosol hygroscopicity and cavity water pool scrubbing (affecting ex-vessel release) were credited.

C.2.2 SYSTEM 80+ NUREG 1465 Source Term Dose Evaluation Results

The MACCS 1.5 computer code was used for the dose evaluation. The median effective dose was calculated to be 0.33 rem. The 90th percentile effective dose was not reported in CESSAR-DC, but based on numerous MACCS calculations performed on System 80+ and other plants as part of ALWR PRA and emergency planning work, the 90th percentile effective dose can be conservatively estimated as a factor of about 5 times the median dose, or 1.65 rem. This is well under the URD 5 rem limit for the 90th percentile effective dose. Thus the doses are less than the corresponding PAGs.

C.2.3 SYSTEM 80+ PRA Intact Containment Source Term Comparison

Comparisons of the System 80+ PAG calculation source term have been made with the source terms from the System 80+ PRA for release categories with intact containment or with

Table C-4
Final NUREG 1465 PWR Release Fractions to Primary Containment*

Nuclide	Gap Release	Early In-Vessel	Ex-Vessel**	Late In-Vessel***	Total
Duration (hr)	0.5	1.3	2.0	10.0	-
Nobles	0.05	0.95	0	0	1.0
I	0.05	0.35	0.25	0.1	0.75
Cs	0.05	0.25	0.35	0.1	0.75
Te		0.05	0.25	0.005	0.305
Sr, Ba		0.02	0.1	0	0.12
Ru		0.0025	0.0025	0	0.005
Cerium		0.0005	0.005	0	0.0055
Lanthanum		0.0002	0.005	0	0.0052

Table C-5
Draft NUREG 1465 PWR Release Fractions to Primary Containment*

Nuclide	Gap Release	Early In-Vessel	Ex-Vessel**	Late In-Vessel***	Total
Duration (hr)	0.5	1.3	2.0	10.0	-
Nobles	0.05	0.95	0	0	1.0
I	0.05	0.35	0.25	0.1	0.75
Cs	0.05	0.25	0.35	0.1	0.75
Te		0.15	0.29	0.025	0.465
Sr, Ba		0.03, 0.04	0.12, 0.10	0	0.15, 0.14
Ru		0.008	0.004	0	0.012
Cerium		0.01	0.02	0	0.03
Lanthanum		0.002	0.015	0	0.017

Notes:
* All numbers are fraction of original core fission product inventory released into the containment.
** The ex-vessel release would be from the ex-vessel debris either to the containment gas space in the volume below the reactor vessel or into a water pool overlying the ex-vessel debris in a flooded cavity design
*** The late in-vessel release is from the fuel remaining in the reactor vessel after lower vessel head meltthrough.

frequencies greater than ~10-7 per year. Release Categories RC1.1E and RC1.1M are the only System 80+ PRA release categories with frequencies greater than 10^{-7} per year (i.e., 1.36E-6 and 3.83 E-7 per year, respectively). They are also the only two intact containment release categories. The System 80+ PAG source term involves an iodine release fraction of approximately 1.6E-5. RC1.1E involves a much smaller iodine release fraction (2.3E-7), while RC1.1M (with only 30% of the frequency of RC1.1E) involves an iodine release fraction only slightly greater than that of the PAG calculation (2.08E-5). Therefore, the PAG calculation source term for System 80+ generally envelopes the potential source terms from PRA intact containment sequences.

C.3 SUPPORTING PRA EVALUATION

The requirement for a supporting PRA evaluation for emergency planning appears in Chapter 1, Section 2.6.3 of the URD and is repeated in Appendix A. A summary of the supporting PRA requirement is provided here for completeness.

The requirement is to: (1) demonstrate that the core damage frequency is less than 10^{-5} per year; (2) demonstrate that cumulative frequency for sequences resulting in a dose at 0.5 mile greater than 1 rem for 24 hours is less than 10^{-6} per year, and (3) demonstrate that the prompt accident qualitative health objective of the NRC Safety Goal Policy is met with no credit for offsite evacuation prior to 24 hours.

The supporting PRA evaluations are to be performed in accordance with Volume II, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules (KAG) with the exception that the off-site dose exceedance limit is 1 rem. The KAG specifies that the PRA address internal events plus external events with seismic risk to be addressed by the seismic margin approach per Chapter 1, Section 2.5.3.4, of Volume II of the URD. In addition, the KAG specifies quantitative assessment of uncertainties, including propagation of distributions for Level 1 analysis, consideration of phenomenological uncertainties in Level 2 and 3 analysis, and sensitivity studies to assess the effect of particularly important uncertainties.

ABB-CE has performed a PRA for the System 80+ in accordance with the KAG. The total mean frequency of core damage was estimated to be 1.7×10^{-6} per year for internal events at power. For external events the core damage frequency for fires and internal floods was estimated to be about 3.3×10^{-7} per year. Other external events are site specific, but on the basis of design characteristics and features provided to address such events the contribution of these events to core damage frequency is expected to be negligible. For shutdown conditions the core damage frequency was estimated to be 8.4×10^{-7} per year. Thus, the total core damage frequency is 2.9×10^{-6} and has over a factor of 3 margin to the 10^{-5} per year URD goal.

The System 80+ complementary cumulative distribution function (CCDF) for offsite dose for 24 hours has been developed in the PRA. The cumulative frequency for sequences resulting in greater than 1 rem at 0.5 mile is approximately 2.8×10^{-7} per year, again providing over a factor of 3 margin to the URD 10-6, 1 rem goal.

Based on the PRA performed by ABB-CE for the System 80+ design, the System 80+ design meets the supplementary PRA requirements with margin.

C.4 CONCLUSIONS REGARDING PASSIVE PLANT CONFORMANCE TO ALWR REQUIREMENTS

Based on this assessment, the System 80+ design meets the emergency planning design criteria. It is recognized that plant specific designs, will continue to evolve during remaining activities required to complete the design and construct the plant. These design evolutions are not expected to impact the conclusions of this assessment. ABB-CE is responsible to demonstrate that their certified designs continue to meet the emergency planning design criteria through the remaining design activities.

C.5 REFERENCES

C-1. ABB-Combustion Engineering, "Combustion Engineering Standard Safety Analysis Report - Design Certification", Amendment W, June 1994

C-2. Electric Power Research Institute, "ALWR Utility Requirements Document", NP-6780-L, Revision 7, May 1996.

C-3. ABB-CE letter to EPRI, October 29, 1998

C-4. L. Soffer, et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG 1465, Final Report, February, 1995

C-5. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C., May 1992.

C-6. L. Soffer, et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG 1465, Draft Report, June, 1992

D
ASSESSMENT OF ABWR DESIGN CONFORMANCE WITH ALWR EMERGENCY PLANNING DESIGN CRITERIA

The General Electric ABWR has been certified by the NRC under 10 CFR 52. The following assessment of this standard evolutionary advanced light water reactor (ALWR) plant design is provided to describe how the ABWR design meets the ALWR emergency planning design criteria. The assessment is based on information contained in the ABWR Standard Safety Analysis Report (SSAR) [D-1] and other documentation referenced by the SSAR. As described in Revision 8 of the Utility Requirements Document (URD) [D-2], Volume II, Chapter 1, Section 2.6, the ALWR emergency planning design criteria include a containment performance criterion and a site boundary dose criterion. It is necessary to assess conformance of the design to these criteria. In addition, the URD specifies that a supplementary PRA evaluation be performed to support the assessment.

D.1 CONTAINMENT PERFORMANCE CRITERION

The containment performance criterion for emergency planning appears in URD Chapter 1, Section 2.6.1 and is repeated in Appendix A. The criterion can be summarized as follows:

> Plant design characteristics and features shall be provided to preclude core damage sequences which could bypass containment and to withstand containment loads from core damage sequences. Containment loads from core damage sequences should be evaluated and should not exceed the combination of peak loads from a DBA LOCA and a hydrogen burn from oxidation of 75% of the active fuel cladding, or not exceed ASME Source Level C limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours there shall be no uncontrolled release.

The methodology which is specified in the URD for demonstrating the containment performance criterion includes the following:

- Incorporate the design characteristics and features specified in the URD to address severe accident challenges.
- Evaluate containment responses for a reference accident sequence. This sequence shall be a low pressure core melt into an intact containment with containment systems functioning as designed to confirm that containment loads do not exceed peak LOCA-plus-hydrogen loads or ASME Service Level C limits.

- Evaluate containment response for functional sequence types with mean frequency > 10^{-7} per reactor year to confirm that containment leak tightness and structural integrity are maintained, and that containment loads do not exceed peak LOCA-plus-hydrogen loads for Service Level C limits. Functional sequence types with mean frequency < 10^{-7} per reactor year should be reported for discussion, including a description of the plant features credited to reach this low frequency.
- Provide protection of the containment for overpressurization beyond 24 hours. Such protection may be provided by the pressure suppression features of the plant, as well as by the size and strength of the containment structure itself.

The steps used for the assessment of ABWR compliance with the containment performance criterion was as follows:

1. Confirm that the design meets the requirements of the URD, Volume II, Chapter 5, Section 6.6.2.1 by comparing the ABWR design characteristics and features to the requirements.

2. Confirm that the containment loads from core damage sequences do not exceed limits as specified in the URD, Volume II, Chapter 5, Section 6.6.2.2 for approximately 24 hours, given realistic severe accident assumptions.

3. Confirm as specified in the URD, Volume II, Chapter 5, Section 6.6.2.3 that containment leaktightness and structural integrity are maintained in response to ABWR functional severe accident sequences with frequencies > 10^{-7} per reactor year and that containment loads do not exceed limits specified in the URD, Volume II, Chapter 5, Section 6.6.2.2. In addition, confirm that functional severe accident sequences with frequencies < 10^{-7} per reactor year are reported with a description of plant features credited to reach the low frequency.

4. Confirm that the design meets the URD, Volume II, Chapter 5, Section 6.6.2.5 and that no uncontrolled release will occur beyond approximately 24 hours.

D.1.1 Plant Design Characteristics and Features to Address Containment Challenges

The assessment was performed by reviewing the ABWR SSAR to confirm, for each containment challenge, the existence of specific design features or characteristics to fulfill the key URD requirements associated with the challenge. The list of challenges and associated requirements as summarized in Appendix A, Tables A.5-1 and A.5-2 was used for this review. A requirement was considered met when an explicit reference to the system, feature, or characteristic was made in the SSAR.

Table D-1 summarizes the results of the assessment for ABWR. This table lists the challenges and associated requirements, and identifies in brackets the sections of the ABWR SAR which address each requirement. Specific ABWR design features or capabilities were identified in response to all of the requirements. Additional information was requested from General Electric for the following items:

- Welds in the beltline region of the RPV
- Limit potential for or accomodate internal containment loading due to in-vessel debris-water interaction

References [D-3] and [D-4] provide this additional information.

On the basis of the assessment and references [D-3] and [D-4], the ABWR design meets the requirements of Volume II, Chapter 5, Section 6.6.2.1 of the URD.

D.1.2 Comparison of Loads From A Reference Severe Accident With Peak LOCA Plus Hydrogen Partial Pressure Loads

For plant designs which meet all of the URD provisions related to containment challenges, the severe accident sequence for which containment loads should be compared with the peak LOCA plus hydrogen partial pressure loads[1] are low pressure core melts into an intact containment with the reactor coolant system (RCS) at low pressure and containment systems functioning as designed.

A comparison of ABWR containment loads from a reference severe accident with peak LOCA plus hydrogen burn loads has been performed by evaluating a low pressure core melt sequence from the ABWR SSAR. The sequence selected is the base case for Accident Class ID, and is reported in Chapter 19D of the ABWR SSAR. As shown in Table D-2, Accident Class ID is the largest contributor (about 45%) to the ABWR core damage frequency. Table D-2 shows the ten accident classes defined for the ABWR SSAR.

The accident is initiated by a station blackout, followed by MSIV closure, reactor scram, and loss of feed water. The reactor coolant system is initially at high pressure, but after 30 minutes the operator opens one SRV in order to depressurize the vessel. The vessel blows down, but the emergency core cooling (ECC) injection systems are assumed to fail. As the vessel blows down the water level falls, the core uncovers, and core damage results shortly thereafter. About 220 kg (485 lbm) of hydrogen are produced from zircaloy oxidation in-vessel. The vessel is calculated to fail at 1.8 hours after accident initiation. Molten core debris, and any water remaining in the vessel, then falls into the lower drywell. Core debris-water interactions in the lower drywell produce considerable steam, causing a pressure increase to about 0.36 MPa (52 psia).

[1] Hydrogen burn loads need not be considered for the ABWR because the containment is inerted.

Table D-1
ASSESSMENT OF ABWR DESIGN CONFORMANCE WITH ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
1. Containment Isolation	Isolation	• Isolation provisions and leakage rate testing per standards [6.2.6.3]. • Valves capable of closure with maximum flow and full containment pressure [6.2.4.3.1]. • Control room position indication for automatic and remote manual valves [6.2.4.4]. • Manual valve configuration permits locking only in fully closed position [6.2.4.2]. • Closed systems penetrating containment evaluated for ex-vessel severe accidents [19.9.17]. • Fail closed or DC powered isolation valves [6.2.4.2]. • Capability for periodic gross check of containment integrity [6.2.6].	
2. Interfacing System LOCA	Bypass	• Reduced number of interfaces between the Reactor Coolant System (RCS) and low pressure systems [5.2.5]. • High to low pressure interfaces provided with isolation valve leak testing capability [5.2.5], isolation valve position indicator in control room [App. 3M], and high pressure alarm [App. 3M]. • RSDC designed for full RCS pressure [19.8.1.3]. • Double valve isolation [6.2.4.2].	• Pressure Relief [5.2.2]. • Design pressure such that full RCS pressure is below rupture pressure and no leaks will occur which exceed RCS makeup capacity [19.8.1.3].
3. Blowdown Forces	Containment Pressure Control	• Design and ISI in accordance with ASME BPV Code [3.6.2.3.3 & 5.2.3]. • Leak Before Break [3.6.3 & App. 3E].	• Design containment for double-ended guillotine break of largest pipe [6.2.1.1.1 & 6.2.1.1.3.3].

Table D-1 (Cont'd)
ASSESSMENT OF ABWR DESIGN CONFORMANCE WITH ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
4. Pipe Whip and Jet Impingement	Bypass	• Design and ISI in accordance with ASME BPV Code [3.6.2.3.3 & 5.2.3]. • Leak Before Break [3.6.3 & App. 3E]. • Proven materials and fabrication processes [3.2]. • Use of EPRI water chemistry guidelines [5.2.3.2.2].	• Protection from jet/pipe whip where leak before break is not demonstrated [6.3.1.1.3 & 6.2.1.1.1].
5. Steam Generator Tube Rupture	Bypass	Not Applicable	
6. ATWS	Reactivity Control	• Diverse reactor protection system (RPS) [App. 7C.5]. • Diverse means of rod insertion [4.6.1.2].	• Standby liquid control SLC) [7.4.1.2]. • Negative moderator temperature coefficient over entire fuel cycle improves ATWS response [4.3.2.3.2].
7. Suppression Pool Bypass	Containment Pressure Control	• Vacuum breakers: potential loads accounted for [19.8.4.3]. • Position indication, minimal leakage [19.8.7].	• ADS use of SRVs which discharge to suppression pool and thus ensure vapor suppression despite leakage [5.2.2]
8. Catastrophic RPV Failure	Internal Containment Loading	• $RT_{NDT} \leq 10°F$; low fluence at vessel wall [5.3.2.1, 5.3.2.1.5] • Phosphorous, copper, vanadium and sulfur material limits in high fluency region [5.3.1.2, 5.3.3.1.1.1] • No welds in beltline region [see reference [D-3]]. • Relief valves prevent overpressure, backed up by depressurization system and low-head injection [5.2.2, 6.3.2.2.4]. • Design to ASME code [5.3.3.1.1.1]. • Design features to avoid relief valve opening for expected plant transients [7.7.1.8].	
9. Internal Vacuum	Containment Pressure Control		• Design for external pressure loads [6.2.1.1.4]. • Vacuum breakers [6.2.1.1.2.1]

Table D-1 (Cont'd)
ASSESSMENT OF ABWR DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
10. Internal (Plant) Missiles	External Containment Loading	• Turbine overspeed protection [10.2.2.4]. • Improved turbine integrity/one-piece rotors [10.2.3].	• Turbine orientation avoids missile contact with containment [3.5.1.1.1.3 & fig 3.5-2]. • Missile protection for any safety related components in missile path (SRP 3.5.1.3) [3.5.1, 3.5.3, & 3.5.4].
11. Tornado and Tornado Missiles	External Containment Loading	• Conformance with ANSI 2.12 and ANSI 51.5 (in accordance with ASCE 7-88, "Minimum Design Loads for Buildings and other Structures," formerly ANSI A58.1-82) [3.3].	
12. Man-Made Site Proximity Hazards	External Containment Loading	• Conformance with ANSI 2.12 [2.1 & 2.3.2].	
13. Seismic	External Containment Loading	• Sitting requirements exclude the most vulnerable sites [no effect on design].	• SSE at 0.3g [1.2.2.1.2.3]. • Margins assessment for > SSE as part of design process [19.4.3]. • Address vulnerabilities from past experience, e.g., provide common basemat [3.8.1.1.1, 19.4.3, & 19.8.2].
14. High Pressure Melt Ejection (HPME)	Reactor Pressure Control	• Depressurization systems [6.3.2.2.2, 5.2.2 & 7.3.1.1.1.2].	• Cavity configuration to limit transport of fragmented core debris [19.8.7].
15a. Hydrogen Generation to Detonable Limits	Combustible Gas Control	• Inerted [6.2.1.1.10.2, 6.2.5 & 19.8.7]	
15b. Hydrogen Deflagration	Combustible Gas Control	• Inerted [6.2.1.1.10.2, 6.2.5 & 19.8.7]	
16. In-Vessel Debris-Water Interaction	Internal Containment Loading	• Large-scale phenomena limited in probability [see Ref [D-4]]. • In-vessel geometry limits interacting quantities and size of any interaction [see Ref. [D-4]].	• Rugged reactor vessel contains forces [see Ref. [D-4]]; as backup, rugged reactor cavity contains lower head failure [see Ref. [D-4]].

Table D-1 (Cont'd)
ASSESSMENT OF ABWR DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS	
		LIMIT POTENTIAL FOR CHALLENGE	ACCOMMODATE CHALLENGE
17. Ex-Vessel Debris-Water Interaction	External Containment Loading	• Large-scale phenomena limited in probability [19E.2.3.1, 19E.2.6.7, &19EB]. • Ex-vessel geometry limits interacting quantities and size of any interaction [19E.2.3.1, 19E.2.6.7, &19EB].	• Rugged reactor cavity confirmed by evaluation [19E.2.3.1, 19E.2.6.7, &19EB]. • Containment design accommodates steam generation [19E.2.3.1, 19E.2.6.7, &19EB].
18. Noncondensible Gas Generation	Fuel/Debris Cooling	• Features limiting concrete erosion [6.2.1.1.10.3] limit noncondensible gas generation as well. • Overlying pool cools gases from core-concrete interaction [19E.2.1.5.2.8, 19E.2.6.8, &19EC].	• Containment size and pressure retention capability [19E.2.1.5.2.8, 19E.2.6.8, &19EC].
19. Basemat Erosion and Vessel Support Degradation	Fuel/Debris Cooling	• Lower drywell spreading area of $0.02m^2$/MWt promotes core debris cooling [19.8.7]. • Lower drywell flooding [19.8.7]. • Backup capability for water addition from sources outside containment [19.8.1 & 19.8.7].	• Sacrificial concrete where debris on floor contacts boundary structures [19.8.7].
20. Core Debris in Sump	Fuel/Debris Cooling	• Special lower drywell sump design prevents localized unterminated core-concrete interaction [19.8.7]. • Sump drainline configuration precludes gravity transport of debris ex-containment [19.8.7]. • Lower drywell flooding [19.8.7].	
21. Core Debris Contact with Liner	Fuel/Debris Cooling	• Liner protected by concrete [19E.2.5.2]. • Lower drywell flooding [19.8.7]. • Design features to limit debris dispersal including ADS [19.8.7].	
22. Decay Heat Generation	Containment Pressure Control	• Main condenser [10.4.1] • Reactor water cleanup system [5.4.8]	
23. Tube Rupture from Hot Gases	Bypass	Not Applicable	

Table D-2
SUMMARY OF ABWR SEVERE ACCIDENT SEQUENCES FOR LEVEL 2 PRA ANALYSIS BASE CASE SEQUENCES

ACCIDENT CLASS	LEVEL 1 DEFINITION	LEVEL 2 BASE CASE DEFINITION	CORE DAMAGE FREQUENCY per year	CONTAINMENT MAXIMUM PRESSURE AND WALL TEMPERATURE	CONTAINMENT FAILURE	FEATURES & CHARACTERISTICS CREDITED
IA	Transients followed by failure of high pressure cooling and failure to depressurize the reactor	Loss of all core cooling with vessel failure at high pressure (LCHP)	4.2×10^{-8}	7.2 bar (104 psia) 533K (500^{o} F)	rupture disk opens	passive flooder, drywell spray, rupture disk
IB-1	Station blackout events (short term) with RCIC failure	Loss of all core cooling with vessel failure at high pressure (LCHP)	2.6×10^{-8}	7.2 bar (104 psia) 533K (500^{o} F)	rupture disk opens	passive flooder, drywell spray, rupture disk
IB-2	Station blackout events with RCIC available for about eight hours	Station blackout with RCIC available (SBRC)	1.6×10^{-8}	7.2 bar (104 psia) 533K (500^{o} F)	rupture disk opens	passive flooder, rupture disk
IB-3	Station blackout events (long term) with RCIC failure	Loss of all core cooling with vessel failure at high pressure (LCHP)	8.9×10^{-10}	7.2 bar (104 psia) 533K (500^{o} F)	rupture disk opens	passive flooder, drywell spray, rupture disk
IC	ATWS events without boron injection coupled with loss of core cooling	Concurrent loss of all core cooling and ATWS with vessel failure at low pressure (NSCL)	2.8×10^{-13}	7.2 bar (104 psia) Drywell temp. not reported.	rupture disk opens	ADS, passive flooder, rupture disk
ID	Transients with loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling	Loss of all core cooling with vessel failure at low pressure (LCLP)	7.0×10^{-8}	7.2 bar (104 psia) 533K (500^{o} F)	rupture disk opens	ADS, passive flooder, rupture disk

Table D-2 (Cont'd)
SUMMARY OF ABWR SEVERE ACCIDENT SEQUENCES FOR LEVEL 2 PRA ANALYSIS
BASE CASE SEQUENCES

ACCIDENT CLASS	LEVEL 1 DEFINITION	LEVEL 2 BASE CASE DEFINITION	CORE DAMAGE FREQUENCY per year	CONTAINMENT MAXIMUM PRESSURE AND WALL TEMPERATURE	CONTAINMENT FAILURE	FEATURES & CHARACTERISTICS CREDITED
II	Transient, LOCA, and ATWS (with boron injection), successful core cooling, possible containment failure	Loss of containment heat removal (LHRC)	1.1×10^{-10}	7.2 bar (104 psia) Drywell temp. not reported.	rupture disk opens	RCIC, ADS, HPCF, rupture disk
IIIA	Small or medium LOCAs with failure of high pressure core cooling followed by failure to depressurize the reactor	Loss of all core cooling with vessel failure at high pressure (LCHP)	3.9×10^{-10}	7.2 bar (104 psia) 533K (500° F)	rupture disk opens	passive flooder, drywell spray, rupture disk
IIID	LOCAs followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling	Large LOCA with failure of all core cooling (LBLC) (represented by a main steam line break)	3.0×10^{-10}	7.2 bar (104 psia) 533K (500° F)	rupture disk opens	passive flooder, rupture disk
IV	ATWS events without boron injection but with core cooling available	Concurrent station blackout with ATWS (NSRC)	2.7×10^{-10}	7.2 bar (104 psia) Drywell temp. not reported.	rupture disk opens	RCIC, ADS, rupture disk, passive flooder

ADS - automatic depressurization system
FS - fire water spray system into drywell
HPCF - high pressure core flooder
LOCA - loss of coolant accident
RCIC - reactor core isolation cooling system
RCS - reactor coolant system
RHR - residual heat removal system

For comparison, the Service Level C allowable drywell head pressure load, corresponding to peak LOCA plus hydrogen produced from reacting 100% of the zirconium in the active fuel region, has been determined in the SSAR to be 111.7 psia (7.6 bar) at 500 F. The median ultimate strength of the containment is 149.9 psia (10.2 bar). Thus the containment design provides substantial margin to loads which would be expected should a severe accident occur.

Although the Level 1 analysis gives a very high likelihood of power recovery between thirty minutes and two hours after accident initiation, the MAAP analysis reported in the SSAR conservatively assumes no power recovery and no containment heat removal. Two design features function to keep the containment intact. A passive flooder system allows water to cover and quench debris in the lower drywell, thus limiting ablation of the concrete basemat; and a rupture disk would open in the wetwell air space, thus preventing containment building failure.

For the sequence considered here, water remaining in the reactor pressure vessel follows the core debris into the lower drywell and temporarily cools it down, such that the energy is carried away as steam. This water boils away in about one hour. The drywell pressure then temporarily decreases as steam is condensed on the containment heat sinks. Then, the core debris temperature increases and the containment begins to heat up until it reaches 533K (500^{o} F), when the passive flooder opens. Water then pours from the wetwell into the lower drywell, covers the core debris, and quenches it. As long as the cooler water from the wetwell flows into the lower drywell, the partial pressure from steam remains low. Soon, however, the water level in the lower drywell increases to the point where the flow stops. The water in the lower drywell reheats, reaches saturation, and steam is produced from the core debris-water interactions.

The containment pressure increases due to the resultant steam production until it reaches 0.72 MPa (104 psia) in the wetwell at about 20 hours. At this time the rupture disk in the wetwell air space opens to relieve the containment pressure. During the blowdown any airborne fission products flow into the suppression pool, which effectively removes all species except the noble gases. Essentially all of the noble gas inventory exits the pool, flows through the opened rupture disk, and up the stack to the environment. The containment peak pressure and temperature are 104 psia (7.2 bar) and 500°F (533 K) respectively.

Based on the results in Table D-2, sequences involving low pressure core melt into an intact containment with containment systems functioning as designed are bounded by the peak LOCA plus hydrogen partial pressure loads with significant margin. In addition it is shown in Table D-2 that all base case sequences are bounded by the peak LOCA plus hydrogen partial pressure loads.

From this assessment, it is concluded that the ABWR design meets the criteria regarding a comparison of loads from a reference core damage sequence to the loads from a peak LOCA combined with hydrogen burn. The containment loads representing those from core damage sequences do not exceed the loads from a peak LOCA combined with hydrogen burn as described in the URD, Volume II, Chapter 5, Section 6.6.2.2 for approximately 24 hours under realistic severe accident assumptions.

D.1.3 Evaluation of Containment Response for Functional Sequence Types

The URD requires in Volume II, Chapter 5, Section 6.6.2.3 that containment leak-tightness and structural integrity be analyzed in response to functional severe accident sequences with frequency > 10^{-7} per year, and that containment loads do not exceed peak LOCA plus hydrogen loads described in the URD, Volume II, Chapter 5, Section 6.6.2.2. In addition, functional severe accident sequences with frequency < 10^{-7} per year shall be reported with a description of plant features credited to reach this low frequency. Finally, the frequency of those sequences types which may result in early containment failure are to be identified.

There are ten functional sequence types or accident classes described in Chapter 19 of the SSAR and these are shown in Table D-2. They encompass the numerous accident sequences found in the Level 1 PRA. The specific scenario chosen for Level 2 analysis for an accident class (Level 2 base case) is based on the most probable sequence in the accident class. The Level 2 analysis is used to determine representative sequences for each release category to estimate fission product releases that are input for the off-site dose analysis.

As shown in Table D-2, there are no accident classes with core damage frequency estimated to be greater than 10^{-7} per year, although four of the ten have frequencies above 10^{-8} per year. Table D-2 shows that each of the Level 2 base case analyses have significant margin between the estimated resulting containment loads and the peak LOCA combined with hydrogen partial pressure loads as described in the URD, Volume II, Chapter 5, Section 6.6.2.2. In addition, Table D-2 provides a list of features credited in the Level 1 PRA analysis which support the very low frequency.

Based on the results described above the ABWR design addresses severe accident containment challenges as required in Volume II, Chapter 1, Section 2.6.1 of the URD with significant margin. Functional sequence frequencies are all below 10^{-7} per year and for all sequences the loads generated are below those from the peak LOCA plus hydrogen partial pressure loads.

D.1.4 Assessment of Uncontrolled Release

In order to further mitigate the consequences of severe accidents which could lead to an uncontrolled release of fission products and to limit the effects of uncertainties in severe accident phenomena, the ABWR incorporates a Containment Overpressure Protection System (COPS). The COPS provides a fission product release path through the suppression pool to the environment at a setpoint pressure sufficient to prevent the containment structure, specifically the drywell head, from failing. Thus the COPS will greatly reduce any fission product release to the environment that would otherwise occur should a hypothetical severe accident generate sufficient pressure to potentially fail the ABWR containment.

To illustrate the effectiveness of the COPS, the ABWR SSAR describes a set of sensitivity studies in which the rupture disk of the COPS opens and drywell head failure is prevented. Fission product releases using the COPS are low, about 1E-7 for CsI, after 72 hours. In addition, the elapsed time to the COPS rupture disk opening is greater than 24 hours for most severe accidents.

According to the SSAR, there is a small (~2%) likelihood of drywell head failure given that the rupture disk setpoint pressure of the COPS is reached during a severe accident. For this case, the airborne fission products would be released to the environment from the drywell without having been scrubbed in the suppression pool. The release would be uncontrolled, but slow. Typically the release would begin many hours after accident initiation with an iodine release of less than 0.5% at 24 hours.

In summary, for highly unlikely sequences that could lead to uncontrolled releases the ABWR incorporates an overprotection system that is designed to prevent containment failure and scrub any fission product releases.

D.2 Dose Criterion

The dose criterion for emergency planning appears in Chapter 1, Section 2.6.2 of the URD and is repeated in Appendix A. A discussion of the dose criterion appears in the Main Report. A summary of the dose criterion and associated methodology is provided here for completeness.

The criterion is that the dose at 0.5 mile from the reactor due to fission product source term release from a degraded core shall not exceed the Protective Action Guides (PAGs) for approximately 24 hours.

The methodology for demonstrating the dose criterion includes the use of a physically-based source term similar to that defined in NUREG 1465 [D-5], a probabilistic dose method (i.e., MACCS 1.5), use of a range of meteorological conditions, and use of effective dose equivalent with a 50 year commitment. All NUREG 1465 release phases (gap, early in-vessel, ex-vessel, and late in-vessel) are to be used for this dose calculation. The PAGs are projected dose levels for evacuation (1 to 5 rem effective dose) which are specified by the Environmental Protection Agency in a 1992 report [D-6] as guidance for actions to protect the public in the early phase of a nuclear incident.

Since General Electric did not make use of the NUREG-1465 source term in the radiological DBA for ABWR design certification (unlike System 80+ and AP600), it has been necessary to estimate the PAG comparison source term from other sources.

Two possible sources were considered. One was a comparison to the source term calculated for the Browns Ferry pilot plant application (i.e., for operating plants, the first application of the NUREG-1465 source term to a BWR), and the second was the NCF ("no containment failure") source term calculated for the ABWR PRA. Both of these approaches have limitations. The Browns Ferry application, since it represents a design basis accident (DBA), does not include the ex-vessel and late in-vessel contributions (i.e., vessel failure is not assumed). Therefore, use of the Browns Ferry source term (even with adjustments made to compensate for important plant differences) would yield a source term less than what the PAG-comparison source term should be. The ABWR PRA NCF source term, on the other hand, represents a three-day release and would be larger than appropriate. It also assumes that the containment leakage occurs from either the drywell or the wetwell, and then uses the worst release for each group (the noble gas release being from the wetwell and the particulate release being from the drywell). However, by reviewing both sources, a reasonable estimate between the two has been devised.

D.2.1 Environmental Source Term for Browns Ferry DBA and Application to ABWR PAG Analysis

The environmental source term for Browns Ferry can be obtained from reference [D-7]. Since reference [D-7] presents calculated doses and concentrations in control volumes of the plant complex, the source term must be "backed out".

The source term will have six release fractions: a noble gas release fraction, a particulate iodine release fraction, an organic iodine release fraction, an elemental iodine release fraction, a cesium release fraction, and an "other" release fraction. The "other" release fraction is based on a one percent release of "other" nuclides to the containment and can, therefore, be adjusted proportionally for containment release fractions which differ from one percent.

In reference [D-7], whole body doses are calculated for each radionuclide at the EAB and at the LPZ at 2 hours and at 8 hours. There is no 24-hour dose information presented. By using the X/Q values for the various release points, the following source terms can be "backed-out" of the Browns Ferry analysis for the release up to 8 hours (in fraction of core inventory):

Noble gas	2E-3
Inorganic I	8E-6
Organic I	9E-7
Cesium	1E-5
Other	2E-7

For all but the noble gas and organic iodine, the bulk of the release is expected to be completed by 8 hours. The noble gas and organic iodine release would continue at essentially the same rate for the full 24 hours. Since the containment leak rate for Browns Ferry is a factor of two greater than that for ABWR, the noble gas and organic iodine release will be increased by a factor of 3/2 for application to ABWR, and the release of the other groups will be reduced by a factor of two. The ABWR releases are then estimated to be:

Noble gas	3E-3
Inorganic I	4E-6
Organic I	1E-6
Cesium	5E-6
Other	1E-7

Breaking the "Other" down by the NUREG 1465 containment release fractions (for the groups not explicitly given above), the ABWR source terms become as follows:

Noble gas	3E-3
Inorganic I	4E-6
Organic I	1E-6
Cesium	5E-6
Tellurium	5E-7
Ba-Sr	2E-7
Noble met	3E-10
Cerium	5E-11
Lanthanum	2E-11

These are estimates for a revised source term application to ABWR absent ex-vessel release.

D.2.2 Application of the NCF Source Term to ABWR PAG Analysis

The NCF source term from the ABWR PRA exhibits the following release fractions:

Noble gas	4.4E-2
Inorganic I	2.3E-5
Cesium	2.3E-5

These source terms are for three days, and are maximized in the sense that the noble gas source term assumes all containment leakage from the wetwell and the other two source terms assume all containment leakage from the drywell. If the leakage were equally divided between drywell and wetwell the source term would be approximately as follows:

Noble gas	2E-2
Inorganic I	1E-5
Cesium	1E-5

Recognizing that most of the three day release for the iodine and cesium would occur within the first day while the noble gas release would be approximately continuous, one can estimate that the release at the end of 24 hours would be as follows:

Noble gas	7E-3
Inorganic I	1E-5
Cesium	1E-5

These would be the estimated release at 24 hours for the NCF source term with containment leakage distributed between the drywell and the wetwell. This source term includes vessel failure and the effects of ex-vessel release.

D.2.3 Reconciliation of Source Terms for ABWR PAG Analysis

The two source terms "sets" based on application of the revised DBA source term to Browns Ferry and then to ABWR (with no ex-vessel contribution) and the truncation of the ABWR NCF source term to a 24 hour release (with an ex-vessel contribution) are as follows:

No Ex-Vessel		With Ex-Vessel	
Noble gas	3E-3	Noble gas	7E-3
Inorganic I	4E-6	Inorganic I	1E-5
Organic I	1E-6		
Cesium	5E-6	Cesium	1E-5
Tellurium	5E-7		
Ba-Sr	2E-7		
Noble met	3E-10		
Cerium	5E-11		
Lanthanum	2E-11		

These sets are reasonably similar. For conservatism the second set will be used for the three nuclide groups it describes, and to account for the ex-vessel contribution for the other nuclide groups, the organic iodine will be increased by a factor of two (for completeness only, since it is not included in the PAG dose calculation) and the other groups will be increased by the ratio of ex-vessel release from NUREG 1465 to the DBA source term. This yields the following source term set which is used for the ABWR PAG analysis:

Noble gas	7E-3
Inorganic I	1E-5
Organic I	2E-6
Cesium	1E-5
Tellurium	3E-6
Ba-Sr	1E-6
Noble met	6E-10
Cerium	6E-10
Lanthanum	5E-10

The emergency planning application utilizing the full NUREG 1465 release is intended to be at the limiting end of the spectrum for PRA source terms from core melt accidents with intact containment. Thus this emergency planning source term should generally envelope potential source terms from PRA intact containment sequences. To confirm this, comparisons of the emergency planning source term have been performed with the source terms from the ABWR PRA for release categories with intact containment or with frequency greater than $\sim 10^{-7}$ per year.

D.2.4 ABWR NUREG 1465 Source Term Dose Evaluation Results

Using the releases from the previous section, the doses have been calculated using MACCS 1.5. The results for the median efective dose and 0^{th} percentile dose are 0.88 rem and 4.18 rem, respectively. This is below the 1 rem and 5 rem limits.

D.3 SUPPORTING PRA REQUIREMENT

The requirement for a supporting PRA evaluation for emergency planning appears in Chapter 1, Section 2.6.3 of the URD and is repeated in Appendix A. A summary of the requirement is provided here for completeness.

The requirement is to: (1) demonstrate that the core damage frequency is less than 10^{-5} per year; (2) demonstrate that cumulative frequency for sequences resulting in a dose at 0.5 mile greater than 1 rem for 24 hours is less than 10^{-6} per year, and (3) demonstrate that the prompt accident qualitative health objective of the NRC Safety Goal Policy is met with no credit for off-site evacuation prior to 24 hours.

The supporting PRA requirements are to be performed in accordance with URD Volume II, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules (KAG) with the exception that the off-site dose exceedance limit is 1 rem. The KAG specifies that the PRA address internal events plus external events with seismic risk to be addressed by the seismic margin approach per Chapter 1, Section 2.5.3.4 of Volume II of the URD. In addition, the KAG specifies quantitative assessment of uncertainties, including propagation of distributions for Level 1 analysis, consideration of phenomenological uncertainties in Level 2 and 3 analysis, and sensitivity studies to assess the effect of particularly important uncertainties.

General Electric has performed a PRA for the ABWR in accordance with the KAG. The total mean frequency of core damage was estimated to be 1.6×10^{-7} per year for internal events at power. For external events the core damage frequency for fires and internal floods was estimated to be 1.3×10^{-7} per year. Other external events are site specific, but on the basis of design characteristics and features provided to address such events the contribution of these events to core damage frequency is expected to be negligible. For shutdown conditions the core damage frequency was estimated to be $< 10^{-7}$ per year. Thus, the total core damage frequency is expected to have significant margin to the 10^{-5} per year URD goal.

The ABWR complementary cumulative distribution function (CCDF) for offsite dose for 24 hours has been developed in the PRA. The cumulative frequency for sequences resulting in greater than 1 rem is approximately 4×10^{-8} per year, thus providing significant margin to the URD 10^{-6}, 1 rem goal.

Based on the PRA performed by General Electric for the ABWR design, the ABWR design meets the supplementary PRA requirements with considerable margin.

D.4 CONCLUSIONS REGARDING PASSIVE PLANT CONFORMANCE TO ALWR REQUIREMENTS

Based on this assessment, the ABWR design meets the emergency planning design criteria and supporting PRA evaluation. It is recognized that plant specific designs will continue to evolve during remaining activities required to complete the design and construct the plant. These design evolutions are not expected to impact the conclusions of this assessment. General Electric is responsible to demonstrate that their certified designs continue to meet the emergency planning design criteria through the remaining design activities.

D.5 REFERENCES

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D-2. Electric Power Research Institute, "ALWR Utility Requirements Document, " NP-6780-L, Revision 8.

D-3. Fax, S. Stark (General Electric) to E. Rodwell (EPRI), June 4, 1999.

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D-6. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C., May 1992.

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E
TECHNICAL BASIS FOR INGESTION EXPOSURE PATHWAY EMERGENCY PLANNING DISTANCE

E.1 Introduction

The purpose of this appendix is to provide the technical basis for the ingestion pathway emergency planning distance for ALWR designs. In general, radiation exposure pathways from accidents include exposure to the plume (both directly and through inhalation), direct exposure to ground contamination, and ingestion of contaminated foodstuffs and water. This last pathway is widely referred to as the "ingestion pathway". In 10CFR50 it receives particular attention; in fact, the current emergency planning zone (EPZ) described in 10CFR50.47(c)(2) for the ingestion pathway is five times larger in radius (i.e., 50 miles vs. 10 miles) and 25 times larger in area than the current emergency planning zone for plume exposure.

This appendix does the following:

- It examines the published justification (including bases and criteria) for the current size of the ingestion pathway emergency planning zone given in NUREG-0396 [E-1],
- It identifies the input to the analyses supporting the size determination of the current ingestion pathway emergency planning zone and the manner in which the results of those analyses were compared to the criteria to establish the required zone size,
- It develops revised input for determining the required size of the ingestion pathway emergency planning zone for ALWRs based on ALWR design information in areas affecting that input, and
- It recommends changes to the size of the ingestion pathway emergency planning zone for ALWRs based on revised analyses (using the revised inputs) but using the existing criteria.

E.2 Bases/Criteria for the Current Size of the Ingestion Pathway Emergency Planning Zone

The current size requirements for the ingestion pathway EPZ are stated in 10CFR50(c)(2) and in 10CFR50 Appendix E, Section I, Footnote 1. Footnote 1 of Appendix E, Section I refers to NUREG-0396 (Reference E-1) which, as noted above, is the basic document supporting the EPZ size requirements for US power reactors. The EPZ size requirements are summarized in Section 3.B of NUREG-0396 with a more detailed discussion presented in Appendix I.

In NUREG-0396 Section 3.B and Appendix I the basis for the ingestion pathway EPZ size is tied to the projected dose to an infant's thyroid from the ingestion of cow's milk following fission product release from a core melt accident. At the time of the writing of NUREG-0396 the then current guideline was that dairy cows should be placed on stored feed when the projected dose to an infant's thyroid exceeded 10 rem. However, the authors of NUREG-0396 were aware that this Protective Action Guideline (PAG) was under review, and it was anticipated that this PAG was going to be reduced to 1.5 rem. In anticipation of that reduction Table 1 of NUREG-0396 gives a recommended ingestion EPZ radius of about 50 miles, although it acknowledges that 25 miles would be appropriate if the infant thyroid dose PAG were to remain at the 10 rem level.

The details supporting the recommended 50-mile radius for the ingestion EPZ are presented in Appendix I of NUREG-0396. Figure I-14 in NUREG-0396, reproduced as Figure E-1 in this appendix, shows the probability of exceeding three values of infant thyroid dose as a function of distance from the point of release given a core melt accident. All three of these curves are based on maximum values of ground concentration versus distance; i.e., that concentration which would exist under the centerline of the plume. As such there is considerable "hidden" conservatism in these plots in that a given quantity of milk ingested by a given infant would hardly be expected to come from a single cow or group of cows feeding on grass with only the maximum ground concentration. Nevertheless, this was the assumption made at the time the original EPZ bases were developed.

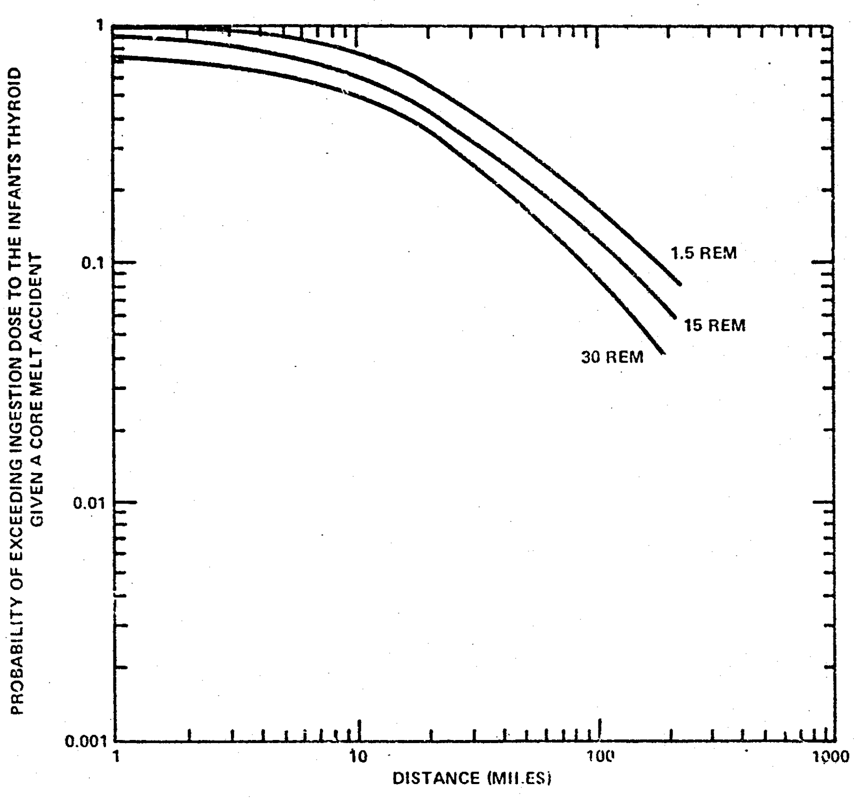

Figure E-1
Reproduction of Figure I-14 of NUREG-0396 [E-1]-Conditional Probability of Exceeding Thyroid Dose to an Infant Versus Distance. Probabilities are Conditional on a Core Melt Accident (5×10^{-5}). Thyroid dose calculated is due solely to radionuclide ingestion through the milk consumption pathway. Dose calculations assumed no protective actions taken, and straight line trajectory.

The range of core melt accident radiological releases to the environment (i.e., the environmental source term) considered in the preparation of NUREG-0396, Figure I-14 (see Figure E-1) are those from the 1975 Reactor Safety Study (RSS), WASH 1400 [E-2]. From this figure it is observed (and it is discussed in NUREG-0396, Appendix I) that at a distance of 25 miles the conditional probability falls below about 0.4 that the dose to an infant's thyroid from cow's milk ingestion will exceed the 10 rem PAG given a core melt accident. This value of about 0.4 is observed to be comparable to the exceedance probability of about 0.3 for the 1 rem whole-body dose PAG at 10 miles for plume exposure (see Appendix I, Figure I-11 of NUREG-0396 and the discussion in the Main Report relative to the bases for the size of the plume exposure EPZ). Appendix I of NUREG-0396 states further that should the infant thyroid dose PAG be reduced to 1.5 rem (as was then anticipated), the 25-mile distance would have to be increased to 50 miles to achieve the same effect (i.e., an exceedance probability for the infant thyroid dose PAG comparable to that for the one rem plume exposure PAG).

Since the time of the publication of NUREG-0396 a few months before the accident at Three Mile Island, a considerable body of evidence has been accumulated (beginning with the Three Mile Island accident, itself) that the source terms of the RSS were considerably over-estimated. For the ALWRs there exists the additional effect that the plants were designed with minimization of plant risk in mind (as assessed using Probabilistic Risk Assessment or PRA); therefore, one would expect that the projected offsite doses using modern source term methods for modern plants would yield lower dose versus distance projections than those of NUREG-0396. This is, in fact, the case.

E.3 Assessment of Modeling Assumptions and Input for Analyses Supporting the Size Determination of the Current Ingestion Pathway EPZ

There are potentially many important inputs to an analysis of ingestion dose. Some are related to the calculation of ground contamination levels (source term, dispersion data, deposition velocity) and others are related to the uptake of contamination in the food chain and the impact of a given level of contamination in the body. No attempt has been made to reproduce or critique the NUREG-0396 supporting analyses in terms of the complete dose pathway (except to observe that peak centerline ground concentrations were used to predict contamination of all of the milk ingested by the infant), but an assessment of ground contamination levels has been made as described in this section and the next.

The current PAG Manual [E-3] provides a means of relating ground contamination to infant thyroid dose. This topic is covered in Chapter 6 of the PAG Manual. Chapter 6 was prepared by the US Food and Drug Administration, Bureau of Radiological Health in August, 1982 and is thus roughly contemporaneous with NUREG-0396.

Table 9 of Chapter 6 of the PAG Manual gives the I-131 ground concentration that would yield an infant thyroid dose of 1.5 rem. This I-131 ground contamination is 0.13 $\mu Ci/m^2$. In order to make clear the relationship between the magnitude of the I-131 release, the atmospheric transport and deposition, and the resultant infant thyroid dose versus distance, the following manual analysis has been undertaken.

One reason for performing a manual analysis (rather than using a consequence code) is because of uncertainty with regard to the meteorological input and the I-131 deposition velocity that was used in performing the NUREG-0396 analysis. By considering a spectrum of meteorological conditions and deposition velocities and applying them manually, the analysis becomes more transparent; and it becomes possible to better assess the impact of these uncertainties.

E.3.1 Transport and Deposition Modeling for the Manual Analysis

There are two parts to the model, a transport part and a deposition part. The transport part consists of a plume expansion model which dilutes the plume both in the vertical (z) and in the crosswind (y) direction as it moves away from the point of release. It is assumed that all of the activity in the plume is concentrated in a plume element of thickness dx where x is the downwind direction. The expansion model assumes a Gaussian concentration distribution in both directions (y and z). The deposition model then imposes a sedimentation velocity on the vertical distribution. Activity is removed from the plume (i.e., deposited on the ground) according to the following expression:

$$dCi\{t\}/dt = -(Ci)(V_d)/(z_{eff})$$

where Ci{t} represents the activity in the plume as a function of time, V_d the deposition velocity, and z_{eff} the effective plume height. The effective plume height takes into account that the concentration in the vertical direction is non-uniform and that the near-ground concentration is the appropriate concentration to use with V_d.

The transport model described above is a simplified version of that from the MACCS consequence code [E-4]. The distribution of activity in the expanding plume is assumed to be normal with the sigma given by the following expressions for the vertical and crosswind directions, respectively:

$$\sigma_y = ax^b$$

$$\sigma_z = cx^d$$

The constants are functions of the atmospheric stability class.

Z_{eff} is a function of σz being approximately equal to $\sigma z/\sqrt{(2/\pi)}$. Therefore,

$$dCi\{t\}/dt = -(Ci)(V_d)/(z_{eff}) = -(Ci)(V_d)\sigma(2/\pi)/(\sigma_z) = -(Ci)(V_d)(0.8)/cx^d$$

Then, by defining the translational velocity of the plume (i.e., the wind speed) as $V_t = dx/dt$,

$$dCi\{x\}/dx = dCi\{t\}/dt/(dx/dt) = -(Ci\{x\})(V_d/V_t)(0.8/c)x^{-d}$$

Integrating,

$$\ln(Ci\{x\}) = -(Vd/Vt)(0.8/c)x^{(1-d)}/(1-d) + \text{Constant}$$

$$Ci\{x\} = Ci_0 \exp[-(0.8/c/(1-d))(Vd/Vt)x^{(1-d)}]$$

Knowing Ci{x} and recognizing that ground deposition, dG{x}/dx, is just dCi{x}/dx with opposite sign,

$$dG\{x\}/dx = (Ci)(V_d/V_t)(0.8/c)x^{-d}$$

$$= Ci_0 \exp[-(0.8/c/(1-d))(Vd/Vt)x^{(1-d)}](V_d/V_t)(0.8/c)x^{-d}$$

At the start of any x-interval $x_1 \rightarrow x_2$ (just in advance of the concentrated plume) the ground concentration is zero; therefore $\Delta G/\Delta x$ (where Δx is $x_2 - x_1$) represents the accumulation of activity on the ground as the plume passes over the interval $x_1 \rightarrow x_2$. Knowing y{x} (the crosswind dimension of the plume which may be assumed to be $4\sigma_y\{x\}$ to capture 95% of Ci{x}), one can calculate the average ground concentration over the interval as

$$\Delta G/\Delta x/4\sigma_y\{x\}/CF \approx Ci_0 \exp[-(0.8/c/(1-d))(Vd/Vt)x^{(1-d)}](V_d/V_t)(0.2/ac/CF)x^{-(b+d)}$$

where CF is a plume-meander correction factor.

For a normal distribution, the peak concentration (i.e., under the plume centerline) will be approximately 1.7 times that of the average, and the CF will be approximately 2.8 for an assumed 10-hour release duration (the maximum value considered in NUREG-0396). Therefore, the final expression for the centerline ground contamination concentration as a function of distance, $GC_{cl}\{x\}$ is:

$$GC_{cl}\{x\} = Ci_0 \exp[-(0.8/c/(1-d))(Vd/Vt)x^{(1-d)}](V_d/V_t)(0.12/ac)x^{-(b+d)}.$$

This expression can be used with a known source term (Ci_0), ratio of deposition velocity to wind speed (V_d/V_t), and plume expansion coefficients (a, b, c, and d) to calculate $GC_{cl}\{x\}$.

E.3.2 Source Term Input

The source terms (releases of radioactivity to the environment) are taken from reference [E-2] (the RSS). The RSS source terms include nine PWR source terms and five BWR source terms. The source terms which correspond to "core melt" are PWR-1 through PWR-7 and BWR-1 through BWR-4. The combined frequency of the PWR core melt source terms is 6E-5 per reactor-year, and for the BWRs the value is 3E-5 per reactor-year. Using a weighted average of 2/3 PWR and 1/3 BWR, the overall core melt frequency used in NUREG-0396 to represent US plants is 5E-5 per reactor-year. With this value as a basis, the conditional probabilities (given core melt) for the RSS core melt source term set is as follows:

S/T ID #	I Release Fract	Frequency /yr	Weighted Freq /yr	Cond Prob
PWR-1	0.7	9E-7	6.0E-7	0.012
PWR-2	0.7	8E-6	5.4E-6	0.108
PWR-3	0.2	4E-6	2.7E-6	0.054
PWR-4	0.09	5E-7	3.4E-7	0.007
PWR-5	0.03	7E-7	4.7E-7	0.009
PWR-6	0.0008	6E-6	4.0E-6	0.080
PWR-7	0.00002	4E-5	2.7E-5	0.540
BWR-1	0.4	1E-6	3.3E-7	0.007
BWR-2	0.9	6E-6	2.0E-6	0.040
BWR-3	0.1	2E-5	6.7E-6	0.134
BWR-4	0.0008	2E-6	6.7E-7	0.013
Total	N/A	9E-5	5.0E-5	1.00

The important feature of this source term array is that the iodine release fraction (and here we mean inorganic iodine forms since organic iodine forms are gaseous and will not deposit) is large (i.e., greater than a few percent) for all but three of the release categories. These three (PWR-6, PWR-7, and BWR-4) have a combined conditional probability of about 0.63. One can say, therefore, that in order for the exceedance probability of a given level of radioiodine ground contamination to be less than about 0.4 at some distance (given this array of core melt source terms), it must be that:

- For most weather conditions PWR-6/BWR-4 will yield less than that level of ground contamination at that distance, and
- For all weather conditions PWR-7 will yield less than that level of ground contamination at that distance.

This can be better appreciated by inspecting Figure E-2. This figure shows graphically the distribution of inorganic iodine release fractions for the above set of source terms. Note the bimodal distribution of the iodine releases. The first block of data (> 20% iodine release) includes the conditional probabilities for PWR-1 through PWR-3 and BWR-1 and BWR-2. The second block of data (approximately 10% iodine release) includes PWR-4 and BWR-3. The third block of data (3% iodine release) is PWR-5. The fourth block of data includes PWR-6 and BWR-4, and the last block of data is PWR-7. Note that the combined conditional probabilities of the first three blocks (inorganic iodine release fractions greater than a few percent) is 0.37. The smallest inorganic iodine release for this group (three percent) is nearly 40 times larger than the largest release for what remains (0.08%). Therefore, one can readily see that for this set of source terms, whenever a given ground contamination level of radioiodine is exceeded roughly 40% of the time, it is likely that source terms greater than a few percent will establish that probability by themselves, and that the remaining source terms (0.08% or less) would never, or at most rarely, exceed that level of ground contamination no matter what the weather (and associated plume dispersion and deposition) might be. In fact, this is the case.

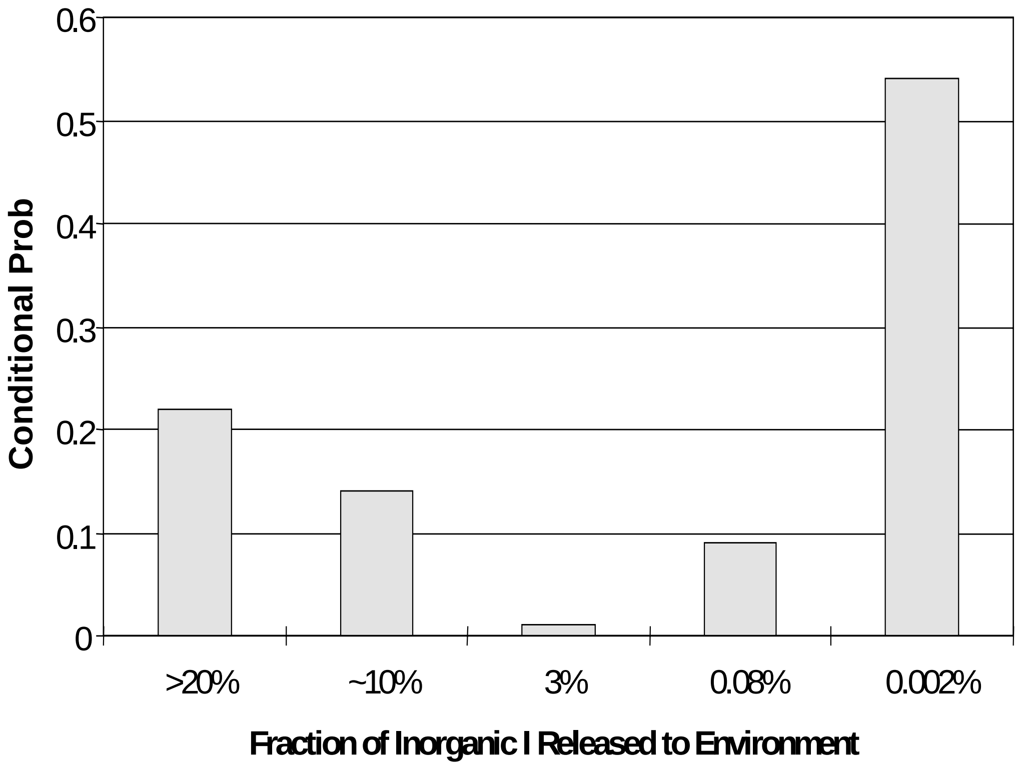

Figure E-2
WASH-1400 Core Melt S/T Distribution

In NUREG-0396 the exceedance probability of 1.5 rem to the infant thyroid (corresponding to a ground contamination of about 1.3E-7 Ci of I-131 per m^2) is stated to be about 0.4 at 50 miles (80000 meters). As alluded to above and as will be seen below, all RSS source terms greater than PWR-6, PWR-7, and BWR-4 will lead to ground concentrations exceeding 1.3E-7 Ci/m^2 at 50 miles for virtually any weather condition. Similarly, source terms PWR-6, PWR-7, and BWR-4 will lead to ground concentrations less than 1.3E-7 Ci/m^2 at 50 miles for virtually any weather condition.

The core power level used in the NUREG-0396 assessment is assumed to be that corresponding to about a 1000 Mw(e) plant (approximately 3200 Mw(t)), leading to an assumed core inventory of about 8.25E7 Ci of I-131. Therefore, it will be assumed that the PWR-5 inorganic I-131 release is about 2.5E6 Ci, the PWR-6/BWR-4 inorganic I-131 release is about 6.6E4 Ci, and the PWR-7 inorganic I-131 release is about 1650 Ci .

E.3.3 Meteorological Input

The meteorological input of interest is the wind speed and stability of the atmosphere carrying the radioactive plume. Wind direction is not important in the analysis because it is assumed that during the period of the bulk of the release and deposition, the wind direction may be constant (or nearly constant), so a uniform wind direction is assumed.

The ALWR Utility Requirements Document (URD) [E-5] treatment of ALWR site meteorology was based on reference [E-6]. A review of reference [E-6] shows the following:

Stability Class	Observations	Percentage	Typ Wind Spd	Percentage
A	1452	16.6	4 - 7 MPH	50.4
B	410	4.7	4 - 7 MPH	53.4
C	762	8.7	4 - 7 MPH	55.0
D	2322	26.5	4 - 7 MPH	49.7
E	2087	23.8	4 - 7 MPH	48.1
F	1727	19.7	1 - 3 MPH	57.7
All	8760	100	N/A	N/A

Given a stability class and a representative wind speed for that stability class, the expression for plume centerline ground contamination concentration as a function of distance can be solved manually for a given source term and deposition velocity. Based on the dominance of wind speeds in the range of 4-7 MPH (average of 5.5 MPH or 2.5 m/s) for Classes A through E and 1-3 MPH (average of 2 MPH or 0.9 m/s) for Class F (approximately 50 percent of the observations or more), these wind speeds will be used to represent the classes. Once the deposition pattern is clear for the ALWR reference site meteorology (a site chosen because of the prevalence of stable conditions tending to maximize centerline concentrations), one can anticipate how the deposition pattern might vary for the more "average" weather conditions assumed in NUREG-0396. This expected variation can then be compared to NUREG-0396, Figure I-14 (see Figure E-1) to determine if the processes modeled in this manual analysis are a fair approximation of the computerized analysis supporting NUREG-0396.

E.4 Manual Analysis of Ground Deposition vs. Distance Using NUREG-0396 Input and Comparison to NUREG-0396

Sections E.3.1, E.3.2, and E.3.3 above describe the model, source term input data, and meteorological input data, respectively, for a manual simulation of the computerized analysis supporting the NUREG-0396 selection of a 50-mile radius EPZ for the ingestion pathway. In this manual simulation, three source term cases will be presented, a three percent inorganic I-131 release (i.e., 2475 KCi), a PWR-6/BWR-4 inorganic I-131 release (i.e., 66 KCi), and a PWR-7 inorganic I-131 release (i.e., 1.65 KCi). For the meteorological input, the following constants will be used (wind speeds are from Section E.3.3 and the sigma constants are from reference [E-4]):

Stability	Wind Speed V_t (m/s)	Sigma-y a	Sigma-y b	Sigma-z c	Sigma-z d
A	2.5	0.3658	0.9031	0.00025	2.125
B	2.5	0.2751	0.9031	0.0019	1.6021
C	2.5	0.2089	0.9031	0.2	0.8543
D	2.5	0.1474	0.9031	0.3	0.6532
E	2.5	0.1046	0.9031	0.4	0.6021
F	0.9	0.0722	0.9031	0.2	0.6020

For the iodine deposition velocity the same value that was used in the PAG manual (i.e., 0.01 m/sec) will be used here. This is believed to be what was used in NUREG-0396, although NUREG-0396 is not explicit in this regard.

Figures E-3 through E-5 show the I-131 ground concentration vs. distance from the point of release for the three source terms and the six cases of atmospheric stability. For the three percent I-131 source term (Figure E-3) the ground contamination exceeds the threshold value of 1.3E-7 Ci/m^2 out to 80 Km (50 miles) for all stability classes except A, and even for A the threshold value is exceeded out to more than 40 Km (approximately 25 miles). Therefore, it is evident that for source terms consisting of several percent of the core inventory of I-131 (and especially, for tens of percent of the core inventory of I-131), the conditional probability of exceeding an infant thyroid dose of 1.5 rem (based on plume centerline ground concentration) approaches unity for virtually any site. For the RSS source term set, this would be the case for all source terms except the two source terms covered by Figures E-4 and E-5.

From Figure E-4 (and the ALWR reference site atmospheric stability conditional probabilities given in the table above) one can observe that for the PWR-6/BWR-4 source term the conditional probability of exceeding the threshold value of 1.3E-7 Ci I-131/m^2 is about 83% at 40 Km and 59% at 80 km. From Figure E-5 one can observe that for the PWR-7 source term the corresponding conditional probabilities are about 50% and 0%. Since the conditional probabilities (given core melt) for the source terms are 54% for PWR-7, 9% for PWR-6/BWR-4, and 37% for the remainder, the conditional probabilities of exceeding the threshold value of 1.3E-7 Ci I-131/m^2 given core melt and ALWR reference site weather can be estimated to be (0.54)(0)+(0.09)(0.59)+(0.37)(1) = 0.42 at 80 Km and (0.54)(0.5)+(0.09)(0.83)+(0.37)(1) = 0.71 at 40 km.

For more average weather conditions, one can assume a more even distribution of stability classes; i.e., a shift from D, E, and F to A, B, and C. For example, if there were the same number of observations in each stability class (16.7%), then the conditional probability of exceeding the threshold value of 1.3E-7 Ci I-131/m^2 given PWR-6/BWR-4 would remain at about 83% at 40 km but would decrease to about 50% at 80 km. For PWR-7 the corresponding values would be 33% at 40 km and 0% at 80 km. The conditional probabilities of exceeding the threshold value of 1.3E-7 Ci I-131/m^2 given core melt and this "uniform" weather would be (0.54)(0)+(0.09)(0.5)+(0.37)(1) = 0.42 at 80 km (a negligible change as compared to the ALWR reference site case) and (0.54)(0.33)+(0.09)(0.83) +(0.37)(1) = 0.62 at 40 km.

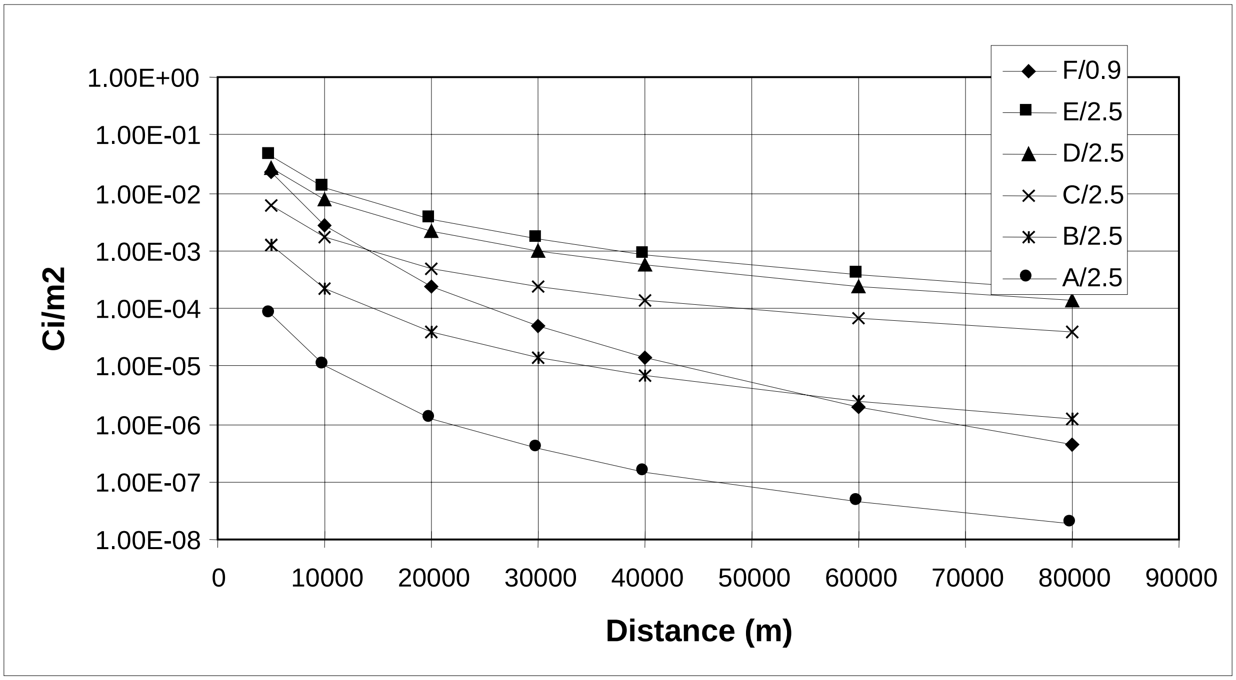

Figure E-3
Ci/m^2 Vs. Distance for 3% (2475 KCi) I-131 Release

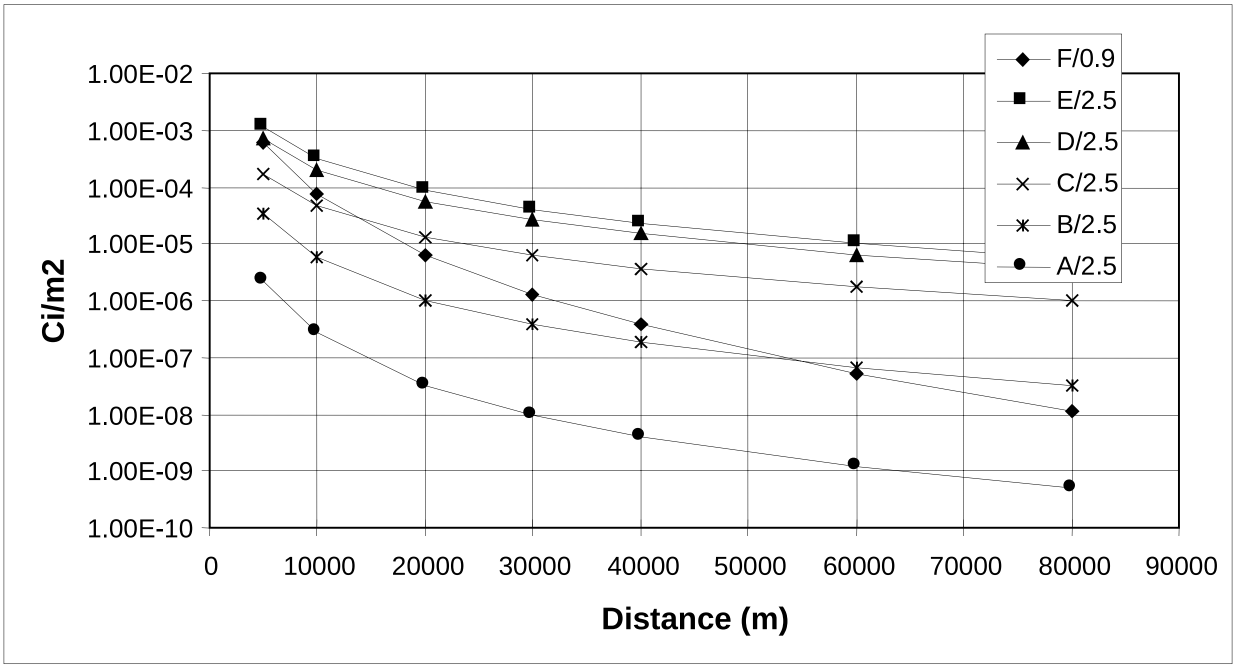

Figure E-4
Ci/m^2 Vs. Distance for 0.08% (66 KCi) I-131 Release

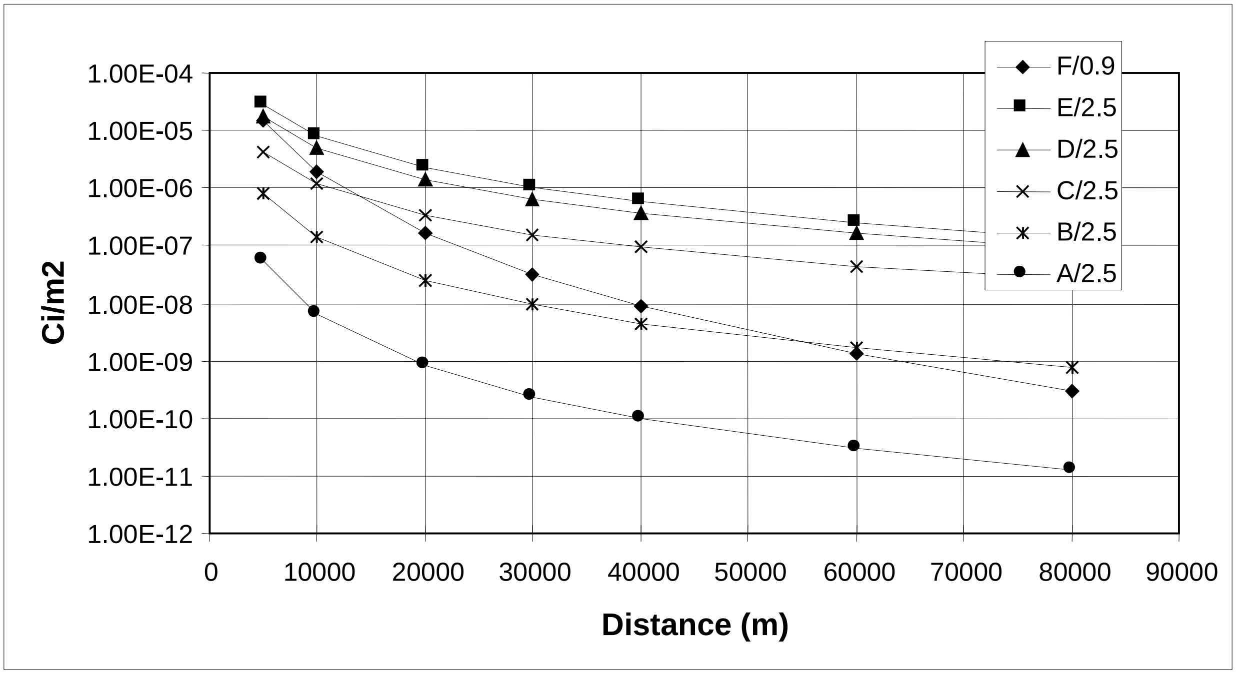

Figure E-5
Ci/m^2 Vs. Distance for 0.002% (1.65 KCi) I-131 Release

The reason for the negligible change at 80 km is that so much of the conditional probability of exceedance at that great distance is determined by the conditional probability of source terms exceeding several percent of the core inventory of I-131. At 80 km the PWR-7 source term will not produce ground concentrations exceeding 1.3E-7 Ci I-131/m^2 for any weather condition, and the PWR-6/BWR-4 source term will do so only for C, D, and E stability. As long as there is no dramatic shift in the conditional probability of the combined C, D, and E stability values in going from site to site, then the conditional probability of exceeding 1.3E-7 Ci I-131/m^2 given the RSS source term set will remain at about 40% (the value pointed out in NUREG-0396) no matter what site is chosen.

At 40 km the situation is different. At this distance the threshold ground concentration value is exceeded for virtually all source terms (independent of atmospheric stability) except for PWR-7. Therefore, one "begins" with a conditional probability of about 0.45 that the value will be exceeded independent of site meteorological variations. The "weather" will then determine the fraction of the conditional probability of PWR-7 (0.54) that will be added to that value. Referring to Figure E-4, if the fraction of D and E stability is high (e.g., about 50% as for the ALWR reference site), then the conditional probability that the 1.3E-7 Ci I-131/m^2 threshold value for ingestion zone planning will be exceeded, given the RSS core melt source term set, will be about 0.7. If the fraction of D and E stability is lower (e.g., about 33% for a uniform distribution of stabilities), then the conditional probability will be only slightly more than 0.6. This value is quite comparable to the value at 25 miles (40 km) plotted on Figure I-14 of NUREG-0396.

Using the manual technique described above, one can create an approximation of Figure I-14 of NUREG-0396 by obtaining the ground concentration vs. distance for each of the RSS inorganic I-131 source terms and stability class/wind speed combinations and then weighting them appropriately according to probability. The result is Figure E-6. Figure E-6 is a comparison of a manually-generated plot of the probability of exceeding 1.5 rem infant thyroid dose vs. distance to that from the actual Figure I-14. The assumption made in generating this manual plot is that all of the six stability class/wind speed combinations have an equal probability of occurring. One can observe that the agreement is reasonably good; if anything, the manually-generated plot is slightly more pessimistic than that from NUREG-0396, Figure I-14.

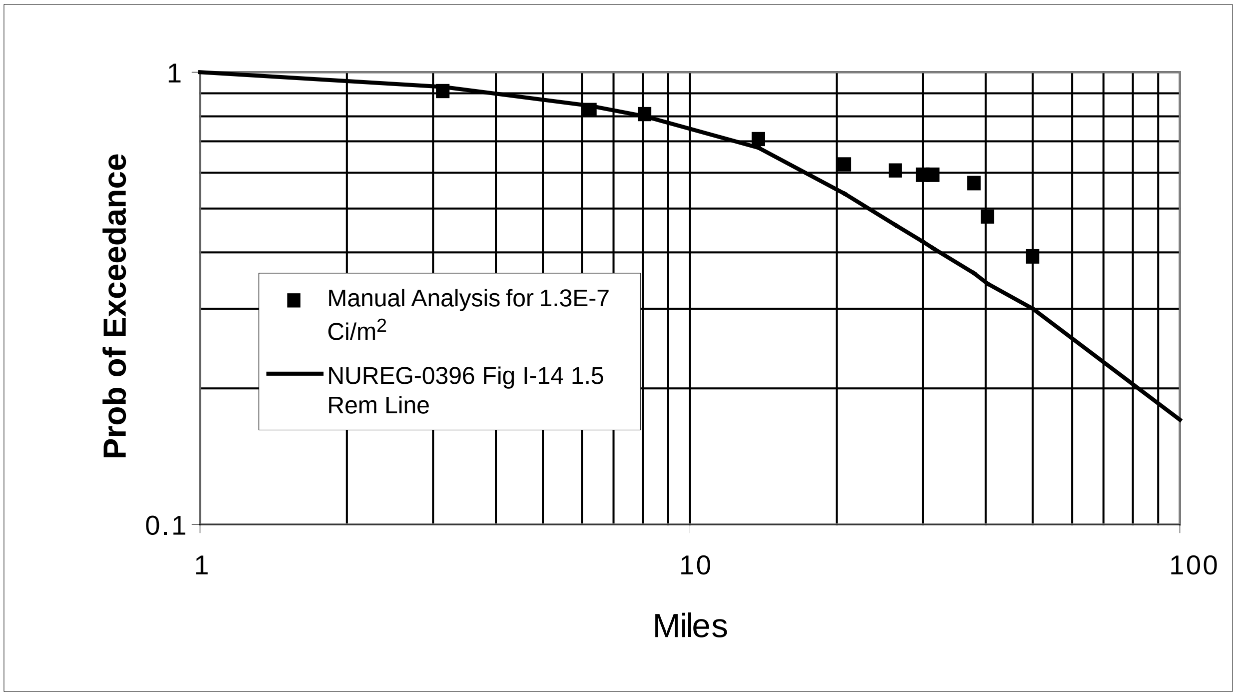

Figure E-6
POE Centerline Dose of 1.5 Rem Thyroid (or 1.3E-7 Ci/m^2 I-131) Vs. Distance

Given the good agreement obtained between NUREG-0396, Figure I-14 (Figure E-1) and the results from the manual analysis with an iodine deposition velocity of 1 cm/sec and a uniform distribution of stability classes, this combination of model and inputs will be used to evaluate the potential for ingestion pathway emergency planning simplification using the improved characteristics of the ALWR source terms.

E.5 Technical Basis for Ingestion Pathway Emergency Planning for ALWRs

In Section E.2 the basis for the current ingestion pathway EPZ radius of 50 miles was explained. This radius is tied to the probability of exceeding the 1.5 rem infant thyroid PAG conditional on core melt, and it is governed by the magnitude and the associated conditional probability of accidental inorganic I-131 releases. Other ingestion pathway exposures were considered in

NUREG-0396 (e.g., Cs-137 uptake in vegetables), but the infant thyroid dose and the associated I-131 source term were found to be controlling. Since the ratio of I-131 release to that of Cs-137 is about the same for the ALWRs now as it was for the plants studied in WASH-1400 (roughly equal fractions of the core inventory for both radionuclides, even though the absolute magnitudes have decreased), it would be anticipated that the I-131 release and the infant thyroid dose would continue to be controlling for the ALWRs.

From the analysis presented in the previous section, it is evident that the dose exceedance curve for the 1.3E-7 Ci I-131/m^2 threshold can be viewed as depending upon two types of source terms:

1. Large inorganic iodine source terms (of the order of a few percent or more of the core inventory of radioiodine) which will yield plume centerline I-131 ground contamination levels exceeding the threshold value for required emergency actions (i.e., placing dairy cattle on stored feed to eliminate any chance that the radiation doses to an infant's thyroid could exceed 1.5 rem) out to 50 miles for virtually any site. These source terms are the result of some kind of containment failure.

2. Smaller inorganic iodine source terms from intact (or nearly intact) containment sequences (similar to PWR-7) in which unfavorable meteorology and/or a reduced distance from the point of release is necessary to bring about a plume centerline I-131 ground contamination level exceeding the threshold value for emergency actions. For example, the conditional probability of such a plume centerline I-131 ground concentration for PWR-7 is essentially zero at 50 miles. At 25 miles it is in the range of 0.3 to 0.5 depending on the site.

For the RSS source term set used in NUREG-0396 the first type of source term was controlling in determining the 50-mile ingestion pathway EPZ radius. For ALWRs, as is discussed below, it is the second type of source term that is controlling; and, therefore, the required distance for ingestion pathway emergency planning is not as great. This change in the de facto basis for ingestion pathway emergency planning follows from ALWR designs and our current understanding of severe accident progression and containment response. That is, while the NUREG-0396 authors believed that almost 40% of the core melt accidents would result in "large" source terms (and in corresponding ground contamination levels exceeding the threshold for ingestion pathway emergency planning out to 50 miles essentially independent of weather), we now know that large source terms arising from containment failure are extremely unlikely. This fact establishes that for ALWRs in particular, "small" source terms will dominate, and the fraction of core melt accidents leading to the threshold level of ground contamination for ingestion pathway protective actions will be largely determined by intact (or nearly intact) containment radioiodine release fractions and the probability of unfavorable weather at distances less than 50 miles.

It is interesting to consider the implications of what we now understand to be the case for the ALWRs as opposed to what the NUREG-0396 authors thought in 1978 to be the case for the then-operating plants. The range of inorganic radioiodine source terms believed in 1978 to have substantial conditional probabilities of occurrence in NUREG-0396 was more than four orders of magnitude (i.e., PWR-1, PWR-2, PWR-3, BWR-1 and BWR-2 at >0.2 vs. PWR-7 at 0.00002). Therefore, while one might say that the combination of a PWR-7 source term and unfavorable weather conditions might just produce the threshold level of radioiodine ground contamination requiring protective actions at 50 miles; in fact, the actual ground contamination level (according to NUREG-0396) might readily be 10,000 times greater. If, on the other hand, one recognizes that the real variation is not so much one of potential source term magnitude as one of weather and deposition, then one sees the potential for having extremely large exceedances of the theshold level being much more remote. For example, the variation between the worst ground contamination and the median ground contamination on Figures E-3 through E-5 is about an order of magnitude. Thus, although the conditional probability of exceeding 1.3E-7 Ci I-131/m^2 might remain the same for small source terms at reduced distances as compared to large source terms at fifty miles, the potential degree of exceedance (given that large source terms are now seen correctly as having negligible conditional probabilities) is much less for the small source terms at the reduced distance than for the large source terms envisioned in NUREG-0396. It should also be noted (referring to Figures E-3 through E-5) that decreasing the ingestion pathway emergency planning distance by half, by itself, increases the ground concentration by generally less than one order of magnitude. This is far less than the perceived uncertainty in the source term in NUREG-0396.

There are two parts to the basis for ingestion pathway emergency planning simplification for ALWRs as discussed below. The first is the fact that the frequency of large core melt source terms (those involving containment failure) has decreased to the point where these source terms now constitute an essentially negligible contribution to the probability of exceedance of the threshold contamination level for ingestion pathway emergency planning at any distance. Though such source terms may still control the <u>fraction</u> of the cases that exceed the threshold contamination value at very great distances (say, 50 miles), the probability of exceeding the threshold value at so great a distance is now only seen to be of the order of one percent or less (as compared to a perceived 40% in NUREG-0396).

The second part of the basis is providing an adequate planning distance for small core melt source terms (those for which the containment remains intact or nearly intact).

E.5.1 Likelihood of Large Source Terms for ALWRs

For the purpose of this section a "large" source term is defined as one involving an inorganic iodine release large enough to produce a plume centerline ground concentration level greater than 1.3E-7 Ci I-131/m^2 at distances out to 50 miles for virtually any weather condition. This means an iodine release of several percent of the core inventory resulting from a loss of containment integrity. In the RSS [E-2] such source terms were expected in about 40 percent of the core melts.

For ALWRs, based on the discussion provided in the ALWR NUREG-0396 Assessment in the Main Report, there are no release categories with large source terms with frequencies exceeding the 10^{-7} per year probability screen. Even the review of release categories less than 10^{-7} per year does not produce releases exceeding 1% iodine due to aerosol retention phenomena and accident management actions. Therefore, the fraction of ALWR core melts exceeding the threshold ground contamination level requiring emergency response attributable to large source terms is negligible.

E.5.2 Definition of an Adequate Ingestion Pathway EPZ Radius for Small Source Terms

Each ALWR has calculated a source term (referred to as the "PAG-comparison source term") to address URD provisions [E-5]. This PAG source term is based on a core melt in which the containment boundary remains intact and containment systems remain functional. The full NUREG-1465 [E-7] source term is released to the containment, and a mechanistic analysis of the release to the environment is made. The duration of the release is set at 24 hours; and as such, this source term is intended to represent a near-maximum (if not an absolute maximum) release to the environment over the first 24 hours for a core melt accident with an intact containment. These releases are discussed in Appendices B, C, and D of this report for AP600, System 80+, and ABWR, respectively.

The maximum value that these releases can have and still be less than the plume exposure PAGs for 24 hours (as required by the URD) is about 1730 Ci of inorganic I-131, or about five percent more than the PWR-7 release used for Figure E-5. This value has been determined using the results from the System 80+ PAG-comparison calculation presented in Section 15.6.5.5 of reference [E-8]. For this System 80+ PAG-comparison calculation the inorganic I-131 release is 935 Ci. The resultant TEDE is about 0.33 rem (compared to a 1 rem PAG) and the resultant thyroid dose is about 2.7 rem (compared to a 5 rem PAG). Since neither PAG may be exceeded, the I-131 release could be increased by only 5/2.7 = 1.85 (or to approximately 1730 Ci) and still be within the thyroid PAG. This result would be largely independent of plant type. (For example, the ABWR PAG-comparison source term involves an inorganic I-131 release of approximately 1060 Ci, far less than the 1730 Ci just described). Therefore, one can say that the 1730 Ci of I-131 in inorganic form released over a 24 hour period represents a very high estimate of the “small” source terms which will dominate ALWR core damage accidents.

Figure E-7 shows the I-131 plume centerline ground contamination concentration as a function of distance for this maximum “PAG-comparison source term”. Figure E-7 differs from Figure E-5 in three ways. First, the I-131 source term is slightly larger (1730 Ci vs. 1650 Ci as explained above). Second, instead of basing the σ_z (vertical plume expansion) coefficients on reference [E-4], the basis for the σ_z coefficients used for this figure is reference [E-5]. The reason reference [E-5] changed the σ_z coefficients is to better match the near-field plume expansion used in the models supporting NUREG-0396 (i.e., within the first one-half to three miles from the point of release). Since no attempt was made to explore the impact of the changed coefficients beyond that distance, it was decided to base Figures E-3 through E-5 on the reference [E-4] values. However, one can see that the difference between the Figure E-7 and E-5 results is small (taking into account the five percent difference in source terms), the only noticeable effects being about an order of magnitude increase in the results for "B" stability and a much less rapid drop-off in the results with distance for "F" stability. Qualitatively the results do not change at all.

The other major difference between Figures E-7 and E-5 is the addition of some MACCS results to Figure E-7 using the maximum “PAG-comparison source term” of 1730 Ci of I-131. Since MACCS does not provide an option for I-131 plume centerline ground contamination concentration, these results were "backed-out" using the median 24-hour centerline ground-exposure dose for I-131. The median dose vs. distance was divided by the dose conversion factor for 24 hour ground exposure and the associated protection factor to obtain the median plume centerline ground concentration of I-131 vs. distance over a range of distances from about 15 miles to about 35 miles. These median results (heavy dark line on Figure E-7) show excellent agreement with the manual model based on the fact that the "D" and "E" stability results (with about a 50% probability of occurring given the ALWR reference site meteorology used in MACCS) exceed the median MACCS results while the other 50% of the data are less than the MACCS results.

Figure E-7 indicates that even using the ALWR reference site weather, the conditional probability of exceeding the threshold value of I-131 ground contamination requiring protective actions for the maximum “PAG-comparison source term” is less than 40% within 25 miles (40 km) of the point of release. This is based on the fact that the MACCS results and the supporting manual results show a median ground concentration well under the 1.3E-7 Ci/m^2 threshold value at that distance (about a factor of two less). For more average weather (in line with the assumptions of NUREG-0396) the distance would be less than 20 miles. (Observe from Figure E-5 that with a uniform conditional probability of the six stability classes, the probability of exceeding 1.3E-7 Ci I-131/m^2 would be about 0.33 at 20 miles (32 km)). Therefore, as long as the conditional probability of a large release remains negligibly low, no "unplanned" protective actions would be required for more than about one-third of the core melts beyond this distance. It is also worth noting that even at larger distances (for example, 35 miles, i.e. 56 km), there is virtually no possibility that the threshold value would be exceeded by more than about a factor of two or three. Given the conservatism of using a plume centerline ground contamination concentration as the starting point for a milk-ingestion pathway, this is not a significant exceedance.

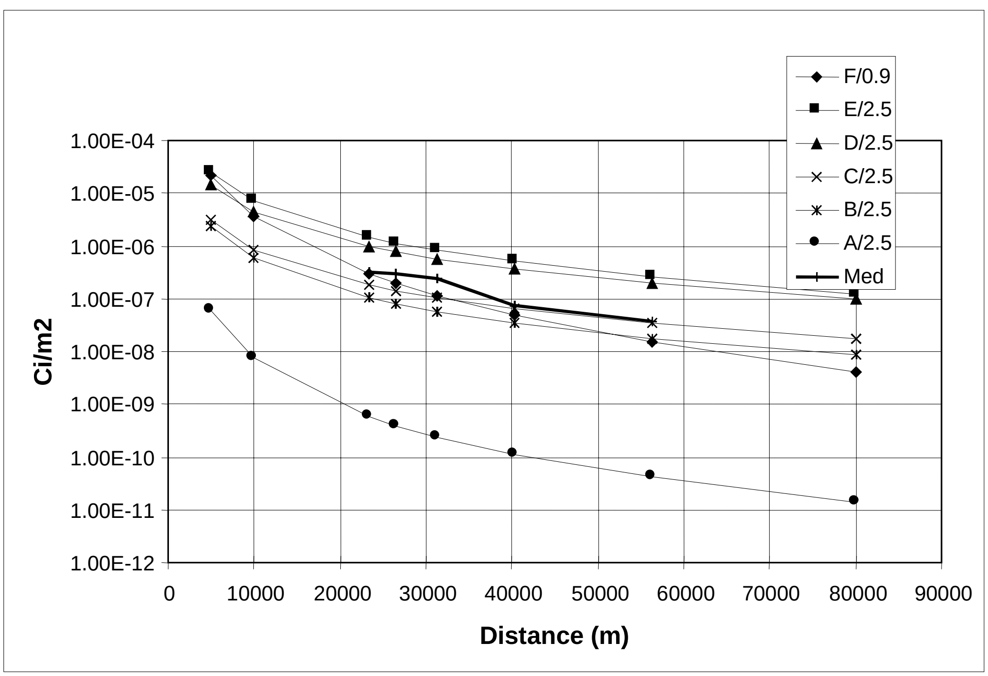

Figure E-7
Ci/m^2 Vs Distance (m) for 1.73 KCi Release, σz Based on URD

E.6 Summary

This appendix has reproduced the NUREG-0396 basis for the current 50-mile radius of the ingestion pathway EPZ, it has re-examined the modeling and inputs for the analyses supporting the NUREG-0396 position, and it has presented analyses which parallel those of NUREG-0396 but which have used source term information and requirements appropriate for the ALWRs. The conclusion from this work is that a basis exists for an ingestion pathway planning distance of 20 to 25 miles for ALWRs.

E.7 References

E-1. "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396/EPA 520/1-78-016, December 1978

E-2. "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH 1400, October 1975.

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F
ALWR SEVERE ACCIDENT MANAGEMENT PROGRAM

Introduction

The purpose of this appendix is to provide the basis for crediting severe accident management actions in evaluating mitigation of ALWR core damage accidents. The ALWR Utility Requirements Document (URD) requires that ALWRs have a Severe Accident Management Program (SAMP), including development of guidelines and training to identify and facilitate actions that may be taken to prevent and mitigate the effects of beyond design basis accidents. While specific plans, training, and guidelines for ALWRs are the responsibility of the Plant Owner, Severe Accident Management Guidelines (SAMGs) are in place at operating plants and have been reviewed here to provide a basis for understanding the role of the SAMP for ALWRs. In addition, work performed by the Plant Designers on ALWR accident management has been reviewed.

For operating plants, severe accident management is the last element to be completed by licensees in a series of activities initiated by a Commission policy statement [F-1] and further described in NRC documentation on severe accident closure [F-2]. A formal industry position was provided in an NEI technical report [F-3]. This report was issued in lieu of a formal regulatory requirement to establish the assessment and enhancement objectives, methods and guidance. Plant specific implementation of industry guidelines is essentially complete. Formal closure of the issue awaits a final decision by NRC with respect to the regulatory oversight process. The target date for implementation of a new oversight process is January 2000.

The purpose of the SAMP for ALWRs is to extend the defense-in-depth principal to the plant operating staff by extending operating procedures well beyond the plant design basis into severe fuel damage regimes. The goal of this program is to take advantage of existing plant equipment and operator skills and creativity in pre-planning and training for ways to terminate possible accidents beyond the design basis or to limit off-site releases. A wide range of existing onsite and and off-site equipment can be useful during beyond design basis accident situations and this program provides for their use through pre-planning, training and guidelines.

Review of Operating Plant SAMGs

The SAMGs for a Westinghouse four-loop plant and two Combustion Engineering plants were reviewed as examples to understand their significance relative to severe accident specific actions referred to in Section 4 of the Main Report. Of particular interest here are actions to mitigate risks from steam generator bypass sequences and the use of a non-safety containment spray system. It is expected that ALWRs will adopt guidelines similar to the SAMGs as part of the SAMP that is required by the URD.

Emergency response personnel of the Technical Support Center (TSC) use the SAMGs when directed from the control room. They provide a diverse method to diagnose and implement corrective actions to mitigate severe accidents. The SAMGs contain a diagnostic section that uses flow charts and tables to determine the candidate high level actions (CHLAs) to be taken. The CHLAs are prioritized for each plant and its plant damage conditions, which are defined in the diagnostic section. A combined general list of CHLAs for Combustion Engineering and Westinghouse plants and their approximate priorities are shown in Table F-1.

Table F-1
Candidate High Level Actions for PWRs

Priority	Candidate High Level Actions
1	Inject into the RCS
2	Depressurize the RCS
3	Inject into the Steam Generators
4	Depressurize the Steam Generators
5	Spray into Containment
6	Operate Containment Fan Coolers
7	Vent Containment
8	Operate Hydrogen Recombiners
9	Flood the Reactor Cavity
10	Flood the Safety Equipment/Radwaste Building
11	Restart the Reactor Coolant Pumps
12	Vent the RCS
13	Spray the Safety Equipment/Radwaste Building
14	Spray the Outside of the Containment

The SAMGs provide specific and detailed implementation guidelines for each CHLA. They include the names and identification numbers of components to be operated as well as engineering data, charts, and other information on the characteristics of equipment and processes incorporated in the action.

The first four CHLAs in Table F-1 are of most interest in addressing steam generator tube integrity risks. The first two address actions that can be taken on the primary side to cool the core and depressurize the RCS thereby preventing or arresting the thermal challenge to steam generators. The next two focus on the steam generators as a means of removing heat from the RCS, lowering primary system pressure (thus further promoting injection in the RCS), as well as protecting the steam generator tubes from over temperature conditions and scrubbing fission product aerosols if a rupture exists.

The specific actions identified in the SAMGs to depressurize and inject into the steam generators supplement those actions prescribed in the Emergency Operating Procedures. These SAMG actions include the use of all possible sources of water and pumps for injection as well the approach to be used to inject water into dry steam generators to minimize thermal shock. Typically the flows needed to maintain tube integrity or assure fission product aerosol retention are less than 100 gpm. Both high head pumps such as the auxiliary or emergency feedwater pumps and main feedwater pumps can be used. Low head pumps such as condensate pumps, makeup water transfer pumps and fire pumps are also possible candidates. In addition, if power is not available, the turbine driven auxiliary pump and diesel driven pumps provide a means of injection. An example of this approach is provided in one of the SAMGs. The CHLA provides detailed information and guidelines for using a portable diesel-powered pump. The CHLA indicates that the portable pump has a capacity of 250 gpm at a discharge pressure of 65 psig.

Severe accident sequences involving steam generator tube ruptures are characterized by the postulated type and timing of support system failures. The support systems required to implement the specific actions called for in the SAMGs are limited to those required to inject water into and vent steam generators such as sources of water, motive power, pumps, level indication, and vent valve operators. When ac power is available, there are a variety of pumps capable of injecting water into the RCS and the steam generators. If power is not available, either the turbine-driven auxiliary feedwater pump or a diesel-powered pump (e.g., portable fire pump) can be used. In either case, the procedure to setup and initiate injection and depressurization is straightforward and involves only a few steps. Thus through pre-planning and training and the use of SAMGs, it should be possible to significantly reduce the likelihood of long term dryout and heatup in the steam generator during a severe accident situation.

For release categories in Table 4-1 where the use of an accident management containment spray system is indicated, the ALWR SAMP will also provide the necessary pre-planning, training and guidelines for maximizing the successful operation of this system. In the passive ALWR, this non-safety system provides spray cooling of containment which decreases containment pressure and significantly reduces fission product aerosol concentration in the containment atmosphere.

Review of ALWR Plant-Specific Accident Management Work

References [F-4] and [F-5] document the accident management framework for AP600. Both references clearly state that steam generator injection is a high priority for accident management. For example, Table D.7-1 of reference [F-1] indicates that injecting into steam generators is a “high level action” for both stabilizing containment and terminating fission product release. A number of SG feed paths will be available. Suction sources include the condensate storage tank, raw water reservoir, fire water storage tank, makeup water storage tank, and condensate system. High pressure pumps include the auxiliary feedwater pumps and main feedwater pumps. Low pressure pumps include the makeup water transfer pumps, portable firewater pump, and condensate and booster pumps. Means for depressurizing the SGs include the various sets of steam dump valves. In the extremely unlikely event of a sequence of type 1A in which natural circulation in the RCS loops heats up a dry steam generator, the AP600 steam generator tubes would not reach creep rupture temperatures for several hours. This allows ample time for steam generator depressurization and water injection if required.

While details of AP600 accident management guidelines are not yet available, based on the above information the accident management priorities of injection into steam generators in AP600 is expected to be comparable to that in Westinghouse operating plants. Per reference [F-6], Westinghouse operating plant accident management training documents give highest priority to preventing failure of the final fission product boundary. The library of accident scenarios built to support accident management drills for Westinghouse plants includes a variety of SG tube failure sequences including spontaneous tube rupture and high RCS pressure, dry SG secondary sequences with potential induced tube rupture.

Reference [F-7] documents ABB commitments and expectations regarding System 80+ emergency operations guidelines (EOGs) and SAMGs for RCS depressurization and addition of water to the SG secondary side. Reference [F-4] indicates that while only input to these guidelines (as opposed to developing the guidelines themselves) was provided as part of System 80+ Design Certification documentation, ABB fully expects System 80+ EOGs and SAMGs to be consistent with what is being implemented for operating C-E plants and with industry guidance available at the time.

Summary

In summary, the operating plant SAMGs and the review of ALWR plant-specific accident management work provide a sound basis for understanding the additional defense-in-depth capabilities that the ALWR Severe Accident Management Program provides. The operating plant SAMGs are part of a formalized program prepared and maintained in response to NRC requirements. They provide a diverse, and focused set of actions for mitigating severe accidents. These actions, which are incorporated in documented guidelines and are based on pre-planning and training, can be quantified using human reliability and assurance (HRA) methods to show the extent to which they reduce the likelihood of severe accidents. Given the diversity of decision making provided by TSC personnel and guidelines, the pre-planned use of all available equipment, and the variety of equipment available, application of the the SAMGs to ALWRs are expected to significantly reduce the likelihood and consequences of steam generator bypass accidents and accidents involving fission product release to the containment atmosphere.

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F-6 R. Lutz, Westinghouse, personal communication to D. Leaver, Polestar, May, 1998.

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G

QUANTIFICATION OF FISSION PRODUCT AEROSOL RETENTION IN AN UNISOLATED STEAM GENERATOR TUBE RUPTURE SEVERE ACCIDENT

Introduction and Background

This report is to summarize and document work performed as part of the industry effort to improve and simplify emergency planning in support of the Advanced Light Water Reactor (ALWR) Program. Specifically, this work quantified the retention of fission product aerosol in the primary and secondary side of the steam generator for an unisolated steam generator tube rupture (SGTR) core damage accident sequence in a pressurized water reactor (PWR). This work has been supported by the Electric power Research Institute (EPRI).

An SGTR-initiated core damage sequence can potentially lead to containment failure, e.g., via a stuck open safety relief valve. Most probabilistic risk assessments (PRAs) and severe accident codes assume that a significant fraction of fission products flowing through an unisolated break in a steam generator tube escape to the environment. For example, in NUREG 1150 [G-1], the median estimate of the fraction of the core inventory of iodine released to the environment for the unisolated SGTR was 27% and the 95th percentile estimate was 80%. This estimate was based on an expert elicitation panel. NUREG 1150 states that based on the work of the panel, “there is a very good chance that there would be little retention of radionuclides in the steam generator.” In the AP600 PRA [G-2], the iodine release from an unisolated SGTR is 17%.

These results indicate that little credit has traditionally been given to aerosol removal mechanisms for unisolated SGTR sequences. In the work reported here, models have been developed for applicable aerosol removal mechanisms for these sequences. Four aerosol removal mechanisms have been considered: turbulent deposition for internal flow in the primary side of the broken steam generator tube(s), inertial impaction for external flow over the tubes on the secondary side (i.e., flow having exited the break), eddy-diffusion driven turbulent deposition for external flow, and thermophoretic deposition for flow over cooler steam generator secondary side structures. These are all potentially important aerosol removal mechanisms for unisolated SGTR sequences and consideration of these removal mechanisms will provide more realistic source term releases and more accurate risk estimates. It is noted that the four removal mechanisms are essentially independent from one another since only inertial impaction depends strongly on particle size and none depends on particle concentration.

Turbulent Deposition for Internal Flow (Primary Side)

The geometry and arrangement of a typical PWR steam generator are shown in Figure G-1. The aerosol-laden gas flows from the damaged core, through the hot leg to the SG inlet plenum, up the broken tube to the break location, and out the break to the secondary side. To predict aerosol behavior, it is first necessary to estimate the gas flow velocity in the broken tube(s) for the unisolated SGTR sequence. According to the theory of compressible fluid dynamics and as developed in reference [G-3] and applied to the SGTR problem in references [G-4] and [G-5], the tube flow under ideal conditions is determined by the pressure ratio, p_2 / p_1, of the primary side to the secondary side across the break, and the adiabatic exponent (k) that is the ratio of the heat capacity at constant pressure to that at constant volume (for steam, k is about 1.3). The flow will be choked when p_2 / p_1 is less than the critical pressure ratio. The critical pressure ratio is determined solely by the adiabatic exponent and has a value of 0.546 for $k = 1.3$. Under choked flow conditions, the flow velocity and mass flow rate are a function of the upstream conditions only. In a broken tube, the flow velocity inside the tube is expected to be of the order of several hundred of meters per second, even for small pressure differences, so the flow will be turbulent. As a result, turbulent deposition is assumed for the inside-the-tube aerosol retention calculation. This is a well-understood phenomena and may be expressed by the following equation [G-6]:

$$E_D = 1 - \exp\left[-4\left(\frac{L}{D}\right)\sqrt{\frac{f}{2}}V_d^+\right] \tag{1}$$

where E_D is the retention efficiency of aerosol particles due to turbulent deposition (defined as the ratio of the particle mass deposited in the tube to the particle mass entering the tube), *D* is the diameter of the tube, *L* is the distance between the bottom of the tube sheet and the break, and *f* is the Fanning friction factor (=$0.046/Re^{0.2}$ [G-7] for turbulent flow in a circular tube with the Re in the range of 3×10^4 to 10^6). V_d^+ is the dimensionless particle deposition velocity and will be a constant (=0.13) for particle diameter greater than 0.3 micron. Thus the turbulent deposition of aerosols in the cases of interest here is practically independent of aerosol particle size.

Often, the decontamination factor (*DF*) is used to represent the removal efficiency of aerosols in a given system. *DF* is defined as the ratio of aerosols entering the system to the rate of aerosols leaving the system. Thus, $DF = 1/(1 - E_D)$.

The *DF* from turbulent deposition in internal flow depends upon the length of tube through which the aerosol-laden gas flows before exiting the break. The minimum length is the thickness of the tubesheet. This would be the case if the tube rupture was located at the tubesheet surface. Since the tubesheet is typically about 27 inches thick, the minimum retention efficiency would be about 0.65 corresponding to a DF of ~3. At ~5 meters above the tubesheet (i.e., roughly half way up the tube bundle), the DF would be 999.

Inertial Impaction for External Flow (Secondary Side)

The gas and aerosols coming out the break in the tube will be in the form of a jet with an initial velocity of the order of several hundred meters per second. The jet is expected to impinge on an array of surrounding tubes, or on surfaces of plates and structures in the near field of the break, causing significant removal of aerosols due to inertial impaction (and eddy diffusion-driven turbulent deposition which is discussed in the next section). Figure G-2 illustrates the flow pattern.

The inertial impaction will be modeled as external flow over a cylinder (i.e., the SG tube). The key parameter here is the Stokes number of the particles in the gas flow which in turn depend on particle (gas) velocity and particle size.

In determining the gas velocity, it is noted that the initial velocity of the jet is high, but the jet will be decelerated, primarily due to the form drag as the jet flows across the neighboring tubes. To estimate the deceleration of the jet, we consider a control volume around a tube. The jet flow rate coming into the control volume from the control surface (A_1) in front of the tube is $\rho A_1 U_1$ and the momentum flux is $\rho A_1 U_1^2$ where A_1 equals the projected area of the tube, ρ is the density of the gas, and U_1 is the velocity of the jet before hitting the tube. The flow rate leaving the control surface (A_2) is $\rho A_2 U_2$ and the momentum flux is $\rho A_2 U_2^2$ where U_2 is the velocity of the jet after passing the tube and before hitting the next tube. The drag force in the control volume (from the tube) is represented by, $1/2 C_D \rho A_1 U_1^2$ where C_D is the drag coefficient. According to reference [G-8], the drag coefficient in the flow region where the cylinder diameter-based Reynolds number is greater than 100 is about unity.

The mass and the momentum balance equations are:

$$\rho A_1 U_1 = \rho A_2 U_2 \tag{2}$$

$$\rho A_1 U_1^2 - \frac{1}{2} \rho A_1 U_1^2 = \rho A_2 U_2^2 \tag{3}$$

so

$$U_2 = \frac{1}{2} U_1, A_2 = 2A_1 \tag{4}$$

Equation (4) indicates that a jet will lose about half of its velocity and will double its flow area as it flows across a tube (due to the form drag of the tube). If we assume that the jet will lose half of its remaining velocity as it flows across each tube, after passing *n* tubes the velocity of the jet in terms of its initial velocity is then expressed by

$$U_{n+1} = \frac{1}{(2)^n} U_1, \mathrm{n} = 1,2,3,...... \qquad (5)$$

In determining initial particle size, three different approaches were considered. The first is discussed in reference [G-4] and is based on NRC sponsored work at Sandia [G-9]. This work assumed a log normal distribution in size with uncertain mean and standard deviation. The mass mean aerosol size was taken to be uniformly distributed over a range of 1.5 to 5.5 μm. The geometric standard deviation was taken to be uniformly distributed over a range of 1.6 to 3.7. The second approach is based on detailed Polestar calculations of aerosol particle agglomeration and resulting particle size in a containment. This is not an unreasonable approximation for initial particle size for the unisolated SGTR problem since the much larger containment volume has compensating effects of lower particle concentration and longer residence time compared to the hot leg/steam generator inlet plenum volume. The mass mean diameter is 1.92 μm and standard deviation is 2.2. The third approach used just the minimum mass mean diameter and standard deviation from the Sandia work (i.e., 1.5 μm and 1.6).

Correlations exist for the collection efficiency of inertial deposition on a cylinder in terms of the Stokes number where collection efficiency is defined as the ratio of the aerosol mass deposited on the tube per unit time to the aerosol mass that would flow across the projected area of the tube per unit time if the tube were not present. Fuchs [G-10] reviewed a number of such correlations and associated data which are in reasonable agreement for Stokes numbers above a critical value of 0.125 up to about 10. The correlation used here is that of Landahl which generally yields the lowest (most conservative) efficiencies of the three correlations presented in reference [G-10]:

$$\varepsilon = Stk^3 / (Stk^3 + 1.54 Stk^2 + 1.76) \qquad (6)$$

where ε is retention efficiency and Stk (Stokes number) is defined as follows:

$$Stk = \frac{\rho_p d_p^2 U Cu}{18 \mu D} \qquad (7)$$

where ρ_p and d_p are, respectively the density and diameter of the particle, μ is the viscosity of the gas and Cu is the Cunningham slip factor, U is the jet velocity, and D is the outer diameter of the tube.

In applying equation (6) to the unisolated SGTR problem, not only the velocity as discussed above but also the particle size distribution will change as the gas and particle flow stream pass over each successive row of tubes. This is accounted for in the quantification of inertial impaction DF. The particle size spectrum was divided into 6 bins. The initial size distributions are given below for the three different approaches discussed above.

	Probability of Particle in Size Bin		
Particle Size Bin	**Sandia (mass mean range 1.5 – 5.5 μm , 1.6< σ <3.2)**	**Polestar (mass mean 1.92 μm , σ =2.2)**	**Sandia (mass mean 1.5 μm , σ =1.6)**
< 0.5 μm	0.04	0.05	0.03
0.5 – 1.5 μm	0.17	0.32	0.47
1.5 – 2.5 μm	0.17	0.26	0.36
2.5 – 3.5 μm	0.13	0.14	0.1
3.5 – 4.5 μm	0.09	0.09	0.04
> 4.5 μm	0.4	0.14	0.0

The DF was quantified for each initial size distribution, keeping track of the decreasing probabilities of larger particle sizes and increasing probabilities of smaller particles for successive rows of tubes. For example, for the Polestar initial distribution, the probabilities changed for successive rows of tubes as follows:

	Probability of Particle in Size Bin					
Particle Size Bin	**Initial**	**2nd Row**	**3rd Row**	**4th Row**	**5th Row**	**6th Row**
< 0.5 μm	0.05	0.09	0.10	0.11	0.11	0.11
0.5 – 1.5 μm	0.32	0.53	0.60	0.62	0.62	0.62
1.5 – 2.5 μm	0.26	0.25	0.24	0.23	0.23	0.23
2.5 – 3.5 μm	0.14	0.07	0.04	0.03	0.03	0.03
3.5 – 4.5 μm	0.09	0.03	0.01	0.01	0.01	0.01
> 4.5 μm	0.14	0.03	0.01	0.00	0.00	0.00

The *DF* for each of the three initial size distributions was 33, 17, and 4.3. The smaller DF for the Sandia mass mean 1.92 μm case is due to the fact that a larger fraction of particles are in smaller size bins.

Eddy Diffusion-Driven Turbulent Deposition for External Flow (Secondary Side)

The phenomena of eddy diffusion-driven turbulent aerosol deposition depends upon the turbulence generated in high velocity flows causing the aerosol particles to diffuse to the surface of the collector. The deposition occurs in the boundary layer or other region where turbulence exists. The aerosol collection efficiency depends upon the eddy diffusivity of the flow where eddy diffusivity is the product of turbulent velocity and length scale. The turbulent velocity in turn is the product of turbulence level and free stream velocity. In the case of the flow exiting the break in the unisolated SGTR, the turbulence level is large due to the free shear flow resulting from the jet and the wake from flow over the cylinder. The length scale in this problem is the tube diameter.

To estimate the DF from this phenomenon, the results of reference [G-11] were used. Reference [G-11] made measurements of and developed a theoretical model for predicting eddy diffusion-driven aerosol deposition on cylinders. The collection efficiency is governed by the Stokes number as shown in Figure G-3 (reproduced from Figure 9 of reference [G-11]). The reference [G-11] model was found to agree very well with the experimental results. The collection efficiency of aerosol particles on a single cylinder was found to be 0.4 for Stokes number of 0.04, decreasing down to a few percent at Stokes number of 0.002. Although reference [G-11] states that the true collection efficiency is represented by the upper boundary of the data envelope rather than the customary median curve, a curve fit through the data is used in this work, giving collection efficiencies somewhat lower than suggested reference [G-11].

To extrapolate the reference [G-11] results to the unisolated SGTR problem, Figure G-4 (reproduced from Figure 10 of reference [G-11]) is applied. According to Figure G-4, for $Sc\,Re^{1/2}$ < ~0.1, collection efficiency is ~0.4 where

$$
\begin{aligned}
&Sc = \nu / \varepsilon \\
&Re = D_c U / \nu \qquad\qquad (8)\\
&Sc\,Re^{1/2} = (D_c U \nu)^{1/2} / \varepsilon
\end{aligned}
$$

where ν is gas kinematic viscosity, ε is eddy diffusivity, D_c is collector diameter, and U is gas velocity. A comparison of the experimental conditions described in reference [G-11] with the unisolated SGTR problem yields the following:

	Experiment	**SGTR Problem**
Collector diameter (m)	0.0016	0.015
Gas velocity (m/s)	11	500
Kinematic viscosity (m^2/s)	1.5E-5	1.5E-4

Thus in order to maintain a $Sc\,Re^{1/2}$ < ~0.1, the eddy diffusivity in the SGTR problem must be larger by about a factor of 50 over the experiment. Since the eddy diffusivity in the experiment is of the order of 25 cm^2/s, this is about 0.1 m^2/s.

To estimate the eddy diffusivity for the SGTR problem, the results of reference [G-12] are used. Reference [G-12] is included as Attachment 1 to Appendix G. Reference [G-12] notes that turbulence levels of free shear flows are generally of the order of 0.2 to 0.4 far from the origin. Close to the origin of the flow, higher levels may be observed. With a turbulence level of 0.2 and a gas velocity of 500 m/s, the eddy diffusivity may be approximated as ~0.2*500*.01 = 1 m^2/s, well above the 0.1 m^2/s necessary to maintain $Sc\,Re^{1/2}$ < ~0.15. Thus it is a conservative extrapolation of the experiment to use a collection efficiency of 0.4 for each pass over a SG tube until $Sc\,Re^{1/2}$ drops below ~0.1. For $Sc\,Re^{1/2}$ > ~0.1, collection efficiency decreases according to the Figure G-4 curve with no collection assumed for $Sc\,Re^{1/2}$ > ~0.5.

Calculations of two SGTR scenarios were performed to estimate eddy diffusion-driven DF. The higher temperature scenario resulted in slightly higher velocities but lower DF due to higher gas viscosity which results in higher $Sc\,Re^{1/2}$. The spreadsheet result for this calculation is provided in Table G-1. The DF for this scenario is ~40. As noted above in the introductory section, this DF is essentially independent of particle size and so is not affected by any inertial impaction removal.

Thermophoretic Deposition in Cooler SG Secondary Side Structures

Thermophoresis causes a diffusive effect due to a temperature gradient in the gas flowing near a cooler surface. Here the gas molecules on the hotter side of the particle collide with it more frequently than those on the cooler side, leading to a net momentum transfer to the particle, which drives it towards the cooler side, i.e., the wall. The thermophoretic velocity is given by the Brock equation [G-13]:

$$v_{th} = \frac{2Cs \cdot Cn \cdot (\alpha + Ct \cdot Kn)(\mu / \rho)}{[1 + 2\alpha + 2Ct \cdot Kn][1 + 3Cm \cdot Kn]}\left(\frac{1}{T}\right)\frac{dT}{dy} \tag{9}$$

where

μ = viscosity of the containment atmosphere

ρ = density of the containment atmosphere

Kn = Knudsen number (= gas mean free path/particle radius)

$Cn(Kn)$ = Cunningham slip correction factor

α = ratio of gas to particle thermal conductivity

dT/dy = temperature gradient at surface (wall or condensate film)

Cs, *Ct* and *Cm* are coefficients with values from reference [G-13] of 1.147, 2.2, and 1.146, respectively.

It is evident that the thermophoretic velocity is proportional to the temperature gradient at the wall (and thus the sensible heat transfer between the gas and wall), and is somewhat dependent on particle size, tending to be higher (with higher DF) for smaller particles.

Per reference [G-4], heat transfer in the range of 0.5 to 1% of decay heat will give a DF from thermophoresis of 2 to 3. Heat transfer of 3 to 5 % of decay heat will give a DF of 20 or higher. To provide a rough estimate the heat transfer for the unisolated SGTR problem, consider gas entering the tube bundle at or near the tube sheet. The gas will transfer heat to the metal in the tubes and surrounding steam generator structure as it flows up the tube bundle. This may be approximated as

$$q = hA\Delta Tx10^{-6}$$

where q is heat transfer rate in MW, h is the convective heat transfer coefficient in $W/m^2/K$, A is the heat transfer area in meters, and ΔT is the gas to metal (tube wall) temperature difference.

For a typical PWR steam generator, the tube bundle height is about 12 m, and the perimeter of a cross section of the tubes is about 690 m giving an area of ~8000 m^2. From reference [G-14], the lower estimate of h for forced convection for superheated steam is about 25 W/m^2K. The gas – metal temperature difference to maintain heat transfer of about 1 % of decay heat (i.e., 0.2 MWt in the case of AP600) may be estimated as

$$\Delta T = \frac{0.2}{(25)(8000)} x10^6$$
$$\approx 1K$$

Thus an average temperature difference between the gas and metal of only 1 K will maintain enough heat transfer to provide a thermophoresis DF of 2 to 3.

To estimate the expected gas – metal temperature difference, a spontaneous SGTR for AP600 is used as an example. The gas temperature during the period of fission product aerosol release (a period of ~2 hours) ranges from about 500 K to 1000 K [G-2], an average of ~750 K. The tube bundle temperature at the time of the beginning of fission product aerosol release is predicted to be approximately 500 K [G-2]. While the metal will heat up (and the gas will cool) as the hot gas flows over the tube bundle and aerosol fission products deposit on the metal surface, the heat capacity of the metal is large enough that the metal will not undergo significant temperature increase during the 2 hour period. This may be seen by estimating the heat capacity of the metal as

$$Q = mc_p$$

where Q is heat capacity in W.s per degree K, m is metal mass, and c_p is metal specific heat in joule/kg/K. For steel, c_p = 670 joule/Kg/K. The mass of the tube bundle and the surrounding metal of the steam generator lower shell varies depending upon the plant design, but can be conservatively estimated as about 150,000 kg. Thus

$$Q = (150000)(670)$$

= 100 MWs/K

The heat transfer to the metal is mainly from deposited aerosol fission products, by far the most important of which is iodine. This heat transfer may be estimated as follows. Assuming 50% release fraction of iodine from the core and ~50% retention in the RCS, there will be ~25% core fraction of iodine remaining to deposit in the steam generator due to the three aerosol retention mechanisms discussed above. Assuming iodine is roughly 25% of total decay heat, for AP600 (total decay heat of 20 MW) the heat energy into the steam generator from fission product deposition is

(0.25)(0.25)(20) ≈ 1 MW

The 1 MW heat transfer for 2 hours will increase the tube and surrounding lower shell metal temperature by (1)(7200)/100 ≈ 70 K. If the metal in the upper shell (housing the moisture separator and dryer assembly) is considered, roughly doubling the metal mass, this temperature increase would be of the order of 35 K. This compares to an initial gas – metal temperature difference of ~250 K.

To confirm that a heat transfer rate of ~0.2 MW (i.e., 1% of decay heat in the case of AP600) or greater is maintained during the ~2 hour period of fission product aerosol release, a time varying, one-dimensional heat transfer problem was solved by numerical integration of coupled partial differential equations describing metal and gas temperature. Reference [G-15] describes this problem. The equations are as follows:

$$\frac{\partial T_g}{\partial t} = -v_g \frac{\partial T_g}{\partial x} + \left(\frac{hP}{c_{pg}\rho_g A_s}\right)(T_m - T_g)$$

and

$$c_{pm}\rho_m A_c \frac{\partial T_m}{\partial t} = k_m A_c \frac{\partial^2 T_m}{\partial x^2} - hP(T_m - T_g)$$

where:

T_g = gas temperature (K)

T_m = tube metal temperature (K)

v_g = gas velocity (m/s)

A_c = metal cross sectional area (m^2) = $N2\pi rt$, N = number of tubes, r = inside radius of a tube (m), t = tube wall thickness (m)

A_s = flow area (m^2) = steam generator area minus $N\pi r^2$

P = total perimeter of the tubes = $N2\pi r$ (m)

ρ_m = metal density in the steam generator (k/m^3)

ρ_g = gas density (k/m^3)

c_{pm} = metal heat capacity (J/kg/K)

c_{pg} = gas heat capacity (J/kg/K)

k_m = metal thermal conductivity (W/m/K)

h = heat transfer coefficient ($W/m^2/K$)

The source of energy for the heat transfer is the stored energy in the hot gas (steam, hydrogen, noble gases) resulting from core melt which enters the steam generator, and the volumetric heat generation from noble gases flowing through the steam generator, a total energy in the range of 0.5 to 1 MW. The results indicate that a heat transfer rate above 0.2 MW is maintained for the roughly 2 hour duration of fission product aerosol release [G-15].

For conservatism, the quantification of overall SGTR DF uses a thermophoresis DF of 2.

Estimate of Overall DF

The table below summarizes the DF results for the four aerosol retention mechanisms considered in the unisolated SGTR problem. These DFs are essentially independent and give a total DF in excess of 1000. The overall DF used in this work is limited to 100 to allow for uncertainties as discussed below.

Mechanism	**DF**
Turbulent deposition inside broken tube	3*
Inertial impaction for flow over tubes	5
Eddy diffusion-driven deposition	40
Thermophoresis	2

*Applies to break location at or near top of tube sheet.

Discussion of Uncertainties

The model for turbulent deposition for flow inside a tube is reasonably well understood and accepted. Thus, given the location (elevation) of the tube rupture, there is not significant uncertainty in the DF for this removal mechanism. However, tube rupture location is uncertain. Based on the ten tube ruptures that have occurred [G-16], the locations can be classified into 3 groups:

Location	**Probability**	**DF**
Near tube sheet	0.3	~3
Several meters above tube sheet	0.2	~1000
Near top of tube bundle	0.5	>10,000

For purposes of this work, this effect will be treated by splitting SGTR sequences into two types: tube rupture location near the tubesheet, and tube rupture location at or above the tube bundle midpoint. For the former, the sequence probability in the PRA is multiplied by 0.3, and an aerosol DF of 100 is used per the discussion above in Estimate of Overall DF. For the latter, the sequence probability in the PRA is multiplied by 0.7, and the aerosol is assumed to be essentially completely removed based on the DF of 1000 or more from turbulent deposition for flow inside a tube plus the removal from the other mechanisms.

The main uncertainties in the treatment of inertial impaction are particle size and the effect of particle bounce. To address particle size, the smallest mass mean from the Sandia work [G-9] was used. This is also conservative based on calculations of particle growth by Polestar. Particle bounce is discussed in references [G-17] and [G-18]. While bounce of solid particles has been measured as velocity increases, no liquid particle bounce has been measured. The aerosol particles in the SGTR problem are likely to be liquid or at least a mixture of liquid and solid depending upon temperature of the gas which would mitigate any bounce effects. Also, even if some bounce occurs, the large size of the SG tube array would retain particles for many bounces. This is evident if one considers that it requires of the order of a hundred seconds or more for the gas to exit the tube bundle (a few cm/s traveling ~10 meters). At 100 m/s velocity and 1 cm tube spacing, there would be $>10^4$ bounces before the particle escapes. Thus even a small collection efficiency per impact would be expected to result in significant overall collection efficiency.

The eddy diffusion-driven turbulent deposition involves a complex flow field which is not easily modeled. The treatment here is conservative with regard to extrapolation of the experimental data. Nonetheless, the modeling complexities and limited experimental data base introduce uncertainty in the estimate of DF. This uncertainty is addressed by the fact that the four mechanisms considered here are essentially independent together with the use of a limited overall DF in this work and performing sensitivity studies to provide confidence that the final results are not overly sensitive to this DF. It is also noted that plans are proceeding to perform an experiment on an actual SG tube bundle to measure this aerosol deposition effect [G-19].

The thermophoretic deposition has uncertainty in the gas and SG metal temperatures and resulting heat transfer rate. However, as detailed above, the initial gas – metal temperature difference, the metal heat transfer area, and the metal heat capacity are expected to combine to provide a heat transfer rate above the ~0.2 MWt for 2 hours which is required for a thermophoresis DF of 2 or greater.

Resuspension of deposited aerosols is not considered to be a significant factor in this work since it is likely that the resuspended material will no longer be micron-sized particles but rather agglomerates that are large enough to settle rapidly once they pass to the secondary side of the steam generator. Further, since the fission product aerosol will be mainly liquid at the high temperatures involved in the secondary side retention problem (600K to 1200K), it is expected that there will be marked adhesion between the deposited aerosol and the tube surface, thus significantly reducing particle resuspension in the gas flow [G-20].

Revaporization could eventually occur in the steam generator tube region as fission products deposit and the metal mass heats up. To assess this effect, scoping heat transfer calculations have been performed for the AP600 SGTR sequence. Under dryout, adiabatic conditions and considering only radiation heat transfer, for AP600 it would take a time duration of the order of an hour to reach metal temperatures at which significant revaporization could occur. This time duration occurs because of the heat capacity of the steam generator metal mass and the tendency of the fission product heat energy to spread out over this mass (due in turn to the aerosol deposition over multiple tubes, the distribution of the gamma and beta decay energy over a large volume inside the steam generator, and the large surface area of the tubes which increases heat transfer). This time duration provides significant opportunity (in addition to the time leading up to the beginning of fission product release) for accident management action to inject into the steam generator. A few tens of gallons per minute will provide enough cooling to prevent significant metal heatup. Thus, long term retention of the fission product aerosol does not depend upon maintaining any particular water level in the secondary side but rather only on a small injection rate. Appendix H discusses accident management programs for the ALWR. An additional point is that even if some revaporization occurs, the fission product vapors will flow through a large, relatively cool heat transfer surface in the upper portion of the steam generator (i.e., separators and dryers) which will tend to promote condensation on the surface and thus removal of the vapors.

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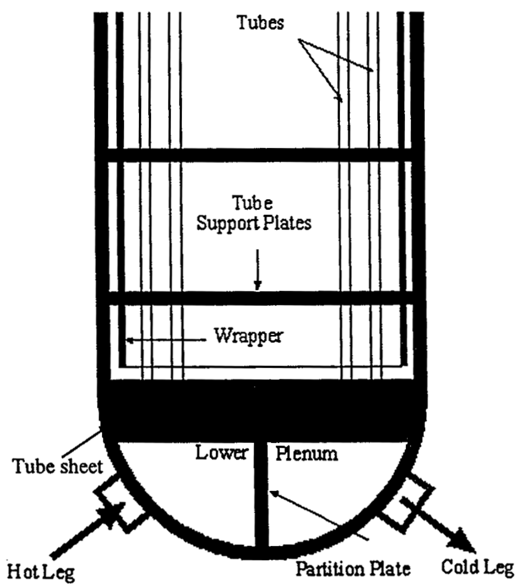

Figure G-1
Schematic of the Steam Generator

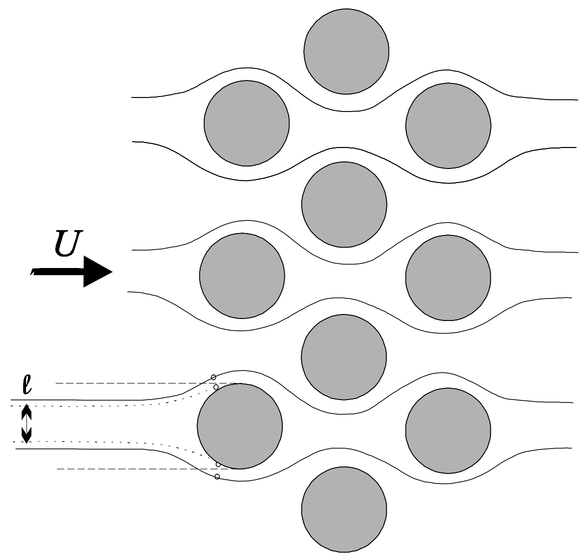

Figure G-2
Schematic of the Flow Across a Bundle of Tubes

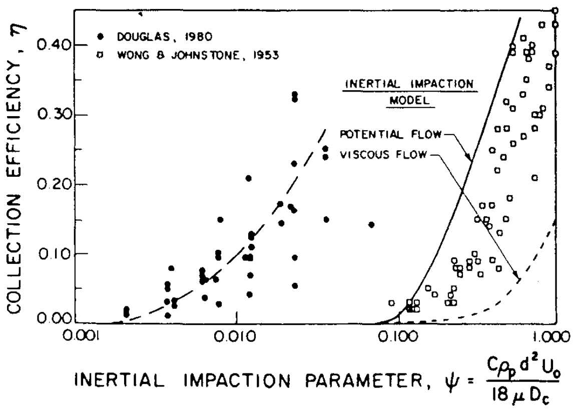

Figure G-3
Data on Particle Collection Efficiency for Cross Flow Over a Single Cylinder vs. Stokes No. (taken from reference [G-11])

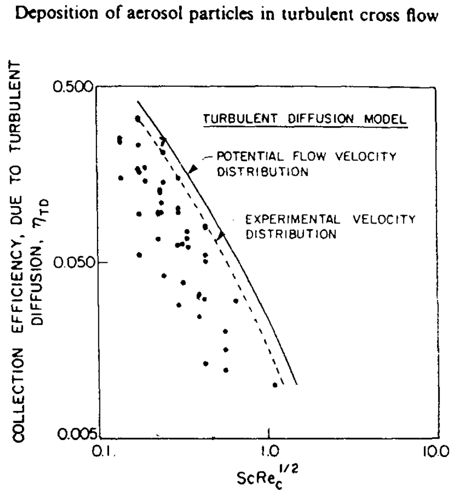

Figure G-4
Comparison of Measured Collection Efficiency as a Function of $ScRe^{1/2}$ with Model Prediction (taken from reference [G-11])

Table G-1
Collection Efficiency for Successive Tube Rows from Eddy Diffusion-Driven Turbulent Deposition

Turb.Level	0.2				
Init. Eddy Diff. (m^2/s)	2.667				
Gas temp(K)	950				
Visc steam (kg/m/s)	3.38E-05	Dens steam (kg/m^3)	0.23	Kin Visc (m^2/s)	1.46E-04

Tube Row	Dc	U	Re_C	Eddy Diff.	Sc	$ScRe_C^{1/2}$	Collection Efficiency
1	0.01905	700	9.11E+04	2.667	5.5E-05	0.017	0.4
2	0.01905	350	4.55E+04	1.3335	0.00011	0.023	0.4
3	0.01905	175	2.28E+04	0.66675	0.00022	0.033	0.4
4	0.01905	80	1.04E+04	0.3048	0.00048	0.049	0.4
5	0.01905	40	5.20E+03	0.1524	0.00096	0.069	0.4
6	0.01905	20	2.60E+03	0.0762	0.00192	0.098	0.4
7	0.01905	10	1.30E+03	0.0381	0.00384	0.139	0.3
8	0.01905	5	6.50E+02	0.01905	0.00769	0.196	0.13
9	0.01905	2	2.60E+02	0.00762	0.01922	0.310	0.07
10	0.01905	1	1.30E+02	0.00381	0.03843	0.438	0.038

Attachment 1
Letter Report on Turbulence Levels of Free Shear Flows
(Reference [G-12] of Appendix G)

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May 11, 1999

To: Dave Leaver, Rudy Sher
Polestar Applied Technology, Inc.
One First Street, Suite 4
Los Altos, CA 94022

fax: 650-948-8244

Dear Dave and Rudy,

In our recent meetings at Polestar we discussed the paper by Douglas and Ilias on the deposition of aerosol particles on cylinders in crossflow. The subject of turbulence levels came up several times and the purpose of this letter is to describe, in general terms, typical turbulence levels observed in free shear flows.

Consider a turbulent flow with stationary statistics. Let the velocity field be decomposed into a mean and fluctuating part

$$\mathbf{u}(x, y, z, t) = \mathbf{U}(x, y, z) + \mathbf{u}'(x, y, z, t)$$

The turbulence level is generally constructed from measurements of the streamwise velocity component at some point in the flow. First the time mean is determined from an average over the velocity record. Then the root-mean-square of the fluctuation away from the mean is determined. The result is the turbulence level, generally expressed simply as u'/U. The turbulence level is a function of position, usually taking its maximum value in the central region of the flow and decaying to zero at the edges. This is not the only way to define the turbulence level and if data for more than one velocity component is available then one can form the numerator from the rms of the turbulent kinetic energy.

It is important to distinguish between turbulence in free shear flows and turbulence near a wall typified by the turbulent boundary layer. The maximum turbulence level in a boundary layer occurs quite close to the wall and is on the order of u'/U = 0.1 where U is the free stream speed.

Free shear flows come in many different forms including jets, wakes, buoyant plumes and mixing layers. The turbulence levels vary from flow to flow but generally speaking are on the order of u'/U = 0.2 to 0.4 depending on the particular flow situation. These numbers apply far from the origin of the flow where the velocity field is regarded to be fully developed. Close to the origin of the flow higher levels may be observed. The choice of normalizing mean velocity is not always clear. In a turbulent jet one would use the mean velocity on the jet centerline, in a mixing layer the appropriate velocity scale would be the velocity difference between the two streams, in a wake the defect velocity on the centerline would be used. In general, the appropriate velocity scale for normalizing the turbulence intensity is the overall velocity difference which drives the flow.

Since the main object of interest here is the turbulence in the wake of cylinders I have enclosed a series of pictures from Van Dyke's *Album of Fluid Motion.* These pictures depict flow past a circular cylinder at several Reynolds numbers ranging from very low values where the flow is laminar and steady up to Reynolds number 10,000 where the wake is highly turbulent. The Reynolds number 2,000 and 10,000 figures provide a qualitative view of the flow close to the cylinder where measured turbulence levels are quite a bit higher than the values quoted above. The enclosed paper by Cantwell and Coles adds quantitative information to this picture at a Reynolds number of 140,000.

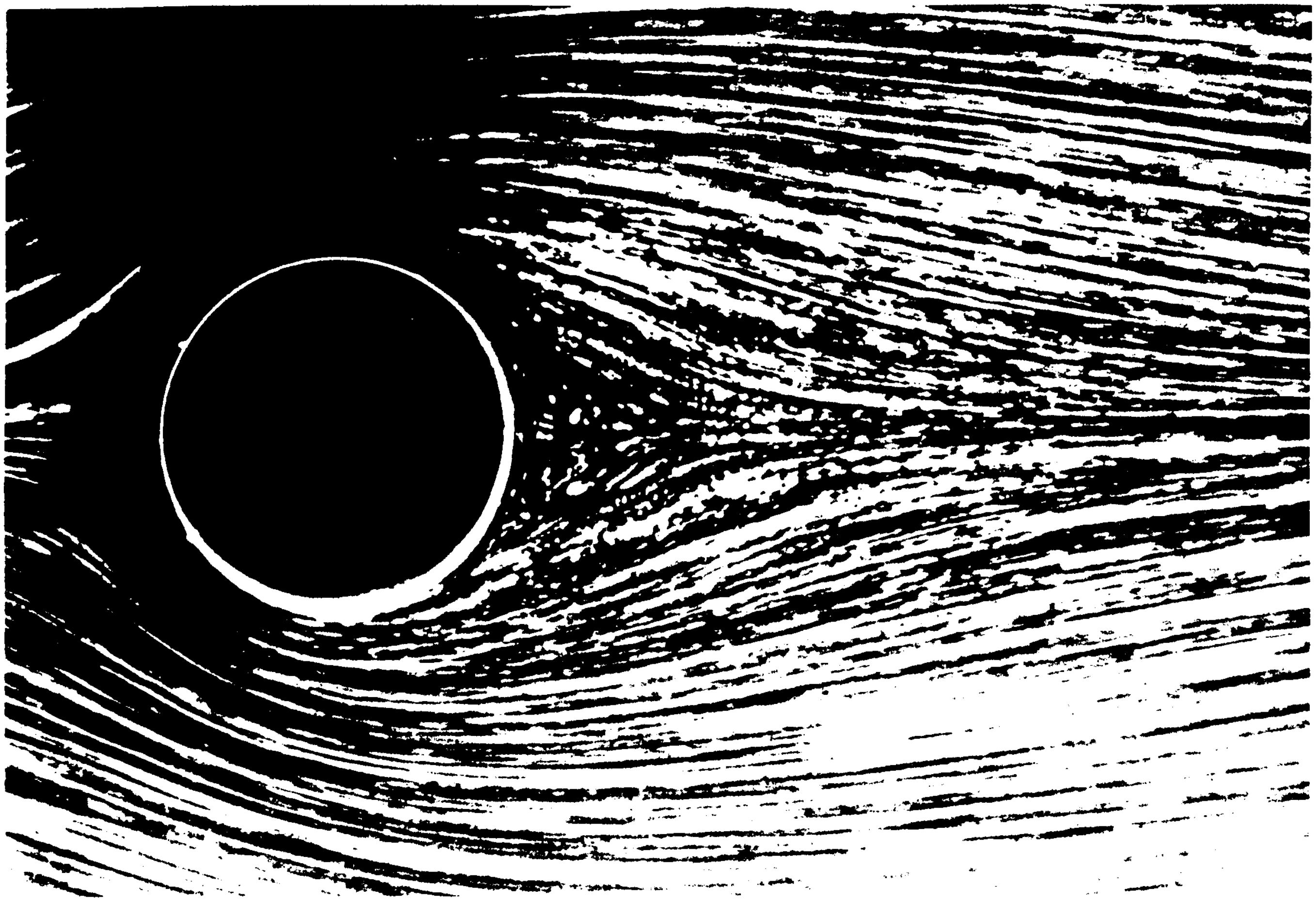
In the experiments described in this paper the fluctuating velocity field was decomposed into three parts; a time mean plus a mean at constant phase of the vortex shedding cycle plus a random fluctuation. Figure 25 shows the data for the periodic and random contribution to the cross stream Reynolds normal stress. At two diameters downstream of the center of the cylinder the periodic part of the stress is at a level of 0.37. At the same position the random component is about 0.13. Adding these two numbers and taking the square root indicates that, at this position in the wake, the cross-stream velocity fluctuations are 70% of the free stream velocity - on average! Toward the end of the measurement domain at 6 diameters from the cylinder center this number has dropped to about 20% of the free-stream speed. But note that when this fluctuation is normalized by the velocity defect at the same position (see figure 11a) the turbulence level, as it would be usually quoted, is approximately one. Unfortunately quantitative data like this is not available at other Reynolds numbers however it is reasonable to expect high turbulence levels in cylinder wakes at Reynolds numbers down to the range depicted in the enclosed photographs.

The complete data set from this experiment is available at my website.

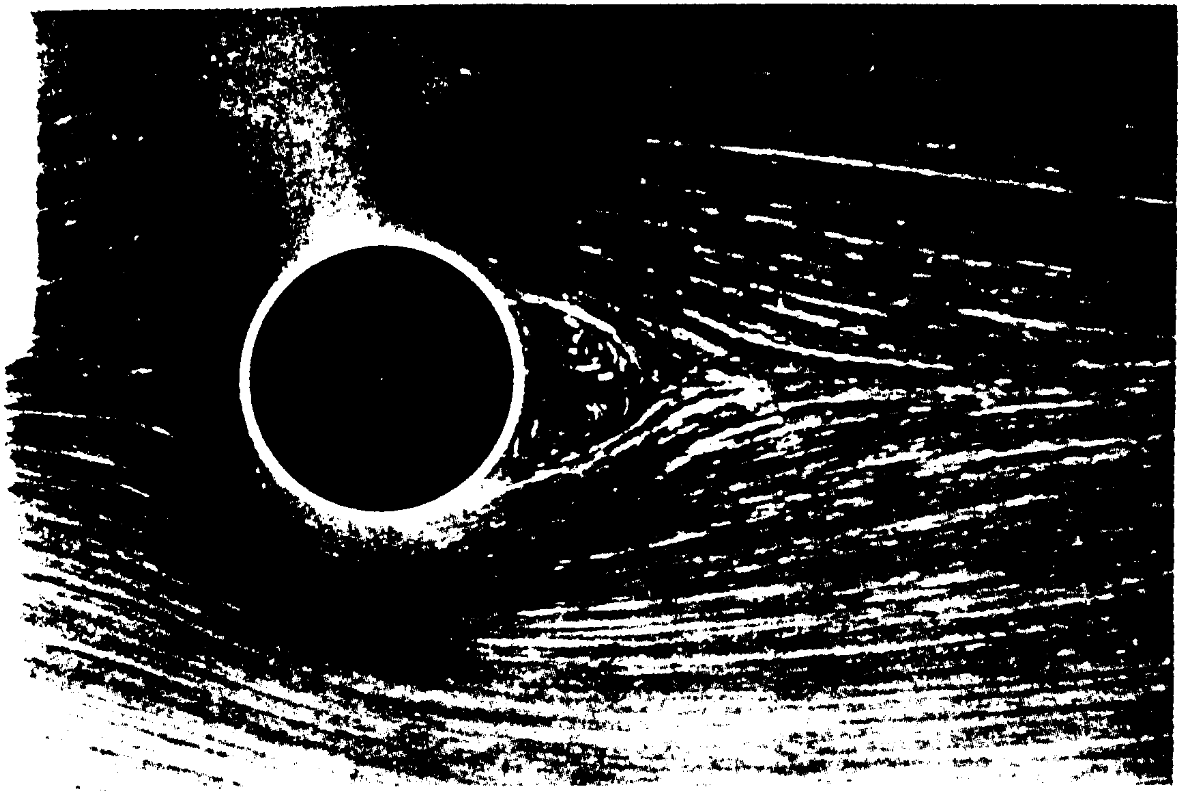
http://thomasc.stanford.edu/~cantwell

Best regards,

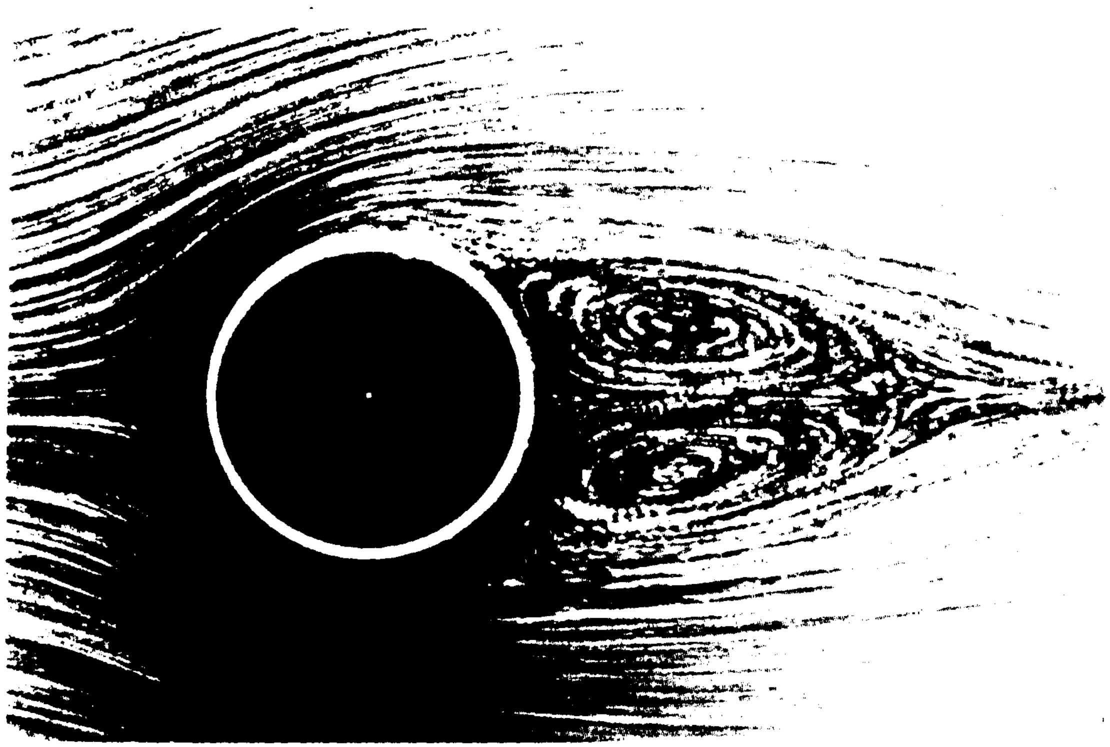
Brian Cantwell



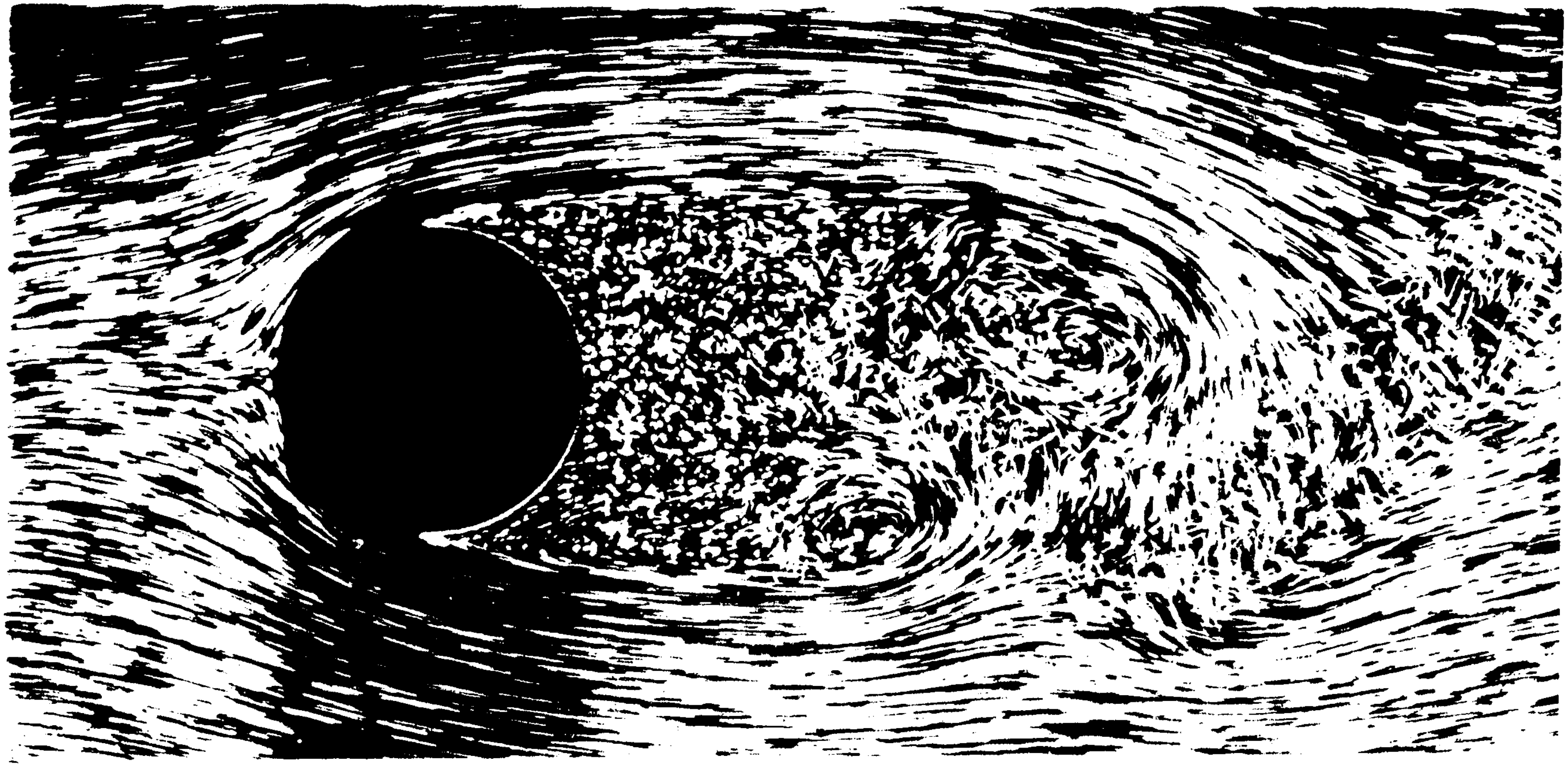
40. Circular cylinder at $R=9.6$. Here, in contrast to figure 24, the flow has clearly separated to form a pair of recirculating eddies. The cylinder is moving through a tank of water containing aluminum powder, and is illuminated by a sheet of light below the free surface. Extrapolation of such experiments to unbounded flow suggests separation at $R=4$ or 5, whereas most numerical computations give $R=5$ to 7. *Photograph by Sadatoshi Taneda*



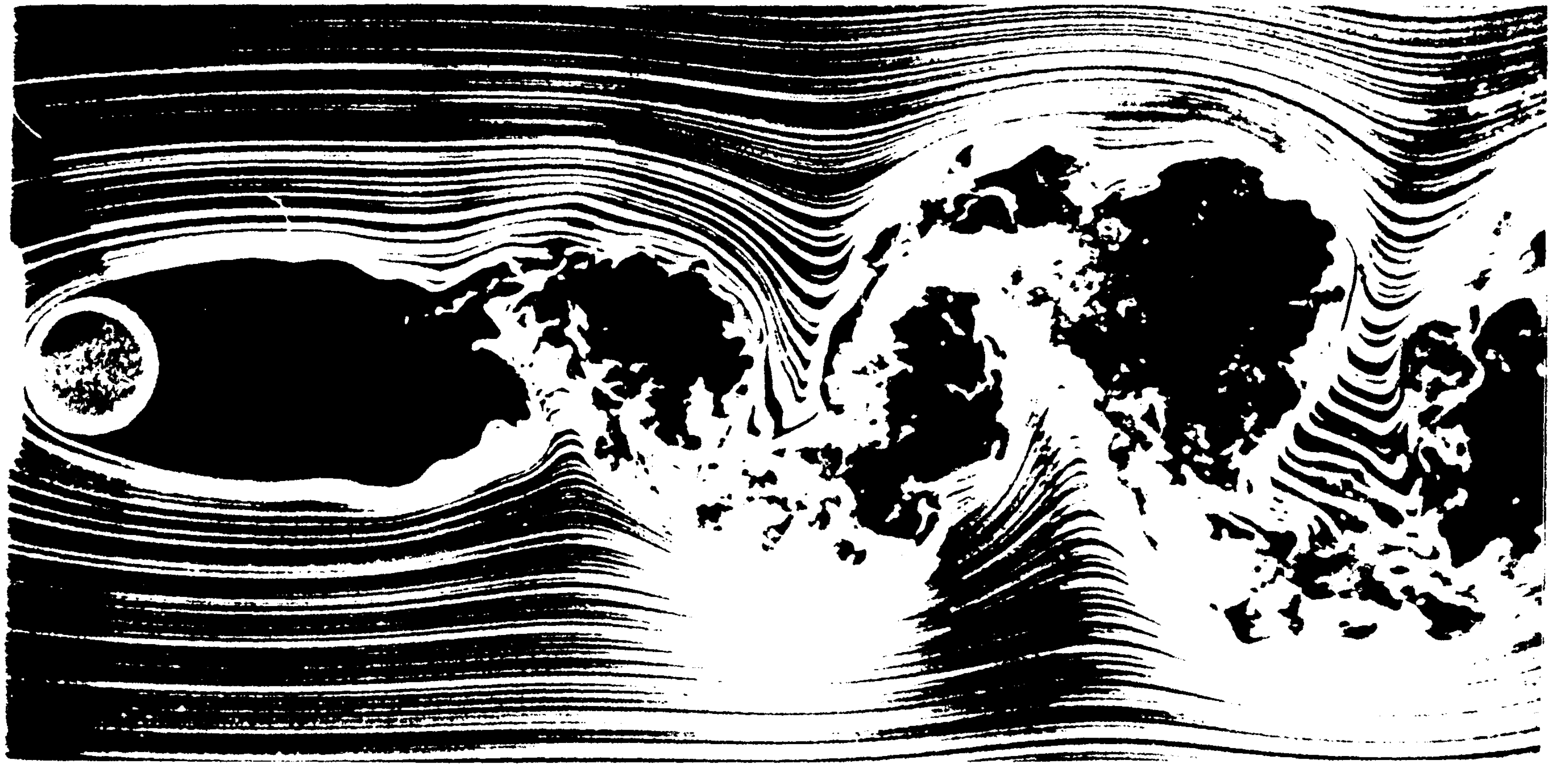
41. Circular cylinder at R=13.1. The standing eddies become elongated in the flow direction as the speed increases. Their length is found to increase linearly with Reynolds number until the flow becomes unstable above R=40. *Taneda* 1956a



42. Circular cylinder at ***R*** **= 26.** The downstream distance to the cores of the eddies also increases linearly with Reynolds number. However, the lateral distance between the cores appears to grow more nearly as the square root. *Photograph by Sadatoshi Taneda*



47. **Circular cylinder at** $R = 2000$**.** At this Reynolds number one may properly speak of a boundary layer. It is laminar over the front, separates, and breaks up into a turbulent wake. The separation points, moving forward as the Reynolds number is increased, have now attained their upstream limit, ahead of maximum thickness. Visualization is by air bubbles in water. *ONERA photograph, Werle & Gallon 1972*



48. Circular cylinder at $R=10{,}000$. At five times the speed of the photograph at the top of the page, the flow pattern is scarcely changed. The drag coefficient consequently remains almost constant in the range of Reynolds number spanned by these two photographs. It drops later when, as in figure 57, the boundary layer becomes turbulent at separation. *Photograph by Thomas Corke and Hassan Nagib*

H
INDUSTRY RESPONSE TO PEER REVIEW COMMITTEE RECOMMENDATIONS

Industry Response to Peer Review Committee Recommendations

During the first two quarters of 1999, a peer review was performed on a draft version of the ALWR emergency planning technical report. Two meetings were held at EPRI offices in Washington, D.C., one on February 2, 1999 and one on March 17, 1999. In attendance were the four peer reviewers and EPRI and Polestar personnel.

Attachment 1 is the report produced by the peer review committee. Below are the peer review recommendations and the industry responses to the recommendations.

Peer Review Recommendation "Presentation of results should be made without truncation to avoid potential masking of risk significant low probability events."

Response. Per the discussion in the peer review report section on tempering, the peer review group "strongly endorses the concept of a screening level", but believes this should be no more than three orders of magnitude below the core melt frequency based on the range of frequency presented in NUREG-0396, Figure I-11. The truncation approach was revised to include all release categories down to three orders of magnitude in frequency below the average ALWR core damage frequency. Section 4.2.4 and 4.3 discuss the methodology and the release categories included in the dose exceedance analysis for the NUREG-0396 assessment. The release categories themselves are identified in Tables 4-1 to 4-3. Figures 4-3 and 4-4 provide the dose exceedance results.

Peer Review Recommendation "All accident sequences of importance to emergency planning considerations, including those resulting from internal events, fires, and low power operation or shutdown events should be included in the analysis."

Response The ALWR Utility Requirements Document (URD) specified a very complete set of design requirements to minimize risk from shutdown and low power operation and external events. Reference [3] from the Main Report tabulated and evaluated the URD requirements for low power and shutdown, as did reference [9] for containment performance aspects of external events.

All three ALWR PRAs addressed risk from low power and shutdown events. Generally, the risk from these events was found to be minimal. The ABWR PRA, for example, stated that the probability of core damage during shutdown periods is negligible and concluded that no modifications to the ABWR plant design and no detailed PRA assessment were required. The AP600 PRA did perform a Level 2 analysis of low power and shutdown risk. This analysis indicates that early releases are extremely low in probability such that they do not significantly impact the internal events early release results.

All three ALWR PRAs also addressed risk from external events. No Level 2 analysis was performed. The results that were reported were generally scoping-type, Level 1 results. The System 80+ PRA indicated, for example, that fire and flood core damage frequency results were a conservative scoping estimate only. The AP600 PRA did not take credit for non-safety systems in its quantification of core damage from internal fires and generally indicated that the core damage frequency estimates for fires and floods should not be added to that for internal events since the former were so conservative that the different analyses are not comparable. For these reasons and the fact that the original NUREG-0396 assessment did not include external events, the ALWR assessment against NUREG-0396 did not include external event sequences.

<u>Peer Review Recommendation</u>. "Additional justification and defense of the decontamination factors utilized in the containment bypass PWR steam generator tube rupture evaluation is needed."

<u>Response.</u> Work was done to provide additional justification and defense of the steam generator tube rupture (SGTR) aerosol retention decontamination factors (DFs). Four retention mechanisms are considered: turbulent deposition inside the tube, inertial impaction for near field (i.e., near the break) external flow over the tubes on the secondary side, eddy-diffusion driven deposition for near field external flow over the tubes on the secondary side, and thermophoretic deposition for far field flow over cooler steam generator secondary side structures. These four retention mechanisms are independent, as are their respective DFs. Appendix G was prepared to document the DFs in detail. Each of the four mechanisms is quantified in Appendix G along with a discussion of uncertainties.

In addition, a sensitivity study was prepared (see Section 4.4.3 and Figure 4-5) which quantifies the effect of a reduction in SGTR DF on the NUREG-0396 assessment dose exceedance curves.

<u>Peer Review Recommendation</u>. "Discussion of the ingestion pathway evaluation should be revised to more clearly show methodology used is consistent with NUREG-0396."

<u>Response.</u> Appendix E was revised to clearly show the consistency of the ingestion path methodology with that from NUREG-0396. The last several paragraphs of Section E.4 of Appendix E provide this discussion.

<u>Peer Review Recommendation</u> "Additional discussion of new accident management guidelines and their implementation is needed as these are important to the decontamination factor assumptions."

<u>Response.</u> A more detailed discussion of accident management guidelines and their implementation was prepared. Appendix F documents this discussion, including addressing operating plant accident management implementation (as a model for what would be done for

ALWR) and ALWR requirements and expectations for accident management guidelines and implementation.

Peer Review Recommendation. "Sequences can be eliminated from the analysis on the basis of extended time to release of radioactive material. However, the minimum delay time used for elimination of sequences should not be less than 24 hours."

Response. The minimum delay time for elimination of accident sequences was increased to 24 hours. This is discussed in Section 4.2.5 and Tables 4-1 to 4-3.

Attachment 1

Peer Review Committee Report

Report Of The Independent Peer Review Group

Of

Advanced Light Water Reactor Emergency Planning Technical Report

July 1999

Charles Jackson

Charles Ader

Richard Denning

Tom Murley

Table of Contents

Executive Summary

An independent review group was formed to review the draft Electric Power Research Institute report EPRI TR-113509, November 1998 "Technical Aspects of ALWR Emergency Planning". The four member group conducted its review during the first quarter of 1999, including two meetings with EPRI and the report authors from Polestar Applied Technology, Inc.

The report updates the information presented in NUREG — 0396, December 1978, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" reflecting improved knowledge of radioactive source terms, improved techniques in performing probabilistic safety assessments, and improved plant designs. NUREG — 0396 provides a technical basis for a 10 mile emergency planning zone (plume exposure) and a 50 mile emergency planning zone (ingestion) for current reactor designs. Using essentially the same approach for the Advanced Light Water Reactor (ALWR) designs and improved analysis techniques, the report develops a technical basis for boundaries proposed to be 0.5 miles (now called a response area) and 25 miles (now called an awareness area) respectively.

The group examined the report methodology, key assumptions, and documentation relied upon. During the review several sensitivity analyses were requested to examine further the appropriateness of the various assumptions made in the report.

Several recommendations were made by the group to provide clarification and improve the technical arguments in the report.

The group concluded that significant improvements have been made in ALWR plant designs through the EPRI Utility Requirements Document as they impact severe accidents in nuclear power plants. As a result, the consequences and probability of occurrence (the product of which is known as "risk") of severe accidents is significantly reduced.

The peer review group concludes, that if due consideration is given to the recommendations contained in this review report, that EPRI TR —113509 can provide a reasonable revised technical basis to support decisions on Emergency Planning for ALWRs.

Introduction

In January 1999, the Electric Power Research Institute convened a small group to conduct an independent peer review of an Advanced Light Water Reactor Emergency Planning Technical Report. Since the current basis for emergency planning, NUREG — 0396, was published in the 1970s, significant improvements in new plant designs, and improvements in knowledge about severe accidents and behaviors of reactor cores (source term) following a severe accident have occurred. Accordingly, an updating of the technical basis is in order. EPRI commissioned such an updated technical basis and it is contained in EPRI TR — 113509.

The peer group was requested to conduct a technical review of EPRI TR — 113509, and to address the report methodology, completeness, treatment of uncertainties, reasonableness of the results, and any other matters which the group feels are significant.

The members of the peer review group are:

Chairman	Charles Jackson — Con Edison
	Charles Ader — NRC — RES*
	Richard Denning — Battelle
	Tom Murley — Consultant

The peer review group began its review of the technical report in January 1999, and had its first meeting with EPRI and the report authors on February 2, 1999. Numerous communications took place between the group and report authors after the first meeting to get clarifications of the report contents and assumptions. A second meeting was held on March 16, 1999 with EPRI and the report authors to obtain responses to questions raised and to discuss final conclusions of the review.

*Although Charles Ader participated in the peer review group as a member of the NRC staff, his agreement with the conclusions in the report reflect his personal views and is in no way intended to represent a regulatory endorsement of the methods and conclusions in the report.

Technical Report Methodology

Technical Basis for Existing Emergency Planning

The technical basis for existing emergency planning is contained in NUREG-0396. NUREG-0396 was published in 1978 by a joint NRC-EPA task force, which addressed a request for federal guidance on emergency planning from a conference of state radiation control directors. Four considerations were addressed in NUREG 0396 in determining the recommended emergency planning zone (EPZ). These considerations were later restated in NUREG 0654:

a. projected dose levels from the most severe design basis accident (DBA) should not exceed the protective action guide (PAG) levels outside the zone,
b. projected dose levels from less severe (i.e., "most") core melt accidents should not exceed the protective action guide (PAG) levels outside the zone,
c. for more severe core melt accidents, doses would generally not cause early injuries outside the zone, and
d. the planning which is performed should provide a substantial base for expansion of response efforts in the event this proved necessary.

In addressing these four considerations, the stated approach in NUREG-0396 was to base the rationale on a "full spectrum of accidents and corresponding consequences tempered by probability considerations." The probabilities and consequences of severe accidents, which were used in NUREG-0396, came primarily from WASH-1400. WASH-1400, published nearly 25 years ago, was the first LWR PRA performed in the U.S. and reflected the perspectives and state of knowledge on severe accidents which existed in the early-1970s. The WASH-1400 results were used in NUREG-0396 to generate curves of conditional probability of dose exceedance versus distance from the reactor (i.e., conditioned on the assumed occurrence of a core melt). These curves were generic in that they were for a combined PWR and BWR. Figure I-11 of NUREG-0396 (reproduced as Figure 4-1 in the report) shows these conditional probability of dose exceedance curves, and this figure was the main basis for the recommended 10 mile plume exposure planning distance in NUREG-0396.

Technical Basis for ALWR Emergency Planning

The foundation of the ALWR emergency planning technical basis is the core damage prevention and mitigation provisions of the EPRI Utility Requirements Document. In the development of a technical basis for ALWR emergency planning, each of the four NUREG-0396 considerations is addressed.

The methodology for ALWR evaluations against the NUREG-0396 considerations included the following:

- Plant design features are based on the ALWR designs, consistent with the EPRI Utility Requirements Document.
- The methods of analysis use improved source term methodology (i.e., updated based on the several decades of severe accident research and the severe accident management guidelines now in place at most plants).
- The probability of dose exceedance curves for the three ALWR designs are combined into a single, generic ALWR curve in a manner similar to NUREG-0396 in which PWR and BWR results are combined into a generic curve Figure I-11.
- The ALWR dose exceedance curve is conditional on core damage (assumes a probability of unity of core damage) in the same manner as NUREG-0396.
- Dose calculation assumptions (e.g., acute whole body, pathways and exposure times, straight line plume trajectory, shielding factors) are essentially the same as used in NUREG-0396, Figure I-11.
- ALWR dose exceedance curves are provided for 1 rem, 5 rem, 50 rem, and 200 rem in the same manner as NUREG-0396, Figure I-11.

One difference from NUREG-0396 is that the ALWR evaluations considered frequency truncation of accident sequences. Sequences with an estimated frequency below 10^{-7} per year are not included in the ALWR dose exceedance curve. A sensitivity study is included in the draft report in which a truncation level of 10^{-8} per year is evaluated. At the request of the peer review group, additional sensitivity studies with no frequency truncation were performed and presented to the peer review group.

Key Assumptions

Tempering

In NUREG-0396, a "full spectrum of accidents and corresponding consequences *tempered* by probability considerations" provides the technical basis for selecting emergency planning zones. The draft report EPRI TR-123456 interprets *tempering* to imply that accident sequences can be excluded from the analysis on the basis of their frequency and proposes to use a cutoff for event probabilities of 10^{-7} per year. That is, events with lower probabilities would be excluded from evaluation. Numerous bases are provided for such a number, including regulatory precedent. The peer review group took issue with this concept as presented, however, as events with lower probabilities might have significant consequences thereby masking a risk that should be evaluated. Because the exceedance curves are conditional on core damage, it is inconsistent and potentially misleading to eliminate sequences that affect the shape and magnitude of the curves over the range over which they are presented (three decades in NUREG-0396). EPRI/Polestar were asked to present the conditional probability of exceeding various dose levels with and without truncation. The table below presents the results:

	NUREG 0396 (10 miles)	ALWR (0.5 miles)	ALWR- no cutoff (0.5 miles)
Conditional Probability of Exceeding 1 Rem	**~0.3**	**~0.1**	**~0.25**
Conditional Probability of Exceeding 5 Rem	**~0.25**	**~0.01**	**~0.06**
Conditional Probability of Exceeding 50 Rem	**~0.12**	**<0.001**	**~0.01**
Conditional Probability of Exceeding 200 rem	**~0.01**	**<0.001**	**~0.002**

As can be seen from the above comparison, truncation of sequences at a level of 10^{-7} per year has a significant impact on the results. However, the comparison with the NUREG-0396 results as a basis for the proposed 0.5 mile boundary of the ALWR response area remains valid.

The peer review group strongly endorses the concept of a screening level for core damage frequency below which events are so improbable that they should not be factored into a regulatory decision process. However, because the NUREG-0396 process is conditional on core damage, for consistency the ALWR analyses must include all core damage sequences that affect the results over the range that they are presented. (This should include accident sequences resulting from fires, internal floods, or those occurring during low power or shutdown conditions.) In practice, this implies that the truncation level can be no higher than three orders of magnitude below the core melt frequency.

Finally, the peer review group notes that, although the use of the dose exceedance curve conditional on core damage frequency is consistent with the presentation in NUREG-0396, it does not fully provide for a comparison of the lower risks of severe accidents estimated for the ALWRs. Accordingly, the peer review group believes that an additional comparison with the results of NUREG-0396, based on absolute probability of exceeding 1, 5, 50, and 200 rem, would provide additional insights useful in consideration of revised emergency planning for the ALWRs.

Treatment of Decontamination Factors

Integrity of the reactor containment building is very important in limiting the consequences of a severe accident. The exposure of members of the public to dose levels that could be immediately life threatening due to radiation sickness would only be predicted to occur in accident sequences involving early containment failure or containment bypass. Through adherence to Advanced Light Water Reactor Design Utility Requirements Document, the ALWR designs provide much higher assurance that the containment will remain intact for an extended time period in a severe accident and that the containment will not be bypassed. As a result of decreasing the likelihood of the early containment failure and interfacing system loss of coolant accident bypass scenarios, the severe accident scenarios involving unisolated steam generator tube rupture in the PWR ALWRs have assumed greater residual significance. Although the frequency of these scenarios has also been reduced and is now extremely low, they remain as a mechanism by which the containment can be bypassed.

In probabilistic risk assessments that have been performed to date, very little credit for radionuclide aerosol retention within the steam generator has been given for this type of scenario. The draft Technical Aspects of ALWR Emergency Planning report, in contrast, estimates a substantial decontamination factor of approximately 100 for aerosol deposition processes within the steam generator before release the environment. Three aerosol deposition mechanisms are assessed: turbulent deposition within the broken tube, thermophoresis (temperature gradient driven deposition from the hot gases to the colder steam generator tube surfaces), and turbulence-enhanced eddy diffusion driven deposition on the external surfaces of tubes as the jet from the broken tube expands across a number of rows of tubes. The strongest aspect of the analysis is that the results depend on three independent mechanisms, rather than on a single mechanism. (Inertial impaction, which would be a fourth independent mechanism, was not included in the analysis.) The turbulent deposition and thermophoresis models have been used previously and have received considerable review. The eddy diffusion deposition model has not been reviewed previously.

The committee has concerns with the manner in which the authors estimate the magnitude of turbulence-enhanced inertial deposition (based on the data of Douglas and Ilias). Polestar has proposed an alternative application of the data which is more acceptable. Nevertheless, the committee believes that steam generator decontamination factor is an area of potential weakness in the technical case that has been developed. Because of the importance of a large decontamination factor to the overall results, more effort to substantiate a large decontamination factor is warranted to assure acceptance in the broader technical community.

Another assumption in the report that was important to the estimate of a large decontamination factor, was the assumption of successful accident management to maintain minimal water injection into the steam generators for cooling. However, the report provided limited discussion of the basis for this assumption and did not attempt to quantify the likelihood of successful accident management to inject water into the steam generators. A stronger technical basis is needed in the report to support the assumption of successful accident management in this area.

Recommendations

1. Presentation of results should be made without truncation to avoid potential masking of risk significant low probability events.

2. All accident sequences of importance to emergency planning considerations, including those resulting from internal events, fires, and low power or shutdown events should be included in the analysis.

3. Additional justification and defense of the decontamination factors utilized in the containment bypass PWR steam generator tube rupture evaluation is needed.

4. Discussion of the ingestion pathway evaluation should be revised to more clearly show methodology used is consistent with NUREG-0396.

5. Additional discussion of new accident management guidelines and their implementation is needed as these are important to the decontamination factor assumptions.

6. Sequences can be eliminated from the analysis on the basis of extended time to release of radioactive material. However, the minimum delay time used for elimination of sequences should not be less than 24 hours.

Conclusions

The current technical basis for nuclear power plant emergency planning is contained in NUREG-0396. NUREG-0396 was published in 1978 and reflected evaluations of plant designs of that time. It relied on the 1975 Reactor Safety Study — WASH-1400. Although it was the best available probabilistic safety assessment at that time (such analysis techniques were in their infancy for the nuclear industry), much has been learned since.

The Advanced Light Water Reactor Program sponsored by EPRI and DOE has led to significant improvement in plant design, especially for severe accident mitigation capability. The three ALWR designs evaluated in the current study were subjected to PSAs utilizing state of the art techniques and methodology.

EPRI —TR-113509 retains the basic methodology of NUREG-0396, while updating the information relied upon for input. Not surprisingly, the results demonstrate lower risks from severe accidents for the ALWR. This lower risk is an important consideration in any decisions on offsite emergency planning for ALWRs.

The peer review group focused on evaluating the technical basis for the proposed changes in emergency planning, rather than the broad policy issues that are needed to effect a change in current requirements. Among those that will eventually have to be addressed are the proper roles of state and local governments in response actions and the use of FEMA all hazards plans for expansion beyond the new smaller planning zones.

Regarding the technical basis provided in the report, the peer review group has made several recommendations to strengthen and improve the presentation of the material. If due regard is paid to the recommendations, it is the consensus of the group that EPRI-TR-113509 provides a reasonable revised basis to support decisions on emergency planning for ALWRs.