ELECTRIC POWER RESEARCH INSTITUTE

Summary of EPRI Research Applicable to Nuclear Accident Scenarios

2011 TECHNICAL UPDATE

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Summary of EPRI Research Applicable to Nuclear Accident Scenarios

This document does NOT meet the requirements of 10CFR50 Appendix B, 10CFR Part 21, ANSI N45.2-1977 and/or the intent of ISO-9001 (1994)

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Alan Ross and Associates

This is an EPRI Technical Update report. A Technical Update report is intended as an informal report of continuing research, a meeting, or a topical study. It is not a final EPRI technical report.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

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Abstract

The events at Fukushima Daiichi Nuclear Power Plant following the March 11, 2011, earthquake and the subsequent tsunami have heightened the need for widespread dissemination of information available within the nuclear industry that addresses subjects pertinent to the on-going situation at the plant. These subjects include, but are not necessarily limited to:

- Hydrogen generation
- Loss of off-site power
- Reactor core performance following a loss of coolant
- Iodine removal
- Emergency response planning
- Emergency diesel generator performance
- Containment structural performance following an accident

Much information on these subjects was gathered following the accident at the Three Mile Island Unit 2 nuclear power plant in March 1979, and the knowledge base has subsequently been significantly supplemented by research projects performed with the participation of and/or under the sponsorship of the Electric Power Research Institute (EPRI) and others.

EPRI maintains a significant library of documents, both electronic and in hard copy, that address these subjects. Brief synopses of the available documents are contained in this report.

Keywords
Emergency diesel generators
Emergency response planning
Fukushima Daiichi
Hydrogen generation
Iodine removal
Loss of off-site power
Reactor core performance
Three Mile Island event

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Section 1: Introduction

The events at Fukushima Dai-ichi Nuclear Power Plant following the March 11, 2011 earthquake and subsequent Tsunami has heightened the need for widespread dissemination of information available within the nuclear industry that addresses subjects pertinent to the on-going situation at the plant. These subjects include, but are not necessarily limited to:

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EPRI maintains a significant library of documents, both electronic and in hard copy, that address the above the subjects. The available documents and a brief synopsis thereof are delineated in the attached listing. This listing includes documents from six general categories:

- Station Blackout Reports
- Hydrogen Generation-Related Reports (on CD-ROM)
- Advanced Containment Experiments (ACE) Phase B – Containment Iodine Behavior Reports
- Advanced Containment Experiments (ACE) Phase C – Molten Corium Concrete Interaction (MCCI) Reports
- Other reports on MCCI, Emergency Response, Containment Response to Missiles and Earthquakes, and Iodine
- Reports Prepared by the Nuclear Safety Analysis Center (NSAC)

Copies of all of the indicated reports are maintained by EPRI and can be made available in electronic media and/or hard copy versions.

Section 2: Station Blackout

One of the major causes of the problems that impacted the Fukushima Dai-ichi plant was the loss of the Emergency Diesel Generators as a result of the tsunami that impacted the plant site following the March 11, 2011 earthquake combined with a loss of off-site power and exhaustion of the station batteries resulting in a total station blackout.

The issue of station blackout arose because of the historical experience regarding the reliability of AC power supplies. There had been numerous reports of emergency diesel generators failing to start and run in operating plants. In addition, a number of operating plants experienced a total loss of offsite electrical power. In almost every one of these loss of offsite power events, the onsite emergency AC power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies had been available. In a few cases, there was a complete loss of AC power, but during these events AC power was restored in a short time without any serious consequences.

The results of WASH-1400 showed that, for one of the two plants evaluated, a station blackout accident could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of station blackout accidents was established. This finding and the concern for diesel generator reliability based on operating experience raised station blackout to an Unresolved Safety Issue in the 1979 NRC Annual Report. A detailed action plan for resolving this issue was published in NUREG-0649, Revision 1.

The final evaluation of station blackout accidents at nuclear power plants was performed by the USNRC staff and published in NUREG-1032. In resolving this issue, the staff performed a regulatory analysis which was documented in NUREG-1109. In June 1988, this USI was resolved with publication of a new rule (53 FR 23203) and Regulatory Guide 1.155. Thus, this issue was considered to be RESOLVED and new requirements were established.

This issue was also addressed in the United States as Generic Safety Issue (GSI) A-44, "Station Blackout." The results of the associated analysis are described in NUREG – 0933 (main report with Supplements 1-33).

The effectiveness of the Station Blackout rule was subsequently evaluated by the USNRC staff in NUREG – 1776, "Regulatory Effectiveness of the Station Blackout Rule," which was issued in August, 2003. This NUREG as well as other EPRI and USNRC reports that address the overall subject are delineated in Table 2-1.

Table 2-1
Station Blackout Reports

Report No.	Title	Synopsis
EPRI TR-102814	"A Methodology for Determining and EDG's Capability to Start its Emergency Loads," August, 1993	This report describes a combined testing and analytic methodology that has been developed and successfully used to determine the emergency load starting capabilities of the Ginna emergency diesel generators.
NUREG - 1776	"Regulatory Effectiveness of the Station Blackout Rule," August, 2003	This report evaluates the effectiveness of the SBO rule by comparing regulatory expectations to outcomes. A set of baseline expectations was established from the SBO rule and related regulatory documents in the areas of coping capability, risk reduction, emergency diesel generator reliability, and value-impact.
NUREG/CR 6890, Vols. 1, 2, 3	"Reevaluation of Station Blackout Risk at Nuclear Power Plants," December, 2005	This report is an update of previous reports analyzing loss of offsite power (LOOP) events and the associated station blackout (SBO) core damage risk at U.S. commercial nuclear power plants.
EPRI TR-1007320	"Probability and Consequences of Double Sequencing Nuclear Power Plants Safety Loads," October, 2002	This report examines the probability, consequences and technical subtleties of double sequencing, (i.e., an unintended series of operations at a nuclear power plant in which safety and accident mitigations loads automatically start, shut down, and restart in rapid succession when called on to operate.
EPRI 1009110	"Probability and Consequences of Double Sequencing Nuclear Power Plant Safety Loads, Rev. 1," 10/2003	This report examines the probability, consequences, and technical subtleties of double sequencing. It is an update of the report cited directly above.
EPRI TR-1007966	"Double Sequencing Analysis for BWRs: The Probability and Consequences of Double Sequencing Nuclear Power Plant Safety Loads Considerations Specific to Boiling Water Plants," October, 2003	This report is a supplement to EPRI report 1009110. This supplement reports on a follow-up BWR-specific effort to identify any subtle differences between a PWR's and BWR's response to double sequencing.
EPRI 1011759	"Frequency Determination Method for Cascading Grid Events," December, 2005	This report provides recommendations to the utility PSA analyst in qualitatively and quantitatively considering the risk implications of power grid events on nuclear plant operation and maintenance.

Section 3: Hydrogen Generation

In the United States, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that light-water nuclear reactors fueled with uranium oxide pellets within cylindrical Zircaloy cladding be provided with emergency core cooling systems (ECCS) that are designed in such a way that their calculated core cooling performance after a postulated loss-of-coolant accident (LOCA) conforms to certain criteria specified in paragraph 50.46(b). Paragraph 50.46(b)(1) requires that the calculated maximum temperature of fuel element cladding not be ~ greater than 2200˚F. In addition, paragraphs 50.46(b) (2) through (b) (5), which contain required limits for calculated maximum cladding oxidation and maximum hydrogen generation, require that calculated changes in core geometry remain amenable to cooling and that long-term decay heat removal be provided. At temperatures above 2200˚F, the Zircaloy reacts with steam and water to create hydrogen. The overall subject of calculation of Emergency Core Cooling System performance is addressed by 10 CFR Part 50, Appendix K and Regulatory Guide 1.157.

At Fukushima Dai-ichi, the loss of cooling water as a result of the station blackout resulted in the generation of hydrogen that was subsequently vented to the interior of the secondary containment as part of the efforts to reduce pressure within the reactor core and reactor coolant system. Subsequently, the hydrogen mixed with air in the upper reaches of Units 1 and 3 and was ignited by some unknown (at this time) source, blasting away the sheet metal roofs and sides of the top section of the outer secondary containment buildings. At unit 2, workers had time to remove a wall panel at the top of the reactor building providing an exit for hydrogen, avoiding a similar roof-level explosion. The damage to the buildings 1 and 3 and the opening in 2 created an exit route for radioactive releases from the spent fuel pools at the top of the reactor pools.

Following the 1979 accident at the Three Mile Island Nuclear Power Plant, numerous studies were undertaken under the sponsorship of EPRI and others related to the generation of hydrogen and its removal following a loss of coolant accident. Reports addressing those activities as well as other more recent studies related to the subject are delineated in Table 3-1 and are available on CD-ROM.

Table 3-1
Hydrogen Generation Reports

Report No.	Title	Synopsis
NUREG/CR-2017	"Flammability Limits and Pressure Development in Hydrogen-Air Mixtures," January, 1981	This paper reviews the then-current state of knowledge of the flammability limits and pressure development in H2-air mixtures, and ternary mixtures of H2-Air-N2 and H2-air steam, in constant- volume enclosures.
EPRI NP-2637	"The Effect of Water Fogs on the Deliberate Ignition of Hydrogen," November, 1982	This report presents an experimental evaluation of the effects of water fog density, droplet diameter, and temperature on the lower flammable limit (LFL) of hydrogen-air-steam mixtures.
Paper	"Analysis of the Three Miles Island (TMI-2) Hydrogen Burn," January, 1983	As a basis for the analysis of the hydrogen burn which occurred in the Three Mile Island Containment, a study of recorded temperatures and pressures was made.
EPRI NP-2669	"Hydrogen Mixing and Distribution in Containment Atmospheres," March, 1983	Hydrogen mixing and distribution tests are reported for a modeled hydrogen release from a postulated small pipe break or from a pressurizer relief tank rupture disc into the simulated lower compartment of an LWR ice condenser plant.
EPRI NP-2953	"Hydrogen Mixing and Distribution in Containment Atmospheres," March, 1983	Experiments were conducted to examine the combustion behavior of hydrogen under containment conditions which might occur in a postulated degraded core accident. Parameters included hydrogen concentration, hydrogen and steam flow rates, water vapor concentration, igniter location, and water spray characteristics.
IDCOR TR-12.2	"Hydrogen Distribution in Reactor Containment Buildings," September, 1983	This report describes a review and assessment of hydrogen distribution in reactor containment buildings during postulated degraded core accident scenarios.
IDCOR TR - 12.3	"Hydrogen Combustion in Reactor Containment Buildings," September, 1983	This report provides a review and assessment of hydrogen combustion in reactor containment buildings during postulated degraded core accident scenarios.
EPRI NSAC-59	"Simulation of Hydrogen Mixing using the Containment Atmospheric Prediction Code," November, 1983	The Containment Atmosphere Prediction Code (CAP) has been developed for the analysis of hydrogen mixing in nuclear reactor containments. CAP results have been compared against selected experiments performed at Battelle Institute Frankfurt and at Hanford Engineering Development Laboratory.

Table 3-1 (continued)
Hydrogen Generation Reports

Report No.	Title	Synopsis
IDCOR TR - 13.2-3	"Evaluation of Means to Prevent, Suppress, or Control Hydrogen Burning in Reactor Containments," 2/1983	This report describes a program to evaluate various concepts to prevent, suppress, or control hydrogen burning in containments of light water reactors (LWR).
EPRI NP-2955	"Evaluation of Means to Prevent, Suppress, or Control Hydrogen Burning in Reactor Containments," 6/1984	This report describes a study of combustion of hydrogen-air-steam mixtures in a 2.3-m (8 ft.) diameter sphere and in a pipe-sphere combination consisting of a 0.3-m (1 ft.) diameter. 6-m (20 ft.) long pipe connected to the sphere.
EPRI NP-2956	"Effectiveness of Thermal Ignition Devices in Lean Hydrogen-Air-Steam Mixtures, " March, 1985	Deliberate ignition of hydrogen at low concentrations in reactor containment systems is one method of controlling hydrogen during degraded core accidents. This report describes experiments that were performed to determine the hydrogen-air-steam concentration regimes in which ignitors would be effective.
EPRI NP-3955	"TEMPEST Code Simulation of Hydrogen Distribution in Reactor Containment Structures," March, 1985	The mass transport version of the TEMPEST computer code was used to simulate hydrogen distribution in geometric configurations relevant to reactor containment structures.
EPRI NP-3975	"Analysis of the Hydrogen Bum in the TMI-2 Containment," April, 1985	A comprehensive analysis of the TMI-2 hydrogen deflagration is described encompassing hydrogen generation/release rates, hydrogen mixing prior to the burn, likely ignition sources, combustion and cool-down phenomena, and burn-induced mechanical and thermal damage.
NUREG/CR-4138	"Data Analysis For Nevada Tests" May, 1985	This report provides results from an in-depth analysis of twenty-one premixed large-scale combustion experiments sponsored by the U. S. NRC and the Electric Power Research Institute (EPRI).
EPRI NP-4354	"Large-Scale Hydrogen Burn Equipment Experiments," December, 1985	Experiments were conducted to examine the ability of selected types of equipment in nuclear plants to operate during and after a hydrogen burn event, such as that which occurred in the TMI-2 accident. Both premixed and continuous injection tests were conducted.

Table 3-1 (continued)
Hydrogen Generation Reports

Report No.	Title	Synopsis
EPRINP-5254	"Effectiveness of Thermal Ignition Devices in Rich Hydrogen-Air-Steam Mixtures," July, 1987	Deliberate ignition of hydrogen with glow plug ignitors is one method of mitigating the possible effects of a hydrogen burn on containment integrity in potential reactor accidents. Since these accidents may produce substantial amounts of steam, experiments have been performed to determine the hydrogen-air-steam regimes for which these ignitors would be effective.
RHO-RE-EV-95P	"Lessons Learned from Hydrogen Generation and Burning During the TMI-2 Event," March, 1987	This document summarizes what has been learned from generation of hydrogen in the reactor core and the hydrogen burn that occurred in the containment building of the Three Mile Island Unit No. 2 (TMI-2) nuclear power plant.
EPRI NP-3878 Vols. 1 and 2	"Large-Scale Hydrogen Combustion Experiments," October, 1988	Forty large-scale experiments to investigate the combustion behavior of hydrogen during postulated degraded core accidents were conducted in a 16 m (52 ft) diameter sphere. The performance of safety related equipment and cable also was examined.
EPRI NP-5721-CCM	"HICCUP: A Computer Code for Hydrogen Injection, Combustion, and Cooldown using Phenomenological Models," April, 1988	HICCUP, the computer code described in this report, models single or multiple non-adiabatic burns of lean hydrogen-air-steam mixtures in a closed vessel. Buoyant axisymmetric flame propagation patterns are assumed and empirical burning velocity correlations are used to calculate burning rates.
No Number	"Technical Support for the Hydrogen Control Requirement for the EPRI Advanced Light Water Reactor Requirements Document," August, 1989	To eliminate hydrogen as a significant risk contributor for advanced light water reactors (ALWRs), the Electric Power Research Institute (EPRI) ALWR Requirements Document has established a hydrogen control requirement as an element of the licensing design basis.
ASME Jour. Of Heat Transfer No. 112	"Analysis of Hydrogen Combustion in a 1/4-Scale Boiling Water Reactor Containment Building," February, 1990	A mathematical model has been developed for the analysis of hydrogen combustion in boiling water reactor containment

Table 3-1 (continued)
Hydrogen Generation Reports

Report No.	Title	Synopsis
NEA/CSNI/R(93)4	"International Standard problem ISP-29 on Distribution of Hydrogen within the HDR Containment under Severe Accident Conditions," Final Comparison Report, February, 1993	The present report summarizes the results of the International Standard Problem Exercise ISP-29, based on the HDR Hydrogen Distribution Experiment E11.2. Post-test analyses are compared to experimentally measured parameters, well-known to the analysts.
NEA/CSNI/R(99)16	"State-of-the Art Report on Containment Thermal/ Hydraulics and Hydrogen Distribution," June, 1999	This document describes a state-of-the-art report (SOAR), to bring together what had been learned from several International Standard Problem (ISP) exercises regarding the predictive capability of codes, as well as the need for a discussion of the codes' adequacy (or inadequacy) for plant applications.
NEA/CSNI/R(2000)7	"Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety, State-of-the-Art Report by a Group of Experts," August, 2000	A review of Flame Acceleration (FA) and Deflagration-to-Detonation Transition (DDT) in Containment had been prepared for the NEA Committee on the Safety of Nuclear Installations (CSNI) as a State- of-the-Art Report (SOAR) in 1992. This report updates that earlier report.

Section 4: Advanced Containment Experiments (ACE) Phase B – Containment Iodine Behavior

Accurate determination of the extent of Iodine in the environment following loss of coolant and other accidents at a nuclear power plant and the effectiveness of actions taken to reduce the level of iodine is an important area of concern in the nuclear industry. Of primary concern is the amount of the isotope Iodine-131, which has a half-life of 8+ days.

The NRC's traditional methods for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Regulatory Guides that address the concern include:

> Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors"
>
> Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"
>
> Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors"
>
> Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"
>
> Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"

In 2000, the NRC issued a more comprehensive regulatory guide, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design

Basis Accidents at Nuclear Power Reactors," which supersedes certain aspects of the earlier RG's.

The subject of iodine creation, behavior and removal was addressed in-depth at the internationally supported Advanced Containment Experiments (ACE) Containment Systems Test Facility (CSTF) at Hanford, Washington and other locations. Reports on these activities are delineated and discussed in Table 4-1.

Table 4-1
Advanced Containment Experiments (ACE) Phase B, Containment Iodine Behavior Reports

Report No.	Title	Synopsis
TR-B1	"Experimental Facilities Involved in ACE Phase B (Iodine) Program"	List of Facilities on 35mm Slides
TR-B2	"Review of Analytical Techniques in Support of the Advanced Containment Experiments (ACE) Containment Systems Test Facility (CSTF) Experimental Program," October 1989	A review of techniques, procedures and instrumentation suitable for providing analytical support to large scale tests to be conducted at the CSTF at Hanford, Washington. The tests are intended to study the behavior of Iodine within containment during an accident.
TR-B3	"Final Report on the ACE-RTF Experiments," July, 1992	Results of the Advanced Containment Experiments (ACE) Radioiodine Test Facility (RTF) program are reported. This study consisted of four intermediate-scale experiments that investigated the effects of radiation, pH, surfaces and initial iodine speciation on iodine behavior.
TR-B4	"The Effects of pH, Temperature, Radiation, and Epoxy Paint on Iodine Behavior Under Conditions Relevant to Reactor Accidents,"	The effects of pH, temperature, gamma irradiation, initial speciation and epoxy paint on iodine behavior were studied in bench-scale experiments encompassing 24 conditions relevant to reactor accidents.
TR-B5	"Adsorption of Iodine on Aerosols," April, 1991	Molecular iodine adsorption on various aerosols was determined to be very rapid. No reaction of I2 with CsNo3, HBO2 or MnO was found.
TR-B6	"The Oxidation of Cesium Iodine in Stationary Pre-Mixed Hydrogen Flames,"	The fate of solid cesium iodide in a variety of lean to stoichiometric stationary hydrogen air flames has been investigated by wet-chemical techniques. Variable oxidation of vaporized CsI to molecular iodine and iodates has been demonstrated.
TR-B7	"Summary of Pretest Iodine Chemistry and Transport Calculations for the Radioiodine Test Facility (RTF) Experiments," January, 1991	The report presents and evaluates results from pretest computer-code calculations. The RTF experiments investigate iodine chemistry and transport effects that occur over a 7 day period when iodine species are injected into a water pool.

Table 4-1 (continued)
Advanced Containment Experiments (ACE) Phase B, Containment Iodine Behavior Reports

Report No.	Title	Synopsis
TR-B8	"The Effects of Steam on the Oxidation of CSI Aerosol During Hydrogen Combustion," June, 1992	The effects of hydrogen deflagrations on suspended CsI aerosols have been studied in a series of tests in the Containment Test Facility. In a 10 vol% steam environment, CsI oxidation was not observed, even in near- stoichiometric hydrogen-air-steam mixtures.
TR-B9	"Summary of International R&D Work on Iodine Behavior in Containment During Reactor Accidents," 1990	This document presents the results of an attempt to show how t the Phase B work is related to and supports all the other R & D work that exists or is planned in the area of iodine behavior in containments under accident conditions.
TR-B10	"Results of the Large Scale Iodine Experiment at the Containment Systems Test Facility," May, 1992	The behavior of iodine was studied under conditions simulating its release into a nuclear reactor containment vessel following a possible severe accident. The test environment was a condensing steam-air mixture at 100°C and 220 kPa in a 852-m3 vessel containing typically painted surfaces.
TR-B11	"Bench Scale Studies #2-the Adsorption of I_2 Vapours by Epoxy Paint at 80° C," May, 1992	A limited investigation of the interaction of iodine with epoxy painted surfaces was undertaken. The initial deposition velocity appeared to be controlled by the rate of gas phase mass transfer rather than any specific property of the surface.
TR-B12	"Fission Product Vapor-Aerosol Interactions in the Containment: Simulant Fuel Studies," 8/1990	Experiments have been conducted to study the interaction of fission product vapors released from simulant fuel samples with· control rod aerosols. Measurements were made of the overall deposition profiles of the elemental components.
TR-B13	"Interaction of Simulant Fission Product Vapors With Control Rod Aerosol in a Flowing System," 8-1990	Aerosols generated from various reactor materials during a severe accident could dominate the transport behavior of fission products released from the damaged core. Experiments have been undertaken to study the interaction of a variety of simulant fission products with aerosols generated from overheated Ag-In-Cd control rods in a flowing system.
TR-B14	"The Interaction of Molecular Iodine Vapor With Silver-Indium-Cadmium Control Rod Aerosol," August, 1990	The interaction of labile fission product vapors with structural- materials aerosols has been identified as a major uncertainty in the mechanistic assessment of severe reactor accidents. Experiments have been undertaken to characterize and quantify the interaction of non-radioactive iodine vapor with control rod aerosol.

Table 4-1 (continued)
Advanced Containment Experiments (ACE) Phase B, Containment Iodine Behavior Reports

Report No.	Title	Synopsis
TR-B15	"The Interactions of Fission Product Vapors With Aerosols in Severe Reactor Accidents," 8-1990	Complex mechanistic codes have been developed to model the physical and chemical characteristics of the radioactive materials transported through a nuclear power plant during the course of a severe reactor accident.
TR-B16	"Review of Available Data for Iodine Species Retention on Selective Filter Packs," 10-1990	From a review of data on retention and blow-off (penetration) of iodine species on components of selective iodine species samplers, quantitative values are proposed for the standard Harwell "Maypack" species sampler.
TR-B17	"The Radiolysis of Aqueous Solutions of Caesium Iodide and Caesium Iodate," October, 1990	Yields of the radiolysis products I_2 and I_3 have been measured from CsI and CsIO3 solutions irradiated under a range of conditions.
TR-B18	"Aerosol Deposition and Resuspension in Tubes and Deposition in Bends," January, 1991	The main object of the study was to analyze deposition and resuspension of deposited wet aerosol on tube surface. The investigation was carried out using horizontal straight tubes and tubes with 90° bends.
TR-B19	"Condensational Growth of Some Hygroscopic Aerosols in Moist Air," January, 1991	The behavior of hygroscopic aerosol was studied at 40°C under different R.H. levels. The· main interest was given to high humidity conditions.
TR-B20	"Numerical Simulation of Binary Nucleation of Hydrogen Iodide and Water Vapors," January, 1991	Heteromolecular homogeneous nucleation of water and hydrogen iodide vapors has been investigated. Numerical simulation shows that significant nucleation will occur in some conditions.
TR-B21	"Experimental Research of Radioactive Iodine Behavior in Gas-Liquid Systems With Active Sprinklers," May, 1992	In this paper the possibility of formation of the organic iodine combinations is examined with CsI-131 finding in the water, not containing special chemical additions on plants with the ejector and sprinkler systems.
TR-B22	"User's Handbook for the Iodine Severe Accident Behavior Code, Impair 2.2," May. 1992	This publication describes the second version of the Severe Accident Code (IMPAIR 2.2). This code aims to model postulated conditions of iodine chemistry present in a Iodine deposition and atmosphere) during a postulated severe accident in a LWR.

Table 4-1 (continued)
Advanced Containment Experiments (ACE) Phase B, Containment Iodine Behavior Reports

Report No.	Title	Synopsis
TR-B23	“Chemical Behavior of Iodine in the System Modeling WWER Type of NPP,” June, 1992	This study addresses the radiation chemical processes occurring in various aqueous systems used in the NPP of the WWER type. Emphasis was put on the chemistry of iodine as a difficult element for managing of the removal processes.
TR-B24	“Interpretation of ACE/RTF Test 2 Results- Preliminary Technical Note,” June, 1992	This technical note gives the state of the interpretation with the IODE code of the second test series realized in RTF (Radioiodine Test Facility)
TR-B25	Interpretation of ACE/RTF Test 3 Results- Preliminary Technical Note,” April, 1992	This document provides an update of the results presented inTR-B24 above.
TR-B26	“An Assessment of the Current Database on Iodine Behavior in Containment During Severe Accidents,” December, 1992	The adequacy of the current data base on iodine behavior in reactor containments is assessed with respect to its ability to support useful safety analyses and source term calculations for severe accidents.
TR-B27	“Containment Iodine Behavior Experiments, Advanced Containment Experiments (ACE) Project: Summary Report,” November, 1993. Also issued as EPRI Report 103212.	This report summarizes the experimental design of the separate program components and describes key results from 25 reports. Also discussed is the significance of this body of new information to the problem of predicting iodine behavior In containment during severe accidents.
TR-B-01	"Specifying Useful Mass Transfer Models,” 11/1995	Final report subsumed into TR-B-05
TR-B-02	“Quantifying the Effects of Structural Surface Interactions,” November, 1995	Final Report subsumed into TR-B-06
TR-B-03	“Methodology for the pH Prediction in PWR Sump Water under Severe Accident Conditions,” December, 1997	This paper lists all of the systems that provide the sump with chemical species such as boron, lithium hydroxide and soda in the case of pressurized water reactors and to determine a method for calculating the pH of the sump water in the event of a severe accident.

Table 4-1 (continued)
Advanced Containment Experiments (ACE) Phase B, Containment Iodine Behavior Reports

Report No.	Title	Synopsis
TR-B-04	Quantifying the Effects of Organic Reactions," 7/97	As a consequence of the perceived importance of organic iodide formation to reactor siting, extensive experimental programs were put in place in the late 1960's and 1970's to develop quantitative guidelines for predicting organic iodide formation. Since that time, a number of models have been developed to rationalize the results from these experiments
TR-B-05	"Interfacial Transfer of Iodine in Containment," July, 1997	A review of mass transfer theory, correlations and parameters relevant to the interfacial transfer of iodine in containment was performed. A number of potentially useful correlations for estimating interfacial transfer coefficients were identified.
TR-B-06	"The Reaction of Iodine with Surfaces," July, 1997	This report summarizes work carried out on reactions of iodine with structural surfaces as part of the ACE extension program. The work comprised assessment of the experimental database processes and recommendations for further work.
TR-C35	"Reduction of the INSPECT Reaction Set for Modeling Aqueous Iodine Chemistry," January, 2000	A study has been carried out to determine the minimum set of iodine reactions which can satisfactorily reproduce the predictions of the full INSPECT model over a range of conditions of relevance to nuclear reactor safety studies.
TR-C36	"A Description of the Aqueous Iodine Database in LIRIC and the Development of a Reduced Iodine Chemistry Subset," January, 2000	An updated version of the aqueous iodine data-base from URIC (Library of Iodine Reactions in Containment) is described in this document. Recent modifications to the model include addition of the temperature sensitivities of many of the rate constants in the iodine and water radiolysis reaction sets.
TR-C37	"A Reduced Reaction Set for Iodine Chemistry in Reactor Faults," January, 2000	As part of the Advanced containment tests (ACE) program, coordinated by EPRI, work has been carried out by AEAT and AECL to produce a consensus reaction set for the aqueous iodine database.
TR-C38	"Iodine Behavior Models on Organic Reactions: The Dissolution of Solvents from Containment Paints in the Presence of Water," February, 2001	This report summarizes studies performed at AECL on the rate of release of various solvents from containment paints in contact with water.

Table 4-1 (continued)
Advanced Containment Experiments (ACE) Phase B, Containment Iodine Behavior Reports

Report No.	Title	Synopsis
TR-C39	"Iodine Behavior Models on Organic Reactions: Partitioning and Hydrolysis/ Radiolysis of Organic Iodides," February, 2001	This report deals with the volatility and decomposition of organic iodides.
TR-C40	An MCCI Report	
TR-C41	"The Reaction of Gaseous Iodine with Steel and Paint Surfaces," March, 2002	Measurements have been made on iodine deposition on steel and paint surfaces. The objective of this work was to provide deposition data as a function of temperature, iodine concentration and humidity for use in the development and verification of models.
1011037	"Iodine Behavior Within Confinement, Advanced Containment Experiments Extension (ACEX) Project," July, 2004	The Advanced Containment Experiments Extension (ACEX) consortium assessed available data on iodine behavior within containment. ACEX used this information to validate existing models and computer codes and to develop new ones.

Section 5: Advanced Containment Experiments (ACE) Phase C – Molten Corium Concrete Interactions (MCCI)

A loss of coolant accident (LOCA) at a nuclear power plant can result in the melting of the reactor corium, a lava-like molten mixture of portions of nuclear reactor core, formed during a nuclear meltdown. The corium may consist of nuclear fuel, control rods, structural materials from the affected parts of the reactor, products of their chemical reaction with air, water and steam, and, in case the reactor vessel is breached, molten concrete from the floor of the reactor area.

Thermal decomposition of concrete yields water vapor and carbon dioxide, which may further react with the metals in the melt, oxidizing them and being reduced to hydrogen and carbon monoxide. Decomposition of the concrete and volatilization of its alkali components are endothermic processes. Aerosols released during this phase are primarily based on concrete-originating silicon compounds.

The subject of the potential interaction of the corium with these structural materials was addressed during Phase C of the internationally supported Advanced Containment Experiments (ACE) conducted at the Containment Systems Test Facility (CSTF) at Hanford, Washington and other locations. Reports on these activities are delineated and discussed in Table 5-1.

Table 5-1
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C1	"Investigation of Molten Corium Concrete Interaction Phenomena and Aerosol Release in Small and Intermediate-Scale Tests," December, 1987	A series of small- and intermediate-scale experiments have been performed to study the interaction between molten reactor core materials and concrete. The objectives of this experimental program were to add to the database for both the thermal hydraulic and fission product/aerosol transport aspects of molten core-concrete interactions, and to evaluate predictions of computer code modeling by comparison with the experimental results.
TR-C2	Videotape showing Large-Scale Apparatus Demonstration and Test L5	
TR-C3	"Advanced Containment Experiments, System Design and Operations Description," 10/1988	Determination of the aerosol sampling methods and instrumentation to be included in the system was accomplished through lengthy discussions of the advantages and disadvantages of available instrumentation and sampling techniques by aerosol experts.
TR-C4	"ACE Phase C MCCI Experimental Facilities at Argonne National Laboratory," March, 1989	Includes 20 slides
TR-C5	"ACE MCCI Test L5, Post-Test Operations Information Package," October, 1988	This report contains experimental data from ACE MCCI Test 15 for posttest code calculations. Test L5 was the first MCCI/ performed in the ACE Program.
TR-C6	"ACE MCCI Test L2, Pre-Test Information Package," October, 1988	Test L2 is the second in the series of large-scale MCCI/fission product release tests. It investigated the interaction of a partially oxidized corium melt with siliceous concrete.
TR-C7	"ACE MCCI Test L5, Test Data Report," November, 1988	This report' contains experimental data from ACE MCCI Test LS, the first HCCI/fission product release test performed in the ACE Program. It investigated the interaction of fully oxidized corium with limestone/common sand concrete.
TR-C8	"ACE MCCI Test L2, Post-Test Operations Information Package," March, 1989	This report contains experimental data from ACE MCCI Test L2 for posttest code calculations. Test L2 was the second MCCI /fission product release test performed in the ACE Program. It investigated the interaction of a partially oxidized PWR corium melt with siliceous concrete.

Table 5-1 (continued)
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C9	"ACE MCCI Test L1, Pre-Test Information Package," March, 1989	Test Ll is the third in the series of large-scale MCCI/fission product release tests and it will investigate the interaction of a partially oxidized corium melt with limestone/common sand concrete similar to that used in test 15 and the Zion reactor.
TR-C10 Vol. 1	"ACE MCCI Test L2, Test Data Report-Thermal Hydraulics," April, 1989	This report contains experimental data from ACE MCCI Test L2, the second MCCI/fission-product release test. It investigated the interaction of partially oxidized PWR corium with siliceous concrete.
TR-C10 Vol. 2	"ACE MCCI Test L2, Test Data Report Aerosol Analysis," November, 1989	The focus of Volume II of this report is the analysis of the aerosol data and the examination and analysis of the aerosol collected in ACE MCCI Test L2.
TR-C11	"ACE MCCI Test L1, Post-Test Operations Information Package," May, 1989	This report contains experimental data from ACE MCCI Test Ll for posttest code calculations. Test Ll was the third MCCI/ fission product release test and it investigated the interaction of a partially oxidized PWR corium melt with limestone/common sand concrete.
TR-C12	"ACE MCCI Test L6. Pre-Test Information Package," August, 1989	L6 is the fourth in the series of large-scale MCCI/fission product release test and it investigated the interaction of a partially oxidized PWR corium melt with siliceous concrete.
TR-C13	"Vaporization of Strontium, Barium, Lanthanum, and Uranium from Mixtures of Urania, Zirconia, Steel, and Concretes at 2150 K and 2400 K." January 1990. (EPRI Report NP-6613)	The vaporization of strontium, barium, and lanthanum from mixtures of their oxides with urania, zirconia, and concrete was determined with the objective of understanding the release of these refractory fission products during the core- concrete interaction phase of a degraded-core accident.
TR-C14 Vol. 1	"ACE MCCI Test L1, Test Data Report Thermal Hydraulics,"	This report contains experimental data from ACE MCCI Test Ll, the third MCCI/fission-product release test that investigated the interaction of partially oxidized PYR corium with limestone/common sand concrete.
TR-C14 Vol. 2	"ACE MCCI Test L1, Test Data Report Aerosol Analysis," March, 1990	The operating conditions for ACE MCCI Test L6 are described in this report.

Table 5-1 (continued)
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C15	"ACE MCCI Test L6, Post-Test Operations Information Package," July, 1990	
TR-C16	"ACE MCCI Test L4, Pre-Test Information Package," August, 1990	Test L4 is the fifth in the series of large-scale ACE HCCI/fission product release tests. It will investigate the interaction of a partially oxidized boiling water reactor (BWR) corium melt with a two-layered Soviet serpentine/ordinary structural concrete basemat.
TR-C17	Videotape showing Test L6 (11:48 minutes)	
TR-C18	"ACE Phase C MCCI Experimental Facilities at Argonne National Laboratory," 11/1990	Includes 20 slides
TR-C19	"Analysis of Aerosol Samples from Test L5," 5/1990	A series of molten core-concrete interaction (MCCI) tests were conducted. Various analytical techniques have been used to obtain elemental and chemical data on aerosol debris deposited on tantalum and stainless steel coupons from MCCI test LS.
TR-C20	"ACE MCCI Test L4, Post-Test Operations Information Package," November, 1990	This report contains experimental data from ACE MCCI Test LA for posttest code calculations. It investigated the interaction of a partially oxidized boiling water reactor (BWR) corium melt with a two-layered Soviet serpentine/ordinary structural concrete basemat.
TR-C21	"ACE MCCI Test L7, Pre-Test Information Package," January, 1991	Test L7, the sixth in the series of large-scale ACE MCCI/fission product release tests, investigated the interaction of a 70% oxidized boiling water reactor (BWR) corium melt with a limestone/common sand concrete basemat.
TR-C22	"ACE MCCI Test L7, Post-Test Operations Information Package," March, 1991	This report contains experimental data from ACE MCCI Test L7, the sixth large-scale ACE MCCI/fission product release test, that was conducted to investigate the interaction of a 70% oxidized boiling water reactor (BWR) corium melt with a limestone/common sand concrete basemat.

Table 5-1 (continued)
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C23	"ACE MCCI Test L8, Pre-Test Information Package," March, 1991	Test L8, the seventh in the series of large-scale ACE MCCI/fission product release tests investigated the interaction of a 70% oxidized pressurized water reactor (PWR) corium melt with a limestone/limestone concrete basemat.
TR-C24	"Thermochemical Data for the Silicates and Zirconates of Barium, Strontium, Lanthanum and Cerium," March, 1991	This report presents tables of the assessed thermodynamic values for a selected number of compounds of the less volatile fission products of which data are required in calculations of core-concrete interactions.
TR-C25	"ACE MCCI Test L8, Post-Test Operations Information Package," July, 1991	This report contains net electric power and temperature data from ACE MCCI Test L8 for blind posttest code calculation purposes.
TR-C26 Vol. I	"ACE MCCI Test L6, Test Data Report Thermal Hydraulics," August, 1991	This report contains experimental data from ACE MCCI Test L6 that investigated the interaction of a partially oxidized PWR corium melt with siliceous concrete similar to that used in the BETA facility at Karlsruhe.
TR-C26 Vol. II	"ACE MCCI Test L6. Test Data Report Aerosol Analysis," November, 1991	The focus of this report is the analysis of the aerosol data and the examination and analysis of the aerosols collected during ACE MCCI Test l6.
TR-C27	"ACE MCCI Test L6 Post-Test Blind Code Comparison: Thermal Hydraulic Behavior," 9/1991	This report summarizes the results of a comparison of the results of ACE tests L6 and L7 to the model predictions for the thermal-hydraulics of an MCCI and the associated fission product release.
TR-C28	"ACE MCCI Test L6 Post-Test Blind Code Comparison: Aerosol Releases," 9/1991	The focus of this report is the results of the code comparison exercise for MCCI aerosol release for ACE MCCI Test L6. Comparisons are made between aerosol release calculations and the experimentally determined release.
TR-C29	"Molten Core-Concrete Interaction: Analysis of Aerosol Sample From Test L6 of the ACE Project," July, 1991	This report addresses a series of molten core-concrete interaction (MCCI) tests. Various analytical techniques were used to obtain elemental and chemical data on aerosol debris deposited on a Nuclepore filter, tantalum coupon and cyclones from MCCI test L6.

Table 5-1 (continued)
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C30 Vol. I	"ACE MCCI Test L4, Test Data Report Thermal Hydraulics," October, 1991	The purpose of the addressed accident simulations was to investigate the thermal-hydraulic and chemical processes of the MCCI and to expand the database on release of low volatility fission products.
TR-C30 Vol. II	"ACE MCCI Test L4, Test Data Report Aerosol Analysis," February, 1992	This report contains experimental data from ACE MCCI Test L4 that investigated the interaction of a 50% oxidized boiling water reactor (BWR) corium melt with a two-layered Soviet serpentine/ordinary structural (siliceous) concrete basemat.
TR-C31 Vol. I	"ACE MCCI Test L7, Test Data Report Thermal Hydraulics," November, 1991	This report contains experimental data from ACE MCCT Test L7 that investigated the interaction of a 70% oxidized boiling water reactor (BWR) corium melt with a limestone/common sand concrete basemat.
TR-C31 Vol. II	"ACE MCCI Test L7, Test Data Report Aerosol Analysis," February, 1992	This report contains experimental data from ACE MCCI Test L7 that investigated the interaction of a 70% oxidized boiling water reactor (BWR) corium melt with a limestone/common sand concrete basemat.
TR-C32 Vol I	"ACE MCCI Test L8, Test Data Report Thermal Hydraulics," December, 1991	This report contains experimental data from ACE MCCI Test L8 that investigated the interaction of a 70% oxidized pressurized water reactor (PWR) corium melt with a limestone aggregate/limestone sand concrete basemat.
TR-C32 Vol II	"ACE MCCI Test L8, Test Data Report Aerosol Analysis," June, 1992	This report contains experimental data from ACE MCCI Test L8 that investigated the interaction of a 70% oxidized pressurized water reactor (PWR) corium melt with a limestone aggregate/limestone sand concrete basemat.
TR-C33	"Analysis of ACE MCCI Test L6 with the CORCONNANESA Code," February, 1992	Analyses of ACE (Advanced Containment Experiment) MCCI (molten core-concrete interaction) Test L6 were performed at JAERI to evaluate the quantity of FP(fission products) release and the behavior of concrete erosion during an MCCI.
TR-C34	"Molten Core-Concrete Interaction: Analysis of Aerosol Samples From Test L4 of the ACE Project," April, 1991	A series of molten core-concrete interaction (MCCI) tests were conducted. Various analytical techniques were been used to obtain elemental and chemical data on aerosol debris deposited on tantalum coupons, cyclones and a filter from MCCI test L4.
TR-C35	"ACE MCCI Test L7 Post-Test Blind Code Comparison: Thermal Hydraulic Behavior," August, 1992	This report summarizes the post-test analysis of L7 and presents a comparison of the blind post-test MCCI thermal-hydraulic calculational results with data.

Table 5-1 (continued)
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C36	"ACE MCCI Test L8 Post-Test Blind Code Comparison: Thermal Hydraulic Behavior," 1/1994	This report summarizes the post-test analysis of L8 and presents a comparison of the blind post-test MCCI thermal-hydraulic calculational results with data.
TR-C37	"Viscosity of Corium Concrete Mixtures at High Temperatures," January, 1994	The viscosity of corium-concrete mixtures at high temperatures was measured using a Brookfield rotational viscometer. Three different concrete types, siliceous, limestone-sand, and limestone, were employed and the corium was a mixture of urania and zirconia.
TR-C38	"ACE MCCI Test L8 Post-Test Blind Code Comparison: Aerosol Releases," June, 1992	Molten corium-concrete interaction (MCCI) tests L6 and L8 were selected for comparison of experimental releases of fission- product elements with code calculations. Results of the code comparison for ACE Test L8 are given in this report.
TR-C39 Vol. I	No report Issued With This Number	
TR-C39 Vol. II	"Ce, La, and Ru Results Addendum to the ACE MCCl Test Data Reports-Aerosol Analysis," September, 1992	This report provides information regarding the detection of elements not addressed in earlier reports.
TR-C40	"Molten Core-Concrete Interaction: Analysis of Aerosol Samples From Test L7 of the ACE Project," April, 1992	A series of molten core-concrete interaction (MCCI) tests were conducted. Various analytical techniques were used to obtain elemental and chemical data on aerosol debris deposited on a tantalum coupon, cyclone and filter from MCCI test L7.
TR-C41	"Molten core-concrete Interaction: Analysis of Aerosol Samples From Test L8 of the ACE Project," April, 1993	A series-of molten core-concrete-interaction (MCCI) tests were conducted. Various analytical techniques have been used to obtain elemental and chemical data on aerosol debris deposited on tantalum coupons, cyclones and filters from MCCI test L8.
TR-C42 Vol. 1	"ACE Phase C Final Report: MCCI Thermal/Hydraulics Results," September, 1997	This report addresses the fission products released during a MCCI and their impact on the source term (within containment) and its possible release to the environment.

Table 5-1 (continued)
Advanced Containment Experiments (ACE) Phase C, Molten Corium Concrete Interaction (MCCI) Reports

Report No.	Title	Synopsis
TR-C42 Vol. II	"ACE Phase C Final Report: MCCI Aerosol Results," March, 1997	This report addresses the magnitude, composition, and fission- product content of the aerosols released during a molten core- concrete interaction (MCCI) as they are important in estimating the source term of postulated accidents in light-water reactors.
EPRI - TR - 103483	"Molten Corium Concrete Interactions, Advanced Containment Experiments (ACE) Project: Summary Report, December, 1993	This report provides a summary of a series of seven successful large-scale molten corium concrete interaction experiments that measured the ablation rates for a range of concrete types and the release fractions for low-volatility fission products and control materials for prototypical accident conditions.
NUREG/CR - 6032	Solidus and Liquidus Temperatures of Core-Concrete Mixtures," ANL-93/9, June 1993	Solidus and liquidus temperatures were measured for four types of concrete (limestone, limestone sand, basalt, and siliceous) and for their mixtures with urania and zirconia.

Section 6: Other Reports on MCCI, Emergency Response, Containment Response to Missiles and Earthquakes, and Iodine Removal

In addition to the reports delineated in the previous sections, a number of investigatory efforts were undertaken under the sponsorship of EPRI and others addressing subjects related to post-accident conditions including some already addressed (i.e., molten corium – concrete interactions (MCCI) and iodine removal), and other areas not previously addressed (i.e., emergency response, containment response to missiles and earthquakes, and iodine removal). The results of those efforts are delineated in the reports listed in Table 6-1.

Table 6-1
Other Reports on MCCI, Emergency Planning, Containment Response to Missiles and Earthquakes and Iodine Removal

Report No.	Title	Synopsis
1022186	"Technical Foundation of Reactor Safety, Revision 1," October 25, 2010	Following the 1979 accident at the TMI Unit 2, EPRI undertook an intensive research program to investigate fission product behavior during light water reactor (LWR) severe accident scenarios. The EPRI program included experiments to obtain fundamental data and the development of computer-based models of the phenomena involved. This report presents a compendium of knowledge—gained over the past 30 years of research—for resolving LWR severe accident issues.
1013040	"Compendium of LACE/ACE Reports," February 6, 2006	Assessing the consequences of a postulated nuclear plant accident and preparing emergency response plans require an understanding of the behavior of radioactive aerosols released in such an accident. The LWR Aerosol Containment Experiments (LACE) program has helped clarify the processes of aerosol removal and has contributed to resolving regulatory issues related to source terms, NUREG-1150, and severe accident closure. This CD contains the summary report and the key reports from related experimental programs
1019579	"Compendium of EPRI Reports Related to the Advanced Containment Experiments (ACE) Phase C Program," September 1, 2009	This DVD contains all of the project reports from Phase C of the Advanced Containment Experiments (ACE), Molten Corium Concrete Interactions Project plus a related U.S. Nuclear Regulatory Commission regulation (NUREG) report.
1018701	"Compendium of Advanced Containment Experiments (ACE) Phase D and Melt Attack and Coolability Experiments (MACE) Reports," March 1, 2009	This CD-ROM contains all but two project reports from Phase D of the Advanced Containment Experiments (ACE), Melt Attack and Coolability Experiments (MACE), and related reports from the ACE Extension (ACEX) Projects.
TR 100742	"MAAP BWR Application Guidelines," June 1, 1992	The Modular Accident Analysis Program (MAAP) supports utilities in performing plant safety and licensing evaluations. This report provides overall guidelines for using the BWR version of the MAAP code and discusses MAAP modeling of key BWR features as well as phenomena important to predicting severe accidents.

Table 6-1 (continued)
Other Reports on MCCI, Emergency Planning, Containment Response to Missiles and Earthquakes and Iodine Removal

Report No.	Title	Synopsis
1013043	"Compendium of EPRI Reports Related to the MAAP Program" - DVD, March 16, 2006	The Modular Accident Analysis Program (MAAP) computer code provides a flexible, efficient, and integrated tool for evaluating the in-plant effects of a wide range of postulated accidents at nuclear power plants and for examining the impact of operator actions on accident progressions. This compilation of EPRI reports presents the experimental and theoretical background information that forms the basis of the MAAP software, particularly MAAP4 and MAAP5.
1020236	"MAAP4 Applications Guidance," July 30, 2010	The existing Modular Accident Analysis Program Version 4 (MAAP4 documentation consists of the MAAP4 User's Manual, the MAAP4 user's guides for input preparation, and transmittal documents that describe individual revisions. This applications guidance document is intended to provide sufficient information to enable users to optimize their efforts and generate high-quality Level 1 probabilistic risk assessment (PRA) analyses.
1015105	"Risk-Informed Evaluation of Protective Action Strategies for Nuclear Plant Off-Site Emergency Planning," September 18, 2007	This report discusses the significant advancements in the knowledge of potential nuclear accident scenarios and emergency response and the potential benefits to management, nuclear operators, regulatory authorities, and the public from using these advancements to develop improved protective action strategies that are more effective at protecting public health and safety at a cost lower than is currently achievable.
1013492	"Probabilistic Risk Assessment Compendium of Candidate Consensus Models," August 16, 2006	This report provides a compendium of candidate consensus models in use in current probabilistic risk assessments (PRAs). This activity consists of clarifying the definition of consensus models and identifying currently available candidate consensus models. The approaches and models identified as candidates for consensus indicate that some elements of the consensus process, such as a recognized peer review, may be missing for some candidate consensus models.
TR-113509	"Technical Aspects of ALWR Emergency Planning," September 1, 1999	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.

Table 6-1 (continued)
Other Reports on MCCI, Emergency Planning, Containment Response to Missiles and Earthquakes and Iodine Removal

Report No.	Title	Synopsis
NP-2745	"Full-Scale Missile Concrete Impact Experiments," February 1, 1983	This report describes four full-scale rocket-sled experiments conducted to provide data on the response of reinforced-concrete containment walls to impact and penetration by turbine missiles. Missile mass, velocity, and attitude were varied in the tests, as was the thickness of the wall's steel liner. Measures of damage included penetration depth, crater volume, and rear face cracking. Test results are compared with empirical penetration formulas.
NP-7305 SL	"Post-Earthquake Analysis and Data Correlation for the 1/4-Scale Containment Model of the Lotung Experiment," November 1, 1991	Uncertainty and unnecessary conservatism in seismic analysis can be reduced with better understanding of response behavior and validation of analytic methods using actual earthquake response data. This post-earthquake soil-structure interaction (SSI) study develops the needed technical bases for more-realistic seismic design and licensing
TR 108854	"Soil-Structure Interaction of the Lotung Quarter Scale Structure: Sensitivity Studies," September 24, 1997	Results from the Lotung large-scale seismic test have formed a technical basis to support the Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) revision as part of the Unsolved Safety Issue resolution, USI A-40. Two sensitivity studies further quantified conservatisms in NRC-accepted approaches and evaluated uncertainties of soil-structure interaction (SSI) parameters based on probabilistic considerations.
NP-6260 NM	"Criteria and Guidelines for Predicting Concrete Containment Leakage," April 18, 1989	Assessments of the structural behavior of reactor containment building under extreme internal pressures can help utilities more realistically estimate risks from severe core accidents. Combining a large body of experimental data and ABAQUS-EPGEN analytic correlations, this study developed criteria and guidelines for predicting failure mechanisms and identifying leakage locations.
NP - 3774 V1	"Concrete Containment Structural Element Tests: Volume 1," November 1, 1984	This study provides test data on the behavior, under biaxial tensile loads, of large-scale segments of a containment building with reinforced and prestressed concrete designs.
NP-1271	"Nuclear Power Plant Related Iodine Partition Coefficients," December 1, 1979	An experimental study of the partition coefficient between liqui gas-phase iodine is described. Coefficients were measured as a function of the iodine concentration, temperature, and pH of the aqueous solutions.
1011037	"Iodine Behavior Within Confinement: Advanced Containment Experiments Extension (ACEX) Project," July 26, 2004	This report summarizes research conducted by the Advanced Containment Experiments (ACE) Project on the behavior of iodine within reactor containment and integrates the results into computer codes that model this behavior.

Table 6-1 (continued)
Other Reports on MCCI, Emergency Planning, Containment Response to Missiles and Earthquakes and Iodine Removal

Report No.	Title	Synopsis
NP-1269	"Iodine Species in Reactor Effluents and in the Environment," December 1, 1979	This study measured I-131 in effluents from two BWRs and in the air and vegetation near the reactor sites and evaluated published information on the sources and behavior of iodine in the environment.
NP-495	"Sources of Radioiodine at Boiling Water Reactors,"	This report determines specific components in operating BWRs that have the potential for being emission sources of radioactive iodine.
NP-6182	"Downstream Behavior of Volatile Iodine, Cesium, and Tellurium Fission Products," January 17, 1989	A better understanding of how iodine, cesium, and tellurium products deposit on primary reactor components during a degraded core accident can help utility analysts improve computer codes that model product behavior. The mathematical model developed in this study predicts the behavior of deposited fission products vaporized or revaporized by self-heating in a reactor accident.

Section 7: Nuclear Safety Analysis Center (NSAC) Reports

The Nuclear Safety Analysis Center (NSAC) was established soon after an accident at Unit 2 of the Three Mile Island Nuclear Power Plant near Harrisburg, Pennsylvania in March, 1979. The initial efforts of NSAC were directed towards a detailed analysis of the TMI accident and the results thereof. Subsequently, the functions were expanded to include: (1) the analysis of generic nuclear safety issues, (2) serving as a repository for information resulting from the effort to restore the Three Mile Island plant, and (3) serving as a clearinghouse for technical information related to the safety and licensing of nuclear power plants.

Through the years, NSAC has performed investigations addressing a variety of problems that have occurred within or have had the potential to impact the United States nuclear industry. Table 7-1 provides a listing of reports that are considered to be of potential interest to those dealing with the situation at Fukushima Dai-ichi.

Table 7-1
Nuclear Safety Analysis Center (NSAC) Reports

Report No.	Title	Synopsis
NSAC 80-1	"Analysis of Three Mile Island – Unit 2 Accident," Revised March 1980	Summarizes the findings of 4 major studies of the TMI accident: (1) the Nuclear Safety Analysis Center report issued July, 1979; (2) the NRC study issued August, 1979 (NUREG 0600); (3) the President's Commission Report (Kemeny report); and (4) the Rogovin Report issued January, 1980.
NSAC 80-1 Supp. 1	"Supplement to Analysis of TMI Unit 2 Accident"	
NSAC 3	"Analysis and Evaluation of Crystal River – Unit 3 Incident," NSAC-3, March 1980.	Evaluation of incident that shutdown the CR3 PWR unit on February 26, 1980 due to electrical equipment failure causing a loss of coolant event.
NSAC 12	"A Prediction of TMI-2 Core Temperatures From the Fission Product Release History," 11/80	Phenomena associated with fission gas and fission product behavior during TMI accident.
NSAC 16	"Analysis and Evaluation of St. Lucie Unit 1 Natural Circulation Cooldown," December 1980.	Analysis of natural circulation cooldown of the St. Lucie 1 PWR on June 11, 1980
NSAC 17	"Designing for Postaccident Radiological Conditions," Dec-80	Review of in-plant radiological consequences of TMI 2 accident and provision of radiation protection design criteria for balance-of-plant.
NSAC 18	"Workshop on Postaccident Sampling" 3/1981	Provides a description of available sampling systems and recommendations of workshop subgroups.
NSAC 20	"Analysis of Incomplete Control Rod insertion at Browns Ferry 3," December 1980.	Analysis of control rod failure to scram event at TVA Browns Ferry Plant in June, 1980.
NSAC 23	"Nuclear Station Postaccident Liquid Sampling System Developed by Duke Power Co. January 1981.	Provides a design manual for obtaining diluted liquid and dissolved gas samples following an accident.

Table 7-1 (continued)
Nuclear Safety Analysis Center (NSAC) Reports

Report No.	Title	Synopsis
NSAC 24	"TMI-2 Accident Core Heat-Up Analysis," 1/1981	Summarizes NSAC analysis of reactor core thermal conditions during TMI-2 accident.
NSAC 25	"TMI-2 Accident Core Heat-Up Analysis - A Supplement," June, 1981	Supplements the results of NSAC 24 indicated above.
NSAC 26	"A Review of the Population Radiation Exposure at TMI-2," August 1981.	Identification and comparison of various investigations of radiation exposure of public following TMI-2 accident.
NSAC 27	"Analysis of Heatup and Pressurization During Dresden-3 Shutdown," September 1981.	Post-event analysis of unintentional heatup and pressurization of Dresden Unit 3 on December 31, 1980.
NSAC 28	"Interpretation of TMI-2 Instrument Data," 5/1982	Analysis of data from various in-core instrumentation to provide best estimate of core conditions during TMI-2 accident
NSAC 29	"The Retarding Effect of Hydrogen on Zircaloy Oxidation," Interim Report, July 1981.	Description of a combined experimental and analytic effort to identify and quantify zirconium-steam reaction rates in a core that is overheating due to decay heat.
NSAC 30	"Iodine-131 Behavior During the TMI-2 Accident," September 1981.	Analysis to determine if I-131 is released to the environment in proportion to its availability in the core following an accident.
NSAC 32	"Workshop on Hydrogen Burning and Containment Building Integrity," July 1981.	Provides details of a 2-day workshop on hydrogen burning and containment building integrity following an accident.
NSAC 59	"Simulation of Hydrogen Mixing Using the Containment Atmosphere Prediction Code," 11/1983	Description of Containment Atmosphere Prediction Code (CAP) and comparison against results of selected experiments.

Table 7-1 (continued)
Nuclear Safety Analysis Center (NSAC) Reports

Report No.	Title	Synopsis
NSAC 79	"A Limited Performance Review of Fairbanks Morse and General Motors Diesel Generators at Nuclear Plants," April 1984.	A review of steps that can be taken to improve emergency generator availability at nuclear power plants.
NSAC 87	"Plant-Specific Compared to Generic Assessment of Station Blackout," July 1985.	Comparison of the results of plant specific PRA assessment of station blackout versus alternative calculations using generic NUREG CR-3226 assessments as a starting point.
NSAC 88	"Residual Heat Removal Experience Review and Safety Analysis – Boiling Water Reactors," 3/1988.	Review of events that took place either during cold shutdown with RHR system operating or during late stages of normal plant cooldown when RHRS was placed in service.
NSAC 108	"The Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants," 9/1986.	Report of a survey of EDG success/failure experience at U.S. nuclear plants during 1983-1985.
NSAC 110	"Leak Detection in Nuclear Piping Outside Containment," March 1987.	Provides the results of experimental investigations by Wyle Labs to evaluate various methods for detecting small leaks in high energy piping.
NSAC 115	"A Risk-Based Evaluation of Emergency Response Planning," November 1988.	PRA results and methods are used for evaluating several emergency response strategies for areas surrounding a nuclear power plant.
NSAC 129	"Analysis of Refueling Incidents in Nuclear Power Plants," December 1988.	A summary of operating experience bearing on refueling incidents occurring at PWR and BWR power plants during the 1980's.
NSAC 142	"The Feasibility of Gas Turbines for Alternate AC Power at Nuclear Power Plants," January 1989.	A report on an effort by EPRI and a vendor to investigate the feasibility of using gas turbines for Alternate AC power II at nuclear power plants.
NSAC 169	"An Analysis of BWR Fuel Heatup During a Loss of Coolant While Refueling," November 1991.	An investigation of the core thermal response during a failure of the decay heat removal systems including the development of a simple analytical model.

Appendix A: Translated Table of Contents

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

Récapitulatif de la recherche EPRI relative aux accidents nucléaires

Ce document **NE RESPECTE PAS** les termes des réglementations 10CFR50 annexe B, 10CFR partie 21 et ANSI N45.2-1977 et/ou le propos de la norme ISO-9001 (1994).

Cette publication est un document corporatif, qui doit être cité dans la documentation de la manière suivante :

Récapitulatif de la recherche EPRI relative aux accidents nucléaires
EPRI, Palo Alto, CA, États-Unis : 2011.
1023403.

Résumé

Les événements qui se sont produits le 11 mars 2011 à la centrale nucléaire de Fukushima Daiichi, au Japon, suite à un tremblement de terre et au tsunami qui s'en est suivi, soulignent l'importance d'une distribution généralisée des informations disponibles dans le secteur du nucléaire, qui peuvent s'avérer pertinentes par rapport à la situation actuelle de cette centrale. Les sujets concernés par ces informations incluent les éléments suivants :

▪ génération d'hydrogène ;
▪ fuites d'alimentation hors site ;
▪ performances du cœur du réacteur suite à des fuites du liquide de refroidissement ;
▪ extraction de l'iode ;
▪ planification des interventions d'urgence ;
▪ performances du groupe électrogène diesel d'urgence ;
▪ performances structurelles des enceintes de confinement suite à un accident nucléaire.

L'accident nucléaire qui s'est produit en mars 1979 à la centrale nucléaire de Three Mile Island (Three Mile Island Unit 2) a permis de rassembler un grand nombre d'informations sur les sujets susmentionnés. Par ailleurs, la base de connaissances mise au point a été renseignée de manière exhaustive via différents projets, effectués avec la participation et/ou sous la supervision financière de l'EPRI (Electric Power Research Institute) et d'autres groupes.

L'EPRI gère un grand nombre de documents traitant de ces sujets, que ce soit au format électronique ou au format papier. Ce rapport inclut un bref résumé des documents disponibles.

Mots clés
Groupes électrogènes diesel d'urgence
Planification des interventions d'urgence
Fukushima Daiichi
Génération d'hydrogène
Extraction de l'iode
Fuites d'alimentation hors site
Performances du cœur du réacteur
Accident de la centrale de Three Mile Island

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原子力事故シナリオに適用可能なEPRI研究のサマリ

本書は以下の文書の要件を満たしておりません。
10CFR50付属書B、10CFRパート21、
ANSI N45.2-1977、またISO-9001(1994)の意図

本書は、以下の方法で文献に引用されなければならない会社文書です。

原子力事故シナリオに適用可能なEPRI研究のサマリ
カリフォルニア州パロアルト EPRI
2011年
1023403.

要約

2011年3月11日の地震とその後の津波以降の福島第一原子力発電所の事例は、発電所の現在の状況に関連した課題に対応するための原子力業界内の情報を広く配布する必要性を高めました。これらの課題には、以下のものなどが含まれます。

- 水素生成
- 外部電源喪失
- 冷却機能の喪失後の炉心のパフォーマンス
- ヨウ素除去
- 緊急対応計画
- 緊急対応計画
- 事故後の汚染構造のパフォーマンス

これらの課題に関する多くの情報は、スリーマイル島第二原子力発電所における1979年3月の事故後に収集されたものたものたもので、ナレッジベースはEPRI (Electric Power Research Institute)
やその他の参加とスポンサーシップの元で実施された研究プロジェクトにより大幅に補足されました。

EPRIは、これらの課題を対象とする資料のライブラリを電子版およびハードウェアで維持しています。利用可能な資料の簡単な抄録が、この報告書に含まれています。

キーワード
緊急ディーゼル発電機
緊急対応計画
福島第一
水素生成
外部電源喪失
炉心パフォーマンス
スリーマイル島事故

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表リスト

핵 사고 발생 시나리오에 적용 가능한 EPRI 연구 요약

본 문서는 10CFR50 Appendix B, 10CFR Part 21, ANSI N45.2-1977의 요건 및/또는 ISO-9001 (1994)의 의도를 충족하지 <u>않습니다</u>

본 출판물은 기업 문서로서 다음과 같은 형식으로 문헌에 인용되어야 합니다:

핵 사고 발생 시나리오에 적용 가능한 EPRI 연구 요약
EPRI, Palo Alto, CA: 2011.
1023403.

초록

2011년 3월 11일 이후의 후쿠시마 제2 원자력 발전소 사태, 즉 지진과 뒤따른 해일로 발전소 내에서 현재 진행중인 상황에 관한 문제들을 해결하기 위한 입수 가능한 정보를 핵발전 업계에 널리 유포할 필요성이 높아졌습니다. 이러한 대상에는 다음과 같은 것들이 주로 포함됩니다:

- 수소 발생
- 발전소 외부 지역의 전력 결손
- 냉각수 결손에 이은 원자로 코어 성능
- 요오드 제거
- 긴급 사태 대응 계획 수립
- 긴급 상황용 디젤 발전기의 성능
- 사고 발생 후 격납 시설의 구조적 성능

1979년 3월 Three Mile Island Unit 2 핵발전소의 사고 발생에 이어 이러한 주제들에 관한 많은 정보가 수집되었으며, 이후에 Electric Power Research Institute (EPRI) 및 기타 주체들의 참여 및/또는 후원 하에 수행된 연구 프로젝트들에 의해 지식 기반이 크게 보강되었습니다.

EPRI는 이 주제들을 다룬 상당량의 문서자료들을 전자문서 및 인쇄본의 형태로 동시에 보존하고 있습니다. 가용한 문서들에 대한 간략한 개요를 본 보고서에 포함시켰습니다.

주요 어휘
긴급상황용 디젤 발전기
긴급 사태 대응 계획 수립
후쿠시마 제 2
수소 발생
요오드 제거
발전소 외부 지역의 전력 결손
원자로 코어 성능
Three Mile Island 사건

목차

표 목록

Resumo da pesquisa da EPRI aplicável a cenários de acidente nuclear

Este documento NÃO atende os requisitos do Anexo B da 10CFR50, 10CFR Parte 21, ANSI N45.2-1977 e/ou a intenção da norma ISO-9001 (1994)

This publication is a corporate document that should be cited in the literature in the following manner:

Summary of EPRI Research Applicable to Nuclear Accident Scenarios
EPRI, Palo Alto, CA: 2011.
1023403.

RESUMO

Os eventos na usina nuclear de geração de energia de Fukushima Daiichi após o terremoto e subsequente tsunami de 21 de março de 2011 aumentaram a necessidade de ampla divulgação das informações disponíveis dentro do setor nuclear que abordem assuntos pertinentes à situação em curso na usina. Esses assuntos incluem, mas não necessariamente se limitam a:

- Geração de hidrogênio
- Perda de energia para fora do local
- Desempenho do núcleo do reator em seguida à perda do líquido de arrefecimento
- Remoção de iodo
- Planejamento de resposta a emergências
- Desempenho do gerador de emergência a diesel
- Desempenho da estrutura de contenção após um acidente

Muitas informações sobre esses assuntos foram coletadas na sequência do acidente na usina nuclear de geração de energia Three Mile Island Unidade 2 em março de 1979 e a base de conhecimentos foi posteriormente complementada significativamente por projetos de pesquisa realizados com a participação e/ou sob o patrocínio do EPRI (Electric Power Research Institute) e outros.

O EPRI mantém uma biblioteca significativa de documentos eletrônicos e impressos que abordam esses assuntos. Breves sinopses dos documentos disponíveis estão contidas neste relatório.

Palavras-chave
Geradores de emergência a diesel
Planejamento de resposta a emergências
Fukushima Daiichi
Geração de hidrogênio
Remoção de iodo
Perda de energia para fora do local
Desempenho do núcleo do reator
Evento de Three Mile Island

Sumário

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Resumen de la investigación de EPRI aplicable a los accidentes nucleares

Este documento **NO** cumple los requisitos 10CFR50 apéndice B, 10CFR parte 21, ANSI N45.2-1977 y/o el propósito de la norma ISO-9001 (1994)

Esta publicación es un documento corporativo que se debe citar en la literatura de la manera siguiente:

Resumen de la investigación de EPRI aplicable a los accidentes nucleares
EPRI, Palo Alto, California (EE. UU.): 2011.
1023403.

Resumen

Los acontecimientos que tuvieron lugar en la central nuclear de Fukushima I tras el terremoto y posterior tsunami del 11 de marzo de 2011 han fortalecido la necesidad de una difusión más amplia de la información disponible dentro de la industria nuclear sobre los temas pertinentes a la situación actual por la que atraviesa la central. Estos temas incluyen, entre otros:

- Generación de hidrógeno
- Pérdida de suministro eléctrico desde el exterior
- Rendimiento del núcleo del reactor tras una pérdida de refrigerante
- Retirada de yodo
- Planificación de respuesta de emergencia
- Rendimiento de generador diesel de emergencia
- Rendimiento estructural de la contención tras un accidente

Tras el accidente de la unidad 2 de la central nuclear de Three Mile Island que tuvo lugar en marzo de 1979 se recopiló gran cantidad de información sobre estos temas y, posteriormente, la base de conocimiento se ha completado con proyectos de investigación desarrollados con la participación y el patrocinio de Electric Power Research Institute (EPRI) y otros organismos.

EPRI mantiene una importante biblioteca de documentos, tanto en formato electrónico como en papel, que trata estos temas. Este informe incluye breves sinopsis de los documentos disponibles.

Palabras clave
Generadores diesel de emergencia
Planificación de respuesta de emergencia
Fukushima I
Generación de hidrógeno
Retirada de yodo
Pérdida de suministro eléctrico desde el exterior
Rendimiento del núcleo del reactor
Desastre de Three Mile Island

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