ELECTRIC POWER RESEARCH INSTITUTE

Technical Evaluation of Fukushima Accidents

Phase 2—Potential for Recriticality During Degraded Core Reflood

2016 TECHNICAL REPORT

Technical Evaluation of Fukushima Accidents

Phase 2—Potential for Recriticality During Degraded Core Reflood

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PRODUCT DESCRIPTION

This report presents some results from the second phase of the Electric Power Research Institute (EPRI) Fukushima Daiichi technical event evaluation. It focuses on gaining greater insight of the severe accidents that occurred at Fukushima Daiichi Units 1F1, 1F2, and 1F3. A study conducted by TEPCO, the operator of the Fukushima Daiichi power plant, provides the most complete assessment of the sequence of events that led to multiple core meltdowns at the site and forms the starting point for the studies described in this report. EPRI Report 1025750, *Fukushima Technical Evaluation – Phase 1*, provides an initial investigation of the event.

Background

On March 11, 2011, the Fukushima Daiichi Nuclear Power Station experienced a seismic event of historic magnitude. As a result of the earthquake, several tsunamis inundated the station. The need for rapid response to restore or maintain critical safety functions was most pressing at the three units operating at the time of the seismic event: 1F1, 1F2, and 1F3. The accidents at these units highlight the challenges nuclear power plants could face in coping with extreme events that disable systems supporting critical safety functions.

Objective

During the initial phases of the Fukushima accident, questions arose as to whether one or more of the cores in Units 1, 2, or 3 could experience a recriticality event as the cores were disrupted. At that time, recriticality was judged to be a low probability event. However, there have been no extensive investigations to definitively answer the question. The scenarios investigated in this study develop insights needed to help industry efforts to enhance reactor safety.

Enhancing the already extensive severe accident knowledge base is a key goal of the EPRI Fukushima Daiichi technical event evaluation. It is central to providing the necessary technical basis needed to strengthen industry efforts improving nuclear power plant defense-in-depth and accident management strategies.

Approach

This report is intended to supplement the comprehensive work conducted by TEPCO. It builds on the existing analyses of severe accident event progression, performed in Phase 1 of this project which employed the most recent version of the MAAP computer code: MAAP5. This assessment provides the geometric descriptions, thermohydraulics, and fissile mapping of possible core configurations that occurred at Daiichi. The project team performed Monte Carlo calculations using the SERPENT computer software to evaluate the neutronic response of various configurations and the effects of temperature and moderator materials. The team considered the presence or absence of various fissile elements and poisons. Because it will still be some time before all of the details of the forensic investigation of the accident are available, EPRI will continue Phase 2 studies on this and other relevant phenomena and will update this report as additional data become available.

Results
This report presents detailed investigation of the potential for recriticality during the Fukushima Daiichi event. It focuses on gaining greater insight through analyses of the nature of the severe accidents that occurred at Units 1F1, 1F2, and 1F3.

Applications, Value, and Use
The question of potential recriticality is important in responding to severe accidents. This study provides immediate insight into the dynamics of such events and the most likely challenges to fuel and containment integrity that occurred. This knowledge is crucial to understanding the conditions that accident management must control or accommodate, providing direct insights to guide measures to enhance plant safety. The timing of fuel melting during a severe accident and subsequent recriticality is also important in the development of advanced accident-tolerant fuels.

Keywords
Severe accidents
Accident management
Recriticality
Core damage
Reactor safety

ABSTRACT

This report presents part of the second phase of the Electric Power Research Institute (EPRI) Fukushima Daiichi technical event evaluation. It focuses on gaining greater insight through detailed analyses of phenomena important to understanding the severe accidents that occurred at Fukushima Daiichi Units 1F1, 1F2, and 1F3. The studies conducted by TEPCO, the operators of the Fukushima Daiichi power plant, provides the most complete assessment of the sequence of events that led to multiple core meltdowns at the site and forms the starting point for the studies described in this report.

This work provides an analysis of the possibility of recriticality during the three distinct BWR Mark I severe accidents at Fukushima by examining conditions at 1F2 in detail. The report examines various configurations that the fissile core materials assumed during the accident as determined by the MAAP5 computer code. The MAAP5 calculations represent the most probable reactor and primary system thermal-hydraulic responses. SERPENT calculations were performed to evaluate the neutronic response of various configurations and the effects of temperature, moderator materials, fissile elements, and poisons. The SERPENT code was chosen because of its accuracy and flexibility in modeling unusual geometries. The scenarios investigated in this study develop immediate insights needed to help industry efforts to enhance reactor safety and assess accident management strategies that might mitigate severe accidents.

EXECUTIVE SUMMARY

The Event

On March 11, 2011, at 1446 Japan Standard Time (JST), the Fukushima Daiichi Nuclear Power Station experienced a seismic event of historic magnitude. The earthquake—known as the *Tohoku-Chihou-Taiheiyo-Oki Earthquake*—originated offshore with an epicenter located 178 km from Fukushima Daiichi. This earthquake, with a magnitude of 9.0 on the Richter scale, was the largest ever recorded in Japan and the fourth largest ever recorded in the world. The earthquake and subsequent events at the Daiichi site have been extensively documented by the Tokyo Electric Power Company (TEPCO).

At the time of the earthquake, three of the six reactors at the Fukushima Daiichi Nuclear Power Station were operating at full power, while the remaining three were in various shutdown operational modes. Units 1, 2, and 3 (referred to as *1F1*, *1F2*, and *1F3* in this report) were operating at full power at the time of the seismic event. Units 4, 5, and 6 were in shutdown. Unit 4 (1F4) had been in shutdown for a reactor pressure vessel (RPV) shroud replacement since November 30, 2010. Because of the shroud maintenance work, all fuel had been removed from the RPV and stored in the spent fuel pool (SFP). Unit 5 (1F5) had been in shutdown for maintenance since January 3, 2011, but was being readied for a return to full-power operation. The fuel had been loaded into the RPV, the upper head reassembled, and the vessel pressurized in preparation for leak testing. As with 1F5, Unit 6 (1F6) was being prepared for a return to full-power operation, with fuel loaded into the RPV and the upper head reassembled.

For all operating units, the available evidence indicates that the safety systems functioned as required immediately after the seismic event. Following the loss of offsite power after the seismic event, the required emergency diesel generators (EDGs) loaded. The safety systems providing core cooling started according to design. The cooling of the SFPs at the plant was maintained. In addition, at each of the Fukushima Daiichi units, post-accident investigations have not identified any structural damage that could have compromised the reactor pressure vessel (RPV) pressure boundary, containment envelope, and SFP integrity following the seismic event. Based on the current state of knowledge, the key safety functions at the Fukushima Daiichi plant were not compromised by the seismic event itself.

The event, however, set in motion additional natural phenomena that would cause the most critical challenge to plant safety functions. As a result of the seismic event, several tsunami waves inundated the station starting at 1527 JST (41 minutes after the earthquake). By 55 minutes after the earthquake, the inundation of the plant by these tsunamis was so severe that a loss of all alternating current (ac) power occurred at 1F1, 1F2, 1F3, and 1F4. The flooding also resulted in all direct current (dc) power being lost at 1F1 and 1F2. Some dc power sources survived at 1F3. Of the five EDGs at Units 5 and 6, one air-cooled EDG for Unit 6 survived. This EDG was later used to supply power to Unit 5.

Without power, critical safety functions were either lost or significantly impaired. The loss of power together with the severity of the aftershocks and risks of additional tsunamis restricted the initial response to the accident. The seismic events and tsunami surges significantly damaged roads and associated infrastructure on and around the site. This made it nearly impossible, in the hours after the tsunami arrived, to supplement each unit's capabilities to cope with the challenge to critical safety functions caused by the loss of power.

Recent Analysis

In August, 2014 TEPCO reported that 1F1, 1F2, and 1F3 all ultimately suffered the melting of nuclear fuel in the reactor cores as the result of the loss of cooling power following the tsunami and the failure of both the off-site AC electrical power and the backup power that was being produced by diesel generators. However, not all reactors behaved identically, and cooling power was lost at different times in different reactors.

Extent of core damage

Based on various findings, it is now believed that most of the melted core fuel had been dropped from the Reactor Pressure Vessels (RPV) to the Primary Containment Vessels (PCV). The recent reevaluation was made after surveys revealed that the emergency water injection system known as "High Pressure Coolant Injection" (HPCI) had not supplied the initially estimated amount of water into the reactor. However, according to the analysis, even if all the melted core fuel had dropped to the PCV, the estimated maximum erosion of concrete mat did not lead to the breach of the PCV boundary.

TEPCO now believes that the Unit 2 pressure increase and fuel melting may have been accelerated by the injection of fire water during the onset of fuel melting. The amount of fire water injected at 1F2 during the meltdown was quite limited due to the back pressure of the system. In general, injection of (fire) water during core melting and disassembly may have important consequences, particularly for recriticality.

Scope of This Report

This report is part of the on-going Phase 2 of the Electric Power Research Institute (EPRI) Fukushima Daiichi technical event evaluation. It focuses on gaining greater insight through analyses of the severe accidents that occurred, particularly at 1F2, which assumed to envelope similar recriticality potential at the other units. The project conducted an analysis using best estimate calculations of fuel debris and the plant conditions in the period immediately after the accident and reviewed the potential for a criticality event. The project evaluated the sub-criticality of molten core configurations including the form of the fuel debris and the water volume changes in association with the meltdown of core and relocation of the fuel.

Several studies of the Fukushima Daiichi event have been or are being conducted. Among these is the work conducted by TEPCO, the operators of the Fukushima Daiichi power plant. The TEPCO work, and similar work by many international bodies including USDOE, USNRC, and EPRI in the United States, provides the most complete assessment of the sequence of events that led to multiple core meltdowns at the site during March 2011. The TEPCO investigations are further supported by computer code analyses with the Modular Accident Analysis Program (MAAP). Work by the Institute of Nuclear Power Operations (INPO) has established a timeline of events and serves as an additional source of information for developing the sequence of key events. EPRI Report 1025750, *Fukushima Technical Evaluation – Phase 1*, provides an initial investigation of the event and is the starting point for examining the potential for recriticality.

- This report serves as a parametric study as to whether damaged cores involved in the accident could become recritical. The study focuses on the details of Unit 2 and also serves as a surrogate for Units 1 and Unit 3. It is believed that the Unit 2 results will envelope the situation in the other two reactor units. The main tool used in the analysis is the Monte Carlo code SERPENT.
- This assessment of the possibility of recriticality in each of the severe accidents is based on physically plausible assumptions about the following:
  - The time history of the configuration of the core materials
  - Effects of the meltdown including reconfigured geometries, rubble beds, and admixtures of structural materials such as steel and Zircaloy.
  - Presence of control materials and other isotopes that tend to absorb neutrons, thereby precluding criticality for occurring
  - Presence of water and other moderating materials
- Using the initial input conditions provided by the MAAP5 computer code, this study provides a detailed assessment of the following:
  - The k_{eff} for each of the configurations studied
  - The likely sensitivity to changes in reactivity as materials move
  - The likely change in k_{eff} in the event that moderator (i.e. water) would be injected into the reactor as part of an attempt to cool the core and stabilize the accident situation

The scenarios investigated develop immediate insights needed to help industry efforts to enhance reactor safety. These studies are also intended to identify, through careful analysis, the aspects of observed plant behavior requiring further study. The enhancement of the already extensive severe accident knowledge base is a key goal of the EPRI Fukushima Daiichi technical event evaluation. It is central to providing the necessary technical basis to inform continual industry efforts toward identifying enhancements to nuclear power plant defense in depth.

Fukushima Daiichi Unit 2

Based on a conservative bounding analysis, this report concludes that there was a potential for recriticality in Fukushima 1F2 reactor if core reflood occurred after control blade melting has begun but prior to significant fuel rod melting. However, the MAAP analysis showed that there was insufficient time between the melting of the control materials and the collapse of the core for recriticality to have occurred, Even if water had been injected during this period the rate of injection during the accident probably would have precluded recriticality from occurring.

Although not in the scope of this report, other studies have shown that once core melting occurred, the state of the reactor became highly subcritical due to the disrupting the geometry into unfavorable configurations. This holds true even if all the core mass melted and accumulated in the bottom reactor head.

Although evolution and final location of the damaged core materials has not yet been ascertained, it is very unlikely that recriticality occurred any time during the event.

Insights of Significance to Reactor Safety

In this study it is clear that in situations where the control blade remained intact within the core, k_{eff} remained well below critical for all water levels inside the pressure vessel. However water that remained in the downcomer regions and below the core has an important effect. This is because water in these regions can reflect escaping neutrons back into the core regions. For the case when the core was under accident pressure and temperature conditions, the maximum k_{eff} was ~0.93 for a completely covered core with intact control rods.

For a hypothetical intact core that contained no control material at all, k_{eff} exceeded 1.0 (i.e. critical) for all water levels studied. Fortunately there does not appear to be a credible scenario that would lead to this situation. More likely is the scenario where a small fraction (~0.05) of the boron would remain behind in some areas of the core—and these scenarios lead to subcritical conditions. It should be noted, however, that the reactivity of the retained B10 is sensitive to the porosity and homogeneity of the eutectic materials formed during the accident and their degree of dispersal in the core region. This was not investigated in this study.

Of special value for accident management strategies, this study confirms that borated water would maintain the core in a highly subcritical state.

ABBREVIATIONS

Abbreviation	Description
1F	Fukushima Daiichi
1F1	Fukushima Daiichi Unit 1
1F2	Fukushima Daiichi Unit 2
1F3	Fukushima Daiichi Unit 3
1F4	Fukushima Daiichi Unit 4
1F5	Fukushima Daiichi Unit 5
1F6	Fukushima Daiichi Unit 6
ac	alternating current
ADS	automatic depressurization system
BAF	bottom of fuel
AM	accident management
BWR	boiling water reactor
CHF	critical heat flux
CRD	control rod drive
CST	condensate storage tank
dc	direct current
D/DFP	diesel-drive fire pump
EDG	emergency diesel generator
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
FIC	flow instrumented controller
GOTHIC	Generation of Thermal Hydraulic Information for Containments
HPCI	high-pressure coolant injection
HZP	hot zero power
IC	isolation condenser
INPO	Institute of Nuclear Power Operations

JST	Japan Standard Time
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LWR	light water reactor
MAAP	Modular Accident Analysis Program
MCNP	A general Monte Carlo N-Particle Transport Code
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MOX	Mixed oxide (fuel)
MSL	main steam line
NRC	U.S. Nuclear Regulatory Commission
ORIGEN	Oak Ridge Isotope Generation and Depletion
PCV	primary containment vessel
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RB	reactor building
RCIC	reactor core isolation cooling
RCS	reactor cooling system
RPV	reactor pressure vessel
SAMG	Severe Accident Management Guidelines
SBO	station blackout
SERPENT	A general Monte Carlo Transport Code
SFP	spent fuel pool
SOARCA	State-of-the-Art Reactor Consequence Analyses
SRV	safety relief valve
TAF	top of fuel
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island
Zr	Zircaloy

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INTRODUCTION

The Event

On March 11, 2011, at 1446 Japan Standard Time (JST), the Fukushima Daiichi Nuclear Power Station experienced a seismic event of historic magnitude. The earthquake—known as the *Tohoku-Chihou-Taiheiyo-Oki Earthquake*—originated offshore with an epicenter located 178 km from Fukushima Daiichi. This earthquake, with a magnitude of 9.0 on the Richter scale, was the largest ever recorded in Japan and the fourth largest ever recorded in the world. The earthquake and subsequent events at the Daiichi site have been extensively documented by the Tokyo Electric Power Company (TEPCO).

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Without power, critical safety functions were either lost or significantly impaired. The loss of power together with the severity of the aftershocks and risks of additional tsunamis restricted the initial response to the accident. The seismic events and tsunami surges significantly damaged roads and associated infrastructure on and around the site. This made it nearly impossible, in the hours after the tsunami arrived, to supplement each unit's capabilities to cope with the challenge to critical safety functions caused by the loss of power.

The need for rapid response to restore or maintain critical safety functions was most pressing at the three units operating at the time of the seismic event (1F1, 1F2, and 1F3). With the 1F1/2 control room and associated reactor buildings (RBs) in darkness and operators at 1F3 attempting to maintain core cooling with limited battery power, the capability to identify and maintain the condition of the reactor cores was severely compromised. With limited ways to cope with the most severe challenge to a nuclear power plant's critical safety functions, the conditions at 1F1, 1F2, and 1F3 worsened over the hours and days following the initial seismic event. The extreme temperatures and pressures that had developed inside the respective containments resulted in a partial loss of containment function. Fission products and flammable gases that had evolved during the degradation of the 1F1, 1F2, and 1F3 reactor cores were released from the containment into adjacent structures. Severe damage occurred at the site as a result of combustion of the flammable gases inside RBs, and off-site radiological releases occurred before the condition of the three severely damaged reactor cores could be stabilized. The valiant efforts of operators at the Fukushima Daiichi plant to restore cooling to the cores eventually stabilized conditions at the site over the ensuing weeks.

Conditions for Recriticality

Criticality is a nuclear term that refers to the balance of neutrons in the system. In a subcritical system the loss rate of neutrons is greater than the production rate of neutrons and the neutron population decreases with time. In a supercritical system the production rate of neutrons is greater than the loss rate of neutrons and the neutron will population increase. For accident conditions such as occurred at 1F2 criticality results in power production which can result in further damage to the system, and severe problems in accident control and mitigation.

A reactor is maintained critical during normal power operations. When an LWR shutdown, boron and other materials, which absorb neutrons, are in place to make sure that re-criticality does not occur. The added neutron absorbers substantially increase the rate of loss of neutrons, to ensure a subcritical system. If, however, during an accident, the reactor still achieves criticality after having been shut down, it is called a 'recriticality' event.

The criticality of a system can be calculated by comparing the rate at which neutrons are produced, from fission and other sources, to the rate at which they are lost through absorption and leakage out of the reactor core. For complicated geometries such as result during accident situations the current best methods include Monte Carlo simulations which statistically track neutrons to determine if their population is decreasing or growing.

In general, it is difficult for LWRs to achieve criticality, unless a fairly narrow range of conditions exist. These include a very specific geometry where a large array of fuel rods roughly 1 cm in diameter are spaced about 2.5 cm apart. Disrupting this lattice arrangement greatly reduces the chances of criticality occurring. Fuel relocation into a more reactive geometry is inherently unlikely because of the nature of commercial light water fuel. In addition, fuel needs to be surrounded by water which serves as a moderator to sustain a nuclear reaction. LWRs are deliberately 'under moderated', a safety feature. This means that loss of water will reduce the reactivity of the core, and shut down power production. Lastly the control rods and other neutron poison materials need to be removed or lost from the core region. Unless all these conditions are met, criticality would not be achieved.

At the start of the 1F2 event the core was shut down when the reactor tripped and the control rods were inserted. During the event, however, the core was disrupted, water was lost (and later re-injected), and control materials most likely melted and flowed down to lower core regions. To aid recriticality, fuel relocation would have to result in either more space available between the fuel rods or fuel pellets, or more/denser water reflood the disrupted fuel. Reactivity would be sustained if the new configuration maintained water spacing more or less evenly among the fuel pieces.

If molten fuel results, the more likely relocation geometry is less reactive because melting will tend to compress a fuel assembly, not spread it apart. If fuel material breaks up within the core region as pellets rather than rods, then the remaining fuel rod cladding and guide tubes will disturb the lattice, i.e. moderator will not be evenly shared. The same condition will occur with fuel particles, which are merely smaller "pellets", but a more even distribution of moderator is possible because the fuel particle lattice is on a smaller scale relative to the fuel rod cladding spacing.

However if, as happened at 1F2 reactor, water heats up, the temperature increase and eventual vaporization of water will tend to place the system in a subcritical condition. There are also large amounts of boron in these systems such as the control rods of the reactor. Even if the fuel does melt, the new geometric configuration will likely not be favorable for slowing down neutrons, so re-criticality is unlikely, even if water should be reintroduced to the system.

This may not be true for the probable situation at 1F2 where temperatures were high enough to first melt the boron control materials, and then the fuel cladding. Since the boron materials melt relatively low temperatures relative to the other core materials redistribution of the boron likely occurred. With the disruption of the core geometry, and lack of water in the core during this event, the likelihood of recriticality was very low. However a principal concern was what would happen when water was re-injected into the core after the disruption. Post-accident investigation of the 1F2 event has indicated that fire water injected during the meltdown phase may have affected the rate of core disruption, and therefore water may have been available in the core region at this time.

Recent Analysis

In August, 2014 TEPCO reported that 1F1, 1F2, and 1F3 all ultimately suffered the melting of nuclear fuel in the reactor cores as the result of the loss of cooling power following the tsunami and the failure of both the off-site AC electrical power and the backup power that was being produced by diesel generators. But not all reactors behaved identically, and cooling power was lost at different times in different reactors. TEPCO [Klein, 2014] recently issued a report on progress investigating the accident.

Extent of Core Damage

Based on various findings, it is now believed that most of the melted core fuel had been dropped from the Reactor Pressure Vessels (RPV) to the Primary Containment Vessels (PCV). The recent reevaluation was made after surveys revealed that the emergency water injection system known as “High Pressure Coolant Injection” (HPCI) had not supplied the initially estimated amount of water into the reactor. However, according to the analysis, even if all the melted core fuel had dropped to the PCV, the estimated maximum erosion of concrete mat did not lead to the breach of the PCV boundary.

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- This report serves as a parametric study as to whether damaged cores involved in the accident could become recritical. The study focuses on the details of Unit 2, and also serves as a surrogate for Units 1 and Unit 3. It is believed that the Unit 2 results will envelope the situation in the other two reactor units. The main tool used in the analysis is the Monte Carlo code SERPENT.

This assessment of the possibility of recriticality in each of the severe accidents is based on physically plausible assumptions about the following:

- The time history of the configuration of the core materials
- Effects of the meltdown including reconfigured geometries, rubble beds, and admixtures of structural materials such as steel and Zircaloy.
- Presence of control materials and other isotopes that tend to absorb neutrons, thereby precluding criticality for occurring

- Presence of water and other moderating materials

Using the initial input conditions provided by the MAAP5 computer code, this study provides a detailed assessment of the following:

- The k_{eff} for each of the configurations studied
- The likely sensitivity to changes in reactivity as materials move
- The likely change in k_{eff} in the event that moderator (i.e. water) would be injected into the reactor as part of an attempt to cool the core and stabilize the accident situation

The scenarios investigated develop immediate insights needed to help industry efforts to enhance reactor safety. These studies are also intended to identify, through careful analysis, the aspects of observed plant behavior requiring further study. The enhancement of the already extensive severe accident knowledge base is a key goal of the EPRI Fukushima Daiichi technical event evaluation. It is central to providing the necessary technical basis to inform continual industry efforts toward identifying enhancements to nuclear power plant defense in depth.

Assessment of Severe Accident Progression

The accidents at 1F1, 1F2, and 1F3 highlight the challenges nuclear power plants could face in coping with extreme events that disable systems supporting critical safety functions.

The subsequent discussion presents a brief overview about the likely event progression at unit 2. A detailed description of the events at each unit can be found in [EPRI, 2013].

Fukushima Daiichi Unit 2

Unlike the accident progression at 1F1, core cooling was not lost at 1F2 immediately following the loss of power across the station. Unit 2 (1F2) is a BWR equipped with a reactor core isolation cooling system (RCIC). It takes steam produced inside the RPV to drive a turbine; the rotation of the turbine powers a RCIC pump used to inject water into the vessel.

After the RCIC turbine has used the steam taken from the vessel, the steam is discharged into the suppression pool where it is mostly condensed. The suppression pool resides in containment; therefore, the decay heat removed from the fuel by this steam is discharged into containment. This increases the temperature of the water in the suppression pool in situations where this water is not cooled. An increase in suppression pool water temperature causes an increase in containment pressure. RCIC pump water can come from either the condensate storage tank (CST) or the suppression pool.

The RCIC system is designed and operated to maintain the water level in the vessel at a certain height above the fuel. This ensures good fuel cooling while the RCIC is operating normally. However, if the height of water in the vessel rises too high—to or above the level of the main steam line (MSL)—water could flood into the RCIC turbine. This could damage the turbine, causing the RCIC pump to stop working. It could also cause the turbine to rotate more slowly, reducing the flow rate of water through the RCIC pump. The automatic and operator control of this system is therefore designed to prevent the water level in the vessel from either falling too low and not removing all of the decay heat generated inside the fuel or rising too high, flooding the RCIC turbine.

The RCIC system was operated prior to the loss of power to maintain the water level in the RPV. During this period, the RCIC system automatically stopped several times because the water level in the vessel rose too high above the fuel. The system was subsequently restarted by the operator after the water level in the vessel had dropped because of continued steam generation. Just prior to the loss of all power at 1F2, the RCIC system was restarted by the operator for the last time.

Without dc power, the operators were not able to control the rate of RCIC injection to the RPV. The RCIC system control logic is designed to fully open valves that control the amount of steam that can flow from the vessel into the RCIC turbine on a loss of dc power. Operators often adjust these valves to a partially closed position to reduce the amount of steam flow into the RCIC turbine. This reduces the turbine rotation speed, which slows the RCIC pump as well as the rate at which water is injected into the vessel.

When the RCIC system functions with this much steam flow to the turbine, the RCIC pump will inject water into the RPV at a rate greater than that required to remove all of the decay heat generated in the fuel. This would have raised the water level in the vessel. Because control power was not available, the RCIC system would not have been automatically stopped due to the level of water in the vessel rising too high. The RCIC system would have continued to work until eventually the water level in the vessel reached MSL level.

The design of typical RCIC turbines would allow the system to continue functioning even with some water flooding the turbine. However, the detailed performance characteristics under the conditions at 1F2 are not clear. What is known is that the RCIC system continued to function with the water level in the vessel near the MSL for nearly three days. It was not until 70 hours after the earthquake that water injection to the vessel was lost. Some amount of seawater injection through fire engine pumps was likely restored following the depressurization of the vessel when operators opened an SRV over 5 hours later.

During the three-day period of RCIC operation, the containment pressure had risen gradually, approaching design pressure around the time at which RCIC injection stopped. However, the pressure in containment became less well controlled following RCIC failure. All operator attempts to control 1F2 containment pressure after the loss of RCIC failed. Based on the available data, venting through the wetwell or drywell did not occur. By 80 hours into the event, containment pressure escalated to twice design pressure. Site boundary radiation monitors indicated significant radiological releases from the station beyond 80 hours into the event. This rise in the observed dose rates at the site boundary provides a signature of likely 1F2 drywell head lifting. The 1F2 containment subsequently depressurized at about 90 hours; this was not the result of successful operator-initiated venting. The 1F2 containment subsequently experienced increases in containment pressure that were limited to approximately the containment design pressure; one such period of re-pressurization occurred at 95 hours.

TEPCO recently updated an inquiry into the sequence of events leading to the melting of fuel in the Unit 2 reactor core. This new evaluation concludes that the insufficient injection of cooling water into the reactor core by fire engines actually accelerated the melting. Steam generation from the injection of water at a time when the reactor’s core had been exposed caused a zirconium-water reaction, in turn generating hydrogen and large amounts of heat, causing pressure in the reactor to rise. If so, it may be that the water injection only reduced the time to complete core melting by only a few minutes, since if no water was added the core was doomed to melt anyway. In AM management space, current EOPs recommend adding water. The addition of water into a relatively intact core opens the possibility of a re-criticality issue. Of importance would be whether the water/steam/oxidation would disrupt the core before a critical configuration could be achieved.

The MAAP5 computer code simulations described in this report have assessed this sequence of events against the observed thermal-hydraulic signatures of accident progression. These simulations indicate that the following assumptions most reasonably represent the observed reactor and containment thermal-hydraulic conditions during the event:

- It is assumed that the RPV water level rose to around the level of the MSL shortly after the loss of electrical power. This would have allowed both water and steam to be discharged from the vessel into the RCIC turbine. As a result, the mass of coolant (water and steam) being lost from the vessel would have increased relative to the amount of mass being lost if only steam was being discharged to the RCIC turbine.
- The observed pressure in the vessel is represented during the majority of RCIC operation. The rate at which water and steam are discharged to the RCIC turbine was adjusted in the MAAP5 simulations to achieve this. With water and steam discharge rates to the turbine between 1½ to 2 times the maximum possible under normal RCIC operation, the best match to the observed vessel pressures is obtained.
- The water level in the vessel a couple of hours after the loss of electrical power was observed to stay at about the level of the MSL. This is achieved by adjusting the rate of RCIC water injection in the MAAP5 computer code simulations. With a rate of RCIC water injection lower than the maximum possible under normal RCIC operation by about 30%, the water level in the vessel can be maintained at about the level of the MSL.

The RCIC system is assumed to stop injecting into the vessel at about 67 hours. This is based on the observation of the pressure in the vessel beginning to increase at this time.

It is assumed that water and steam continue to be discharged to the RCIC turbine until about 70 hours. This is based on the observation that vessel pressure increases slowly until a sharp rise at 70 hours.

After operators opened an SRV at 75 hours, the vessel depressurized to containment pressure. Seawater addition is assumed in the MAAP5 simulations to begin at this point. It is assumed that the rate at which seawater was added to the vessel was insufficient to remove all of the decay heat produced in the fuel.

During RCIC operation, it is assumed that torus room flooding occurred. This would provide some cooling of the water in the suppression pool. The amount of cooling is assumed at a magnitude that best represents the observed drywell pressures during RCIC operation.

Drywell head lifting is assumed to control the pressure inside containment at about twice its design pressure. This is based on the measured drywell pressures staying nearly constant at about twice the containment design pressure between about 80 and 90 hours.

A failure of the drywell head is assumed to occur at approximately 90 hours when the 1F2 containment depressurized. This is based on the observation of a sharp rise in the dose rates measured at the site boundary at about 90 hours.

This set of physically plausible assumptions represents the observed 1F2 reactor and containment conditions very well. This is shown in Figures 1-1 and 1-2, which present the simulated and observed RPV and drywell pressure transients, respectively.

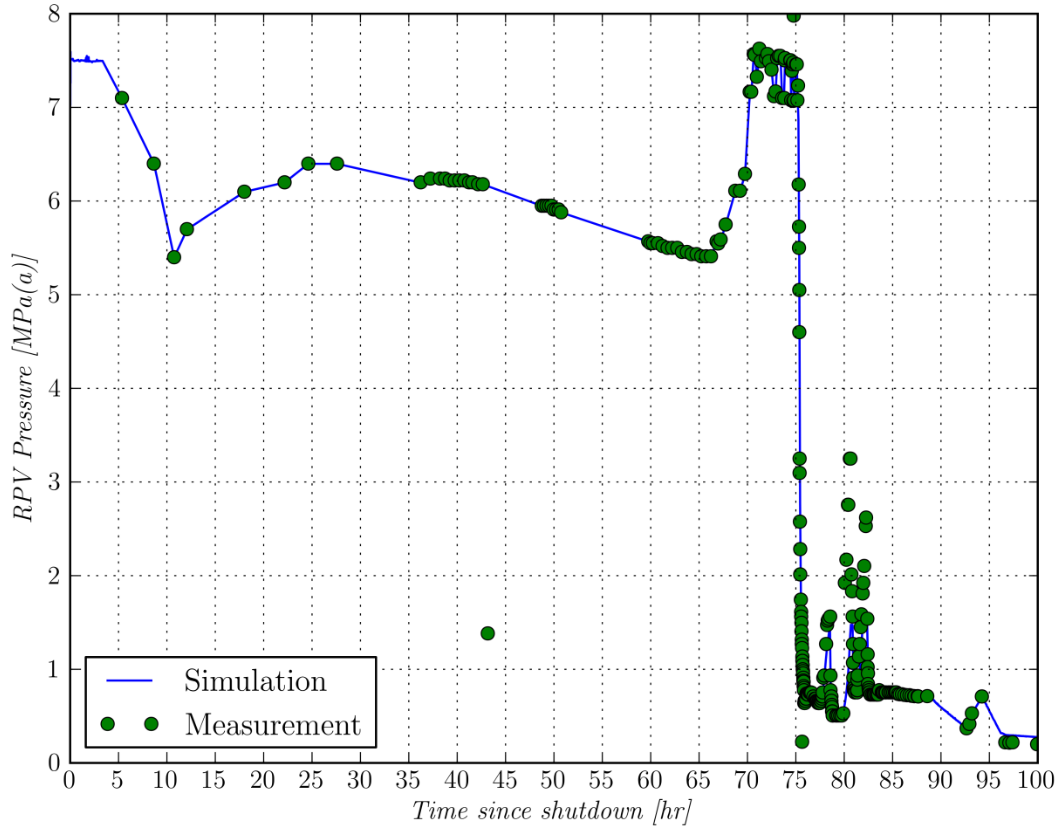

Figure 1-1
Simulation of the 1F2 Reactor Pressure Vessel Pressure Response

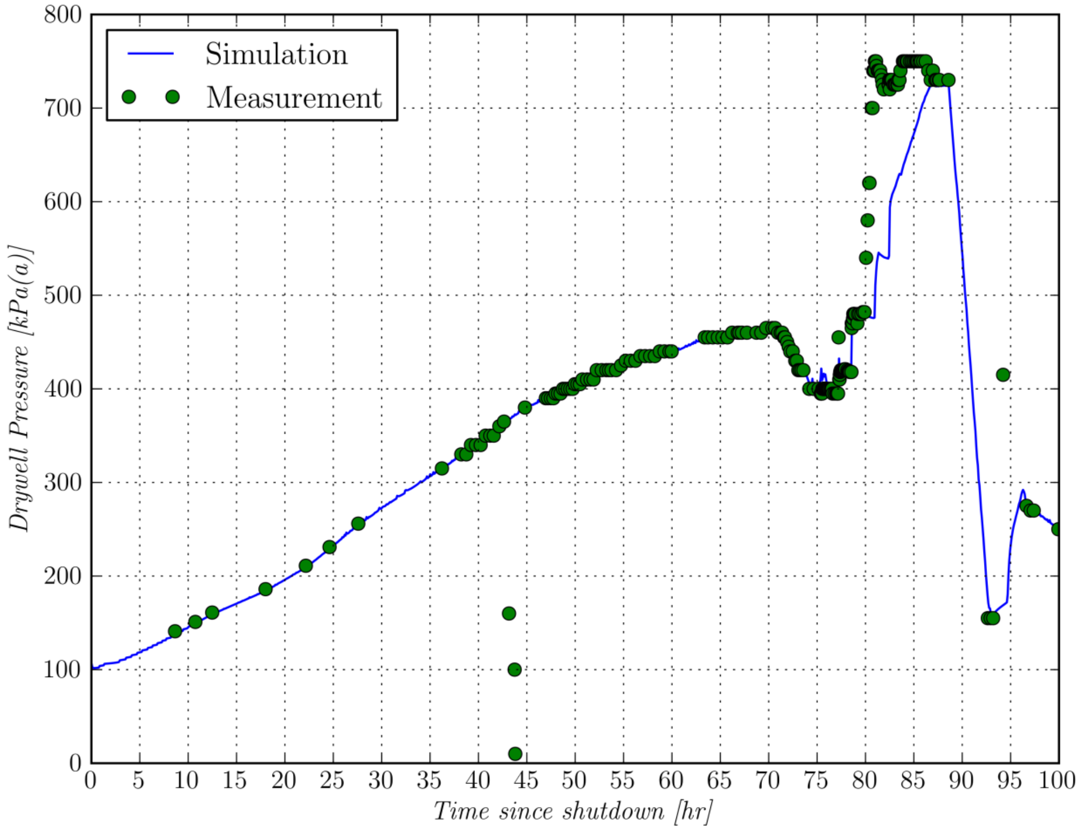

Figure 1-2
Simulation of Containment Pressure Response at 1F2

The MAAP5 simulations of the 1F2 reactor and containment response during RCIC operation show excellent agreement. The representation of RCIC operation and containment pressure suppression from torus room flooding provides a high-fidelity characterization of the reactor and containment conditions at the onset of core damage. It is therefore reasonable to draw the following insights regarding 1F2 severe accident progression from the MAAP5 predictions.

- RCIC performance:
    a) The MAAP5 simulations indicate that the RCIC system operated in a degraded mode that maintained the fuel cool for nearly 70 hours.
    b) This is the single feature of the accident progression at 1F2 that led to a delay in the onset of fuel damage by more than three days.
    c) The RCIC system may be more robust than is typically assumed in analyses and emergency response procedures.
- Extent of core damage:
    a) After RCIC stopped operating, decay heat removal from the fuel was partially restored through the use of fire engine pumps to add seawater to the RPV.

b) It is not known how much of the seawater discharged from the fire engine pumps was actually injected into the vessel. TEPCO now estimates that only 10-20 percent of the cooling water sprayed from the trucks actually reached the reactor core. Data on the water pressure and flow rate from the fire trucks is limited, preventing accurate estimates of the water volume.

c) The water level in the vessel was high at the time RCIC stopped injecting. There was also not a long time before the vessel was depressurized. To account for the type of off-site release of volatile fission products (such as Cs), it is possible that the amount of seawater added to the vessel in the day following RCIC failure was not sufficient to remove all decay heat.

- Drywell head flange integrity:

a) At about 90 hours, a sudden drop in containment pressure to nearly atmospheric pressure occurred. This was accompanied by a sharp rise in measured site boundary dose rates and may have resulted from failure of the drywell head flange. As the pressure in containment drops after 90 hours, the MAAP5 simulations indicate that the hole in the drywell head flange might have reduced.

b) At about 95 hours, the containment pressure began to rise again. It reached approximately the design pressure of containment before it began to decrease again. The MAAP5 simulations indicate that the drywell head flange seal is likely to have reopened to compensate for this increase in pressure at around 95 hours.

c) This is a reasonable indication of damage to the elastomeric seal of the drywell head flange. Such damage would prevent a complete resealing of the drywell head flange as the containment pressure drops.

2
SUMMARY OF LIGHT WATER REACTOR RECRITICALITY PHENOMENA

2.1 Introduction

During normal operation the reactor core is maintained in either a shutdown or an operational (critical) state. In the former, the reactor is shut-down by inserting control rods into the core which absorbs neutrons, terminating power production. In an operational state, the number of neutrons produced is just equal to the number of neutrons consumed in nuclear reactions. To measure criticality, the symbol *k* is used. *k* is the ratio of neutrons born in one generation to the neutrons born in the previous generation. When k =1, the reactor is referred to as "critical", and fission reactions can be sustained indefinitely. If *k* is greater than 1, then power will increase. If *k* is less than 1, then power levels will decrease, and the reactor is said to be subcritical. Thus, the stability of the system is dependent on *k*.

Neutron emitted as part of the fission process may be divided into two classes: "prompt neutrons" and "delayed neutrons." Prompt neutrons appear at the moment of fission – within 10^{-14} sec. They constitute 99% of all fission neutrons. Delayed neutrons make up about 0.65% of the total, and appear as products of the radioactive decay of fission fragments. They are emitted with gradually decreasing intensity over a period of several minutes after the actual fission event. They have an importance much greater than their relatively small number, in that they control the rate of power increase in a critical reactor configuration. For situations where k>1 the power increases exponentially, relatively slowly if the rate is being controlled by the delayed neutrons, but almost instantaneously if *k* exceeds this value. The latter is referred to as "prompt critical," resulting in very high power levels in milliseconds. If such a situation occurs in a reactor situation, the power range at some point becomes self-limiting in that there are various feedback mechanisms that will terminate the power excursion. However if the feedback mechanisms are not sufficiently fast acting, major reactor damage will result. What is important is not so much the peak power level reached in an excursion, but the total amount of energy released.

In a reactor core, the configuration of the fuel assemblies, their geometry, uranium concentrations and enrichments, moderator materials and presence of control materials (B_4C in the case of a BWR) are all carefully managed. It is the precise configuration of these elements that determine the reactivity of the reactor core. Power reactors are normally designed such that if one or more of these parameters are changed, the ability to sustain a critical state is decreased. The reactivity of all these factors taken together is usually referred to as "k effective", or k_{eff}.

However, during a reactor accident such as occurred at Fukushima, where bulk fuel melting was likely, the question is whether the melted fuel could possible reform into a configuration that could be critical, thereby adding considerable amount of additional heat energy to the already damaged reactor system.

This situation is referred to as "recriticality." Such questions arose during the days following the event, although it is currently deemed unlikely to have occurred, based on the analysis provided by [Mosteller, 1995]. Nevertheless, it is important to understand if such a recritical situation could have been possible, if the accident had evolved differently. Therefore this investigation is evaluating how close to critical the core of Unit 2 was during various melt configurations during the event. In addition, during the accident the question arose as to the effects of adding water (a moderator material) into the damaged core. This question is important, since adding water is the primary way of cooling a reactor, and is important in accident management strategies.

2.2 Reactivity and Reactivity Feedback Mechanisms

As discussed above a reactor's kinetic behavior is the result of delayed neutrons in the fission process, but the k_{eff} is determined by the geometry of the materials in the core. Other effects, such as temperature, are also important. The re-distribution of the melted core materials also affects the absorption and leakage of neutrons. There are four main considerations:

1. Poison Concentration – including the effect of control rods, xenon buildup, burnable poisons and boron concentrations. In the Fukushima event, seawater was used to cool the reactor; various elements in the seawater itself, such as chlorine, will tend to poison nuclear reactions. If melted core materials, referred to as "corium," also entrain structural elements such as steel, these materials will also act as neutron absorbers, poisoning the reactivity.
2. Doppler Coefficient – At operating temperatures the Doppler effect is an important mechanism for capturing neutrons, which limits power excursions. As temperature increases, k_{eff} decreases. However as core temperatures decrease, the opposite is true.
3. Coolant Temperature – In BWRs, water is the coolant used, and is also the moderator material. Injecting water into a damaged reactor adds reactivity. k_{eff} is in part determined by the density of the moderator. The lower the temperature, the greater the water density will be.
4. Coolant Pressure – As power in a reactor increase, void formation (boiling) in the moderator may occur. In a damaged reactor voiding may occur as the result of steam flashing if the primary system is damaged, or as a result of the highly exothermic Zr-steam reaction that would probably result as a damage core heated up from decay heat. Both probably occurred in the 1F2 event. As the voids expand, less moderator will remain in the core, limiting the reactivity and a power excursion. The void formation is a function of system pressure.

A fundamental requirement for a reactor core to be subcritical is for k_{eff} to be less than 1, with some margin to spare. This amount of reactivity k_{eff} is below 1.0 is referred to as shutdown margin.

There are time domains in this recriticality study. The first is the immediate period following the loss of all cooling to the core. The control blades consist of B_4C sheathed in stainless steel. Because this composite material forms a relatively low melting point eutectic, the possibility that some, most, or even all, of the boron responsible for controlling the k_{eff} of the core could flow downward because of gravity, in effect introducing positive reactivity into the relatively intact core, resulting in a secondary critical configuration.

Another situation occurs after core melting. First appearances would indicate that recriticality would be unlikely simply on the basis that any core disruption will depart from the narrow geometry configuration necessary for criticality during normal operation. On the other hand, it is theoretically possible for ~3% enriched uranium, as is in the 1F2 reactor, to become critical under sub-optimum moderator/reflector conditions. It is theoretically possible that a few completely clean, moderated fuel assemblies in proximate positions could achieve a critical state. However, based on the energy balances obtained in the EPRI Phase I study, that recriticality in 1F2 was improbable. Nevertheless it is an important goal of this study, to determine the shutdown margin (how close to criticality) during various times and core configurations in the accident.

2.2.1 Effects of Xenon and Samarium

Nuclear reactors are sensitive to a process known as xenon poisoning. Xenon-135 is an isotope produced in the fission process. This isotope strongly absorbs neutrons and therefore acts as a poison. In normal operation, xenon-135 is controlled by continual neutron capture, which produces a stable, but relatively low concentration. After a reactor is shutdown, the xenon-135 builds up in the core. The main source is through a 2-stage decay of tellurium-135 which decays into iodine-135, which in turn decays (with a half-life of 6.7 hours) to xenon-135. When the reactor is shut down, iodine-135 continues to decay to xenon-135, making criticality difficult. This is referred to a xenon poisoning. This poisoning effect quickly builds up to a maximum about 12 hours after shutdown. The higher the power level (i.e., neutron flux) of the core prior to shutdown, the greater will be the xenon poisoning. In an accident situation, this poisoning can be a very serious impediment to criticality in the time period a few hours after shutdown, but gradually decreases with time. However, in situations where the core melts, xenon, a noble gas, can be released from the core region and may be of negligible importance. Other fission products can have a similar, though smaller, poisoning effect, notably samarium-147, an isotope produced from promethium-149, having a half-life of 47 hours. The ability to insert enough excess reactivity to overcome the effects of Xe and Sm is call xenon override. In terms of the accident at Fukushima or other similar event, these reactions may provide a safety window of several days. Eventually however the Xe and Sm decay, and are no longer effective in holding down k_{eff} in the core.

In a core damage accident, there will be a gap release of ~100% of the Xe in the pellet-clad space upon clad ballooning and rupture. A small percentage of the Xe is actually in the gap; the majority ~80-95% of the Xe will reside in the fuel matrix depending on the burnup. Higher burnup fuels tend to have the lower amount retained in the matrix.

The concentration of Xe will be determined by 1) the power history prior to shutdown, and 2) the I-135 to Xe-135 decay, with maximum Xe poisoning about a day into the accident (i.e. reactor scram). During power operations, the Xe-135 and I-135 isotopes are continually produced and decay. However due the short half-live of each, the concentration at the time of an accident would be determined by the operating power history in the few days to a week or so just prior to the event.

2.2.2 Effects of Retained Boron

The dynamics of the BWR accidents involving core melting show that in high temperature transients, materials with lower melting temperatures than the fuel or the cladding can liquefy early, and candle down to lower locations in the core. This would surely be the case with the BWR control blades which are cruciform structures of B_4C encased in steel. The relatively low melting point of the B_4C, aided by a eutectic formation with the steel, results in a significant fraction of the B_4C will flow downward to the lower parts of the core, leaving the core devoid of a control material capable of maintain the core subcritical.

The interaction of boron carbide with steel is verified as initiating effect and the strong attack on the channel box walls was demonstrated in a series of CORA experiments, particularly in QUENCH-07. Most of the relocated boron carbide is transported as component of melts, some as embedded particles or by free falling. Nevertheless, some residual boron carbide remains at place, held by oxidized residual melt or thin scale residues. Stable compounds are expected to precipitate from absorber melt in the form of ZrC, ZrB_2, $(Cr, Fe)_2B$ and (Cr, Fe) carbides. These compounds may form solid particles.

With respect to destruction by absorber material the interaction of boron carbide with Steel, CORA-16 verified the strong attack on the channel box walls is demonstrated. Most of the relocated boron carbide is transported as component of melts, some as embedded particles or by free falling. Residual boron carbide remains at place, held by oxidized residual melt or thin blade scale residues.

In CORA-16, the absorber rod and channel box walls failed (literally disappeared and relocated) over a significant length (~1 m.). Cross sections of the experimental array showed intact steel blades with missing B_4C above and below. The absorber was almost completely gone. Control rods and grid spacers were still in position. Under similar situations in an accident it appears that recriticality a possibility.

Prior studies have shown that a relatively small fraction of B remaining behind in the disrupted region can provide negative reactivity that needs to be taken into account in a criticality evaluation.

2.3 Core Configurations and MAAP5

A number of analytic studies of the Fukushima accident have been produced. Most notable are the Sandia National Laboratory MELCOR studies or the EPRI MAAP5 work. MAAP5 is a fast-running computer code that simulates the response of light water moderated nuclear power plants. It provides a useful tool for analyzing the consequences of a wide range of postulated plant transients and severe accidents for current plant designs and Advanced Light Water Reactors (ALWRs). The code predicts the progression of accident scenarios and it can predict the occurrence of vessel failure and model the containment performance with successful debris cooling or pressurization of containment.

The MAAP5 work forms the basis of the recriticality analysis in this report. MAAP5 code represents the best available industry tool for analyzing plant behavior in core damage scenarios. MAAP5 provides a critical guide to interpreting the accident data and physical phenomena that were involved. As such, it provides immediate insight into the physical progression of events and the most likely challenges to fuel and containment integrity that occurred. This knowledge is crucial to understanding the conditions that accident management must control or accommodate, providing direct insights to guide measures to enhance plant safety. An overview of the Fukushima analyses is in the EPRI Report 1025750 “Fukushima Technical Evaluation.”

2.4 Consequences of Criticality in a Damaged Reactor

The two-fold focus of this study is to answer the questions 1) was 1F2 reactor, during the accident, ever in a configuration where is was critical or close to critical, and 2) if water (demineralized, borated, or sea water) was injected could criticality have been achieved. These questions would be important not only to understanding the Fukushima events, but adds additional insights important to managing severe accidents. Because recriticality was not achieved in the 1F2 event, post-criticality consequences were considered only briefly. A detailed investigation would require an analysis of considerable depth. Nevertheless some insights were obtained and are briefly discussed here. MAAP results that form the basis of our analysis predict the complicated thermal-hydraulic phenomena involved in quenching an overheated core during reflooding, but including such results to the recriticality analysis were outside the scope of this study.

Much of our insights were obtained based on various destructive experimental projects and analyses by others. Oak Ridge [ORNL, 1985] investigated heat transfer correlations and models relevant to potential recriticality situations.

Our analysis determined that the damaged core configuration of core mostly intact with control rods melted could possibly become recritical if water was injected into an intact pressure vessel in the brief interval at the start of core melt. This interval may be as short as 5 or 10 minutes. If electric power returns during this time-window, or if external pumping is established, unborated water will start to reflood the control rod free core. Recriticality might take place for which the only mitigating mechanisms are the Doppler effect and void formation.

The assumption we made to come to this conclusion was that virtually 100% of boron had left the core region. Although total boron relocation was predicted in the MAAP calculations used to determine core geometries over time, in reality small amount of boron or boron-steel eutectic may remain be behind. In a previous paper the authors determined that, because of boron self-shielding effects, retaining a few percent boron in the core may be enough to suppress criticality in a virtually intact core.

2.4.1 General Considerations

For core melting to occur, a loss of the moderating water inventory through a hole in the PCS was required. Depending on the size of the hole, the PCS pressure may be fairly elevated. To reflood the core, the lower portion of the RPV would need to be intact. In the case of 1F2, the pressure was around 7 MPa during the initial melting phase (but rapidly dropping soon after). If cold water would be injected at this point an increase in PCS pressure would be anticipated due to flashing of the water as it reached hot portions of the vessel, lower support structures, and partially intact core. The water (moderator) injected will reduce the fuel temperature in any covered regions, while the balance of the fuel remains at elevated temperatures. In water flooded regions the temperature will also be determined by the system pressure. Thus the combination of system pressure and local temperatures will affect recriticality. The former will tend to collapse any voids in the system, while the latter reduce the local Doppler broadening. Moreover steam cooling probably would be reestablished in the upper (uncovered) core regions. Both these produce a positive reactivity effect.

The steam phase will contain substantial water entrainment. This may result in a slight positive reactivity insertion due to the moderating effects of steam. However, it will contribute to the cooling of the fuel cladding from the bottom up. This two phased region contributes to the rate criticality approaches, and the power excursion. At some collapsed water level, criticality may occur. If the water injection rate is not too high, criticality, if it were to happen, would be oscillatory in nature. The process becomes very complicated at this point, with various feedback mechanisms involved, both positive and negative. Quenching of the fuel is an example of positive feedback, while vapor bubble formation would be negative, and pool swelling would increase the effective volume of the region approaching criticality. The most likely outcome most likely would be either a rapid boil-off of moderator, or a mechanical disruption of the fuel configuration. Both would result in a subcritical state. A pressure spike in the PCV would be almost inevitable. A detailed analysis is beyond to scope of the report since recriticality did not occur in 1F2.

Reactivity will depend strongly on the void fraction. For the 1F2 fuel with middle of life burnup**,** recriticality is possible if no control materials are present in the core. A core with no water probably would not lead to criticality, if there is no water in the downcomer regions. Prior studies have shown that relatively small amount of boron control blade materials, if retained in the core, could prevent recriticality, because the self-shielding effect of the boron on itself would be no longer operable. As a result of the above consideration, recriticality is unlikely in any of the voided regions of the core, although it may be possible, under the right conditions, in core volumes containing water nodes.

The amount of control rods materials remaining in the core and the rate of water injections during reflooding are important crucial factor for the timing of recriticality and for the subsequent power surge.

2.5 Other Investigations

2.5.1 NSAC-1

A post-accident analysis of the Three Mile Island Unit 2 accident investigated the potential for recriticality [NSAC, 1980]. The analysis concluded, based partly on neutron monitor data, that the reactor was less reactive immediately after shutdown due to homogeneous voiding. Although a PWR, the axial void fraction that occurred during this period mimicked BWR behavior. The k_{eff} with the (borated) water level down to two feet above the core plate was ~0.88. Recriticality was found unlikely. However, the report also concluded that recriticality was possible given complete control rod and burnable poison rod destruction/removal.

2.5.2 CORA

CORA Experiments [CORA, 2009] were carried out at the KfK research Laboratory in Germany. The experimental program provided information on the failure mechanisms of LWR fuel elements in the temperature range 1200 – 2000°C. BWR-type bundles consisted of 18 fuel rods and neutron absorber rods with boron carbide, surrounded by a Zircaloy channel box. The tests were run under steam environments typical of a severe core damage event.

2.5.3 KAERI

In 2012 following the Fukushima event, KAERI performed calculations to specifically investigate the severe core damage accident at the Fukushima, the criticality level for using the MCNPX code. The analysis was done for a rubble pile with various metal/water ratios with no boron or burnable poisons.

The potential for criticality was analyzed by varying the total amount of corium in a rubbelized bed. The highest criticality was found for a corium packing faction of 30%, regardless of increasing the total amount of corium. The adequate soluble boron concentration was also evaluated needed to ensure subcriticality of the reactor core. It was found that at around 3,000 ppm H_3BO_3 was required to assure a subcritical condition

2.5.4 NRC (NUREG 5653)

A 1990 study by the NRC [NUREG, 1990] on the consequences of recriticality in BWRs concluded that without the control blades, relatively high reactivities are possible with standing fuel rods or over a broad range of fuel particle sizes and fuel volume fractions for both unborated and fairly heavily borated reflood conditions.

For the conditions studied, the analysis also indicated that a maximum power excursion produces a fuel enthalpy of 73 cal/g, corresponding to a temperature rise of 1300°F in the fuel. Doppler feedback was the principle mechanism for terminating rapid transients in low enriched uranium-water systems.

If the reactor remained critical following an initial excursion at the time of reflooding (i.e., reflood is conducted without boration), it will either enter an oscillatory mode in which water periodically enters and is expelled from the core or it will approach a quasi-steady power level. In either case, the average power level achieved will be determined by the balance between the reactivity added and the feedback mechanisms. This study, concluded a recriticality event is likely to produce core power levels less than about 20% of normal power (and probably not much more than 10% of normal power), but may be significantly above the decay heat level.

A recriticality event will most likely not generate a pressure pulse significant enough to fail the vessel. Instead, a quasi-steady power level would result and the containment pressure and temperature would increase until the containment failure pressure is reached, unless actions are taken to terminate the event.

2.5.5 SARA Project

A 1999 Swedish study, called SARA (Severe Accident Recriticality Analysis) [Frid, 2001], performed a detailed analysis of a BWR Mark I reactor with control material melted out of the core region. The study found that recriticality was predicted:

> “for the studied range of parameters, i.e. with core uncover and heat-up to maximum core temperatures above 1800 K, and water flow rates 45 to 2000 kg/s injected into the downcomer. The criticality arrives earlier with high than with low flow rates since the time to reflood the core up to a critical water level then is shorter. …. The recriticality takes place in the central control rod free part of the core around and below the quench front, where the void fraction is low enough

> for moderation. Since only a small fraction of the core becomes critical the power density there can be considerable…. The codes predicted recriticality with a first super prompt power peak and then a more or less stabilized power level for the applied range of ECCS injection flow rate from 160 to 1350 kg/s."

The authors of the Swedish report also found that the initial high but short duration power peaks due to recriticality have minor effect on the containment, since the total energy in the peaks is small. However, after a sufficient period of time, the steaming could eventually challenge filtered containment venting systems. Thus the prevention of recriticality is crucial in accident management. Contrary to the Swedish study, the current MAAP predictions are for the top core node to melt first, and the B_4C to candle downward. Concerning the quasi steady-state power levels, the SARA results are largely in agreement with earlier studies, i.e. the stabilized power seems to be below 20% of the nominal power for reflooding rates in the range of 90- 1350 kg/s.

The SARA project also showed the sensitivity of recriticality phenomena to thermal-hydraulic modelling, the specifics of accident scenario, such as system pressure and distribution of boron-carbide in the core, and the importance of multi-dimensional neutron kinetics for the determination of local power distributions. Equally important is the ability to model the entire BWR primary system as realistically as possible in order to capture the reactor power - primary system behavior feedback effects, which is now possible in the current study.

2.5.6 NKS Work

NKS (Nordisk Kernesikkerhedsforskning) and EU projects [NKS, 1983] considered reflooding by the ECCS system. It examined coolant re-entering the core due to melt-coolant interaction in the lower plenum, specifically the relocation and fragmentation of the molten control rod metal. This could cause the level swell in the core. Another possibility of a steam explosion in the lower head led to a prompt recriticality peak. In this instance, a water slug entrance into the core would be so violent that the fuel disintegration may occur. After the large power peak water was rapidly pushed back from the core, no semistable power generation was found.

2.5.7 Other Studies and Data

Recent work in Japan [Sato, 2012 and Izawa, 2014] have studied recriticality events. Still other insights might be gained from studies performed on BWR rod drop accidents. For hot zero power (HZP) conditions, effectively all the power is deposited in the first few milliseconds of an accident.

The results indicate a sharp power increase that is stopped by the Doppler coefficient. After the initial power reduction, a clear power tail is calculated. The generation of voids is significant and therefore the reactor power is finished by the void reactivity feedback. Quenching actions include thermal expansion, boiling, ^{238}U Doppler effect, and radiolytic gas bubble formation. Experimental evidence shows expanding void space, consisting of many very small bubbles (microbubbles) with internal pressures of from 10 to 1000 atmospheres, is created by the fission process. [LASL, 2000]

In complicated geometries other mechanism are also operable: heating and density change of the water; heating of the core structure, including its own geometry changes and moderator expulsion from such changes; and finally, the boiling of water next to fuel pins and loss of moderator when water is expelled from the core. Moreover in situations where the reactive volume is small, neutron leakage is quite high.

There may also be pressure spikes that are capable of further core disruption of the already damaged core. In open two-phase BWR systems, the effects of pressure changes are even more important because there is a greater change in moderator density for a given change in system pressure. For partially pressurized systems, large void expansion is possible.

There are also some desultory experiments and reactor accidents, in particular, the destruction of BORAX, SPERT, and SL-1. Although these systems were very different from a BWR reactor, the general progression of events might shed some light as to the consequences of recriticality. The terminators for these cases included heating and density change of the water; heating of the core structure, including geometry changes and moderator expulsion. Very rapid transfer of energy occurred, it happened before any significant volume change took place, and the resulting high pressure destroyed the cores.

2.6 Evolution of Events at 1F2

At 1F2 the sequence of events stating at the time of initial core uncovering is given in Table 2-1. These results were obtained from the MAAP5 calculations. According to these calculations, once the water reaches the top of the active fuel slightly after 3 days into the accident, events move fairly rapidly. Roughly an hour later, about 60% of the core is uncovered, and 30 minutes after that 80% core uncovery is reached. Somewhere around this point, steam cooling of the core from steam generated in the lower reaches of the core volume starts to become ineffective. A very rapid temperature excursion starts when the Zircaloy – water reaction starts in earnest.

MAAP calculates a period of about ~30 minutes between when the control blades started to melt and the fuel collapsed into a rubble bed. Initially the boron control materials melt and candle downward. A large fraction of the boron is still within the core. However there is a very brief period of about 10 minutes just prior to core collapse when all or nearly all of the boron is predicted to have left the core. This brief time is when the greatest potential for recriticality would exist if water were injected in the core without a soluble poison, such as borated water or seawater. It is also worth noting that during this period only a relatively small fraction of the core would be without any the control blade materials, so there probably would be a very large neutron leakage factor.

Table 2-1
Timing of Core Uncovery

Time (sec)	Incremental time	
0		Water reached Top of Active Fuel
4100	1 hr 8 min	60% of fuel uncovered
5900	30 min	80% of fuel uncovered, acceleration of Zircalloy-H20 reaction
7500	26 min	Start of Control Rod melting
7700	3.3 min	CR candling begins in 40% of central core region
8300	10 min	Water reaches bottom of active fuel
8500	3.3 min	CR missing middle 40% core regions of rings 1-3
8800	5 min	CR missing middle 60% core regions of rings 1-3
8900	1.5 min	start of Fuel collapse

Since the core might become critical if water were to be injected during this limited time period, the question would be if sufficient water could be injected into the vessel before the core completely melted and the relatively regular geometry of an intact core remained. Once the core melts down the geometry is much less favorable for a recriticality event.

MAAP results have shown maximum water injection rates of about 600 kg/min (usually much lower) for the functional RCIC system during the event, just prior to core uncovery. This corresponds to a rate of about 0.6 m^3/minute, into a core volume of about 72 m^3. The potentially higher ECCS flow rates were not achieved in the 1F2 event.

When water from the fire engines was used in the event, analysis showed an injection rate of water actually delivered to the core that required about 1-2 hour to fill the RPV to about half the core height. Note that the assumed injection rate is consistent with the total amount of water discharged from the fire engine, adjusted for leakage (i.e. difficulty in fire engine injection against RPV back pressure). Uncertainties still remain as to how much water was actually injected into the RPV given the less than ideal conditions associated with this emergency injection through the fire water system.

3 ANALYSIS OF FUKUSHIMA DAIICHI UNIT 2 EVENT PROGRESSION

Introduction

Detailed Calculations of the 1F2 core were performed using the Monte Carlo SERPENT code to evaluate the neutronic response of various configuration states of the damaged core. The reactivity is highly sensitive to the geometry of the core material, and the presence of various moderators and poisons. The reactivity is also sensitive to the porosity and homogeneity of the eutectics formed by the molten core materials (corium).

3.1 SERPENT

The continuous-energy Monte Carlo method has been used for criticality safety analyses, radiation shielding and dose rate calculations, detector modeling and the validation of deterministic transport codes for decades. The main motivator is usually the need to model geometry and interaction physics to within maximum accuracy, often regardless of the computational cost. Monte Carlo codes are well suited for the job, with the capability to handle complicated three-dimensional geometries and to model neutron interactions at the microscopic level without major approximations.

SERPENT physics is based on ACE format data libraries, mainly because the ENDF reaction laws are reasonably well documented, and the same data format is used by MCNP, which makes code validation easy and straightforward. To simplify the calculation routines, SERPENT does not use the continuous-energy cross sections directly, but to reconstruct the data on a master energy grid that was used for all nuclides. This approach is very efficient as well, since time-consuming grid search iteration is reduced to minimum.

For the sake of simplicity, SERPENT uses the Woodcock delta-tracking method for neutron transport. The tracking routine does not involve the calculation of optical distances to boundary surfaces, which considerably simplified the implementation of the geometry routine. This method turns out to be reasonably well suited for lattice calculations. The main drawback is that the efficiency of the basic delta-tracking method is reduced when localized heavy absorbers, such as control rods or burnable absorber pins are present in the geometry

The Monte Carlo method is a computational algorithm that relies on repeated random sampling to obtain numerical results; i.e., by running simulations many times over in order to approach the correct solution. SERPENT is used in situations where the size of the problem (i.e., complex geometry, physical parameters such as multiple isotopes, materials, etc.) makes the problem difficult to solve using closed forms or analytic approaches. A user can apply the code to quite complicated problems almost without any geometric approximations and get accurate results in a reasonable time when having modem workstations or PCs.

3.2 Core Geometry

In principle, the determination of criticality should be made on a detailed physical description of the entire reactor core (and RPV) volume. However a three-dimensional calculation for the entire volume of interest is extremely complicated and it is extremely time consuming to obtain acceptable statistics. Moreover, except for the very early times in the core melt sequence, the precise geometry is highly uncertain. Meaningful indications of the likelihood of criticality can be obtained from far simpler representations of the geometry—ones consistent with our current knowledge of the actual positions of the corium throughout the accident. The precision of the answer required is fairly low, since the goal was to see if any of the likely configurations would approach a critical state.

Very often symmetry is used to reduce the computational time required to time needed to obtain a solution. In the current analysis several things were done to reduce the memory requirements for the Monte Carlo model. The first one was to impose quadrant symmetry. Most BWR cores are approximately quadrant symmetric with rotational symmetry (i.e., the east line of symmetry corresponded to the north, south, and west lines of symmetry). SERPENT handles rotational symmetry, so that simplification reduces the number of cells and compositions by 75%.

Another strategy was to eliminate those isotopes that produce very slight or negligible reactivity effects. Isotopes to eliminate are determined from a two-dimensional, room-temperature calculation for a single representative (intact) bundle with an isotopic composition at the representative burnup at the time of the accident.

For cases where the corium pooled in the lower hemispherical region of the RPV, unit cells consisting of molten corium surrounding one of the control rod drive mechanism tube was employed. This was a square prismoid with various layers of metallic and oxicidic corium surrounding a central cylindrical tube.

The response from these configurations provides a good indication of the likely behavior of the corium, and can be achieved at a reasonable amount of computational effort. Each of the SERPENT cases was run to a total number of particle histories sufficient to reduce the standard deviation in the eigenvalue to ~0.16 %, which is more than enough for the purposes of this study.

It was expected that once serious core melting commenced, that the core reactivity would quickly decrease due to its departure from the optimal geometry engineered into a commercial reactor core. This assumption was based on an earlier study [Mosteller, 1995] and was later confirmed by the analysis.

Monte Carlo calculations require a separate cross section library for each combination of isotopes and temperatures. At the time of the accident, the fuel had significant burnup and therefore significant concentrations of fission products. The number of isotopic libraries was quite large, so the first step, consistent with the degree of accuracy required in the calculation, was to ignore these that make a small contribution to k_{eff}. Another important assumption was how the fission products behaved once melting commences. While most fission products will follow along in the corium, some, particularly the Xe-135, are volatile, and will be released from the core regions either as a gap release as the cladding ruptured, or from the UO_2 matrix as the fuel melts.

3.2.1 Lower Plenum Geometry

The BWR reactor vessel below the elevation of the core plate forms a lower plenum region. This is a hemispherical section of radius ~319 cm. A large fraction of the volume immediately beneath the core plate is occupied by the control rod guide tubes. Source, intermediate, and power range detector assemblies also transect this region.

There are typically 200 bottom head penetrations as necessary to accommodate the control rod drive mechanisms, instrument guide tube penetrations, and a drain line penetration near the low point of the bottom head. The general arrangement is shown in Figure 3-1.

The control rod drive mechanism assembly and instrument guide tube penetrations are stainless steel. The stub tubes are Inconel. In a reactor accident, most analyzes predict corium attack in this area would first fail the vessel penetrations and not the bottom head itself.

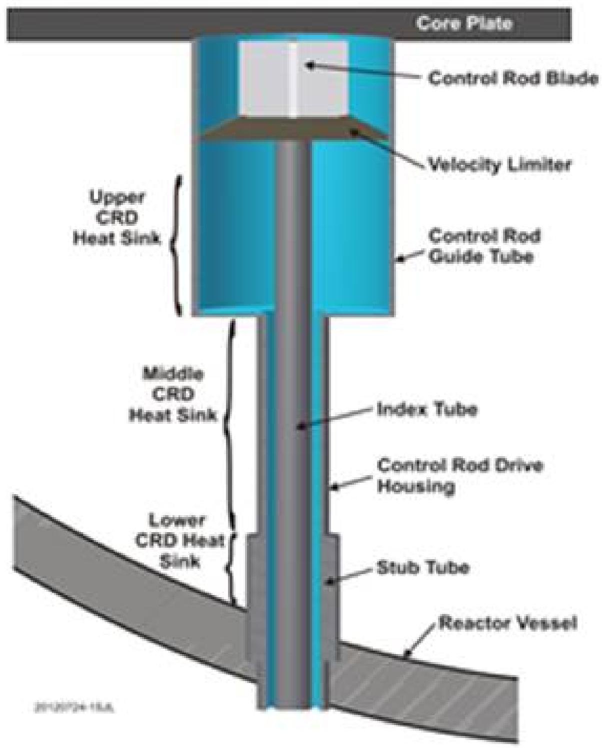

Figure 3-1
Schematic of the BWR geometry in the lower Pressure Vessel Plenum

3.2.2 Geometric Configurations during the 1F2 Core Melt Event

To investigate the potential for recriticality, geometries of the intact and molten core materials have to be chosen to represent the various core states that occurred in the 1F2 event. These geometries are chosen to envelope the most critical geometries that may have occurred. The geometries of interest that were chosen represent time periods:

1. A base case with core and control blades completely inserted (intact) where the water has just boiled off to the top of the active fuel

2. Shortly after the control material have started melting and are no longer in an otherwise intact core
3. An intermediate time during core melting when the core materials have started to candle downward and mostly slumped into the bottom half of the core, top of core region mostly void
4. Time when the core is mostly sitting on top of core plate in a partial molten and partial rubble bed state
5. Pool of stratified molten materials in the lower plenum region, with water on top

For cases 2, 3 and 4, the assumption would be that initially no water present, but the possibility exists that core is reflooded during a recovery effort. While it is not clear the exact trajectory of the 1F2 event, additional calculations were made with the core region reflooded to bracket the estimates of k_{eff} that might have occurred. Such information would be of interest in devising accident management strategies for future core damage events. This study only covered cases 1 and 2, while cases 3 thru 5 will be analyzed in Phase II of this study.

The calculations are consistent with the degree of knowledge available for the damaged core. In the case of Fukushima, our knowledge is very low, and somewhat speculative. The goal is to answer relatively simple question of: was the reactor close to criticality in any of the 1F2 event states. The same applies to other aspects of the calculations. The geometric modeling details or isotopics for this evaluation are less than are typically needed for fuel management calculations. This study attempted to match the calculational efforts to the degree of precision needed, i.e. estimates of k_{eff} to about 2 significant figures.

The calculations used the MAAP5 results [EPRI, 2013] as a starting point. These calculations provided temperature, pressure, material compositions in the core region (and pressure vessel) during the event. MAAP5 calculations were done on a nodalized basis, i.e. the core region was divided up into nodes and the amount of water, fuel, cladding, channel box and control material obtained for each node throughout the accident. Once core dislocation occurred, a relatively simple core model was created that corresponded to the mesh structure in the MAAP5 calculation: 5 radial rings and 12 axial zones. The axial nodes are uniform. The radial nodes are not uniform, but arranged such that the volume in each radial region is equal. Thus, each of the nodes contained an equal volume. In additional MAAP5 provided the geometries of the steel core barrel, upper and lower support plates, and control rod guide tubes in the lower plenum.

The MAAP5 analyses provided multidimensional distributions of temperatures and densities at each of these nodes. This includes temperatures of the fuel, cladding, water, and corium along with the density of the water (and steam/hydrogen.) This forms the basis for estimating the presence absence of fission product poisons (particularly Xe) which may preferentially escape molten materials. The control and structural materials, at least so long as they were intact, were assumed to be at the same temperature as the water, and the variations in density of the fuel, cladding, control, and structural materials were ignored given the many simplifications made, consistent with the other assumptions in the analysis.

Only for the initial state where the core is relatively intact, except for the control blades perhaps having melted (case 2 above), is there any semblance of a distinct geometry. The answer to this configuration, namely that a BWR with no control blades with no water in the core, is unlikely to be critical. With water reflooding the core, at some water level criticality will be achieved. The only question is approximately what water level will result in criticality. This is a very crude question, and only needs an approximate answer.

It is also worth noting that this "rodless core" state would likely exist for only a couple of minutes (10 minutes at most) during a reactor accident similar to what occurred at 1F2. Once the core starts to melt, the highly exothermic Zr-water reaction leads to rapid core melting. A Fukushima unit 2, the time between start of fuel uncovery and complete core meltdown would have be about 25 minutes. It is also highly unlikely that 100% of the boron (in a steel eutectic mixture) will actually depart the core. As shown in a previous paper, a relatively small amount of boron retained in the core could be sufficient to prevent recriticality [Mosteller, 1995].

For other periods during the event, knowledge of the actual core geometry is highly uncertain. The MAAP5 calculations provide the amount of material in the various nodes, and for the late stages of the accident the materials pool (liquid state plus crust) in the lower head of the pressure vessel. Thus estimates of possible criticality are likewise equally uncertain, but nevertheless provide us insights as to how close to (or far from) critical the core was during the event.

3.3 FUEL and ISOTOPICS

The following 1F2 parameters were important to the Monte Carlo calculations:

Table 3-1
1F2 Fuel Load

Fuel Assemblies	548	
U-235 Enrichment	3.6 wt%	high burn-up 8x8 fuel
	3.8 wt%	9x9 fuel
Total Uranium	93t – 94t	high burn-up 8x8 fuel
	95t	9x9 fuel (type A)
	94t	9x9 fuel (type B)
Relative Power	25.6 MWd/t	
Nominal Power	2381 MWt	

The BWR core consisted of repetitive patterns of four fuel bundle elements, centered about the intersection of two wide water gaps. Each fuel bundle is contained in a Zircaloy fuel channel, or box. Each grouping of four fuel bundles has a B_4C control blade in the center. This arrangement varies on the edges of the core in such a way that the pattern approximates a circle.

Each fuel bundle contains Zircaloy tubes or rods in either an 8x8 or 9x9 pattern. These tubes are filled with UO_2 fuel pellets. For purposes of fuel management, some of the fuel rods are replaced with a water filled rod, and some of the UO_2 rods also contain a burnable poison, Gd_2O_3.

The objective of BWR fuel cycle design is to achieve an equilibrium cycle. Prior to the accident, 1F2 had been operating for many years on a roughly 13-month refueling cycle. In this scheme, 1/3 of the core is changed out each 13-month cycle, so that at the time of the accident part way through the refueling cycle, 1/3 of the core had 2+ years exposure; 1/3 of the core had 1+ years exposure; and the balance of the fuel had only part of a year exposure.

A representation of the fuel is shown in Figure 3-2. However this configuration is only valid during the first few minutes of the meltdown scenario; after that the fuel and structural materials start to melt and flow downwards, creating a mélange of materials in lower nodes. Further complicating this picture is the phenomena of eutectic formation, wherein certain materials start to melt at lower temperatures than others. For instance, boron and steel will form a liquid mixture at much lower temperatures than urania.

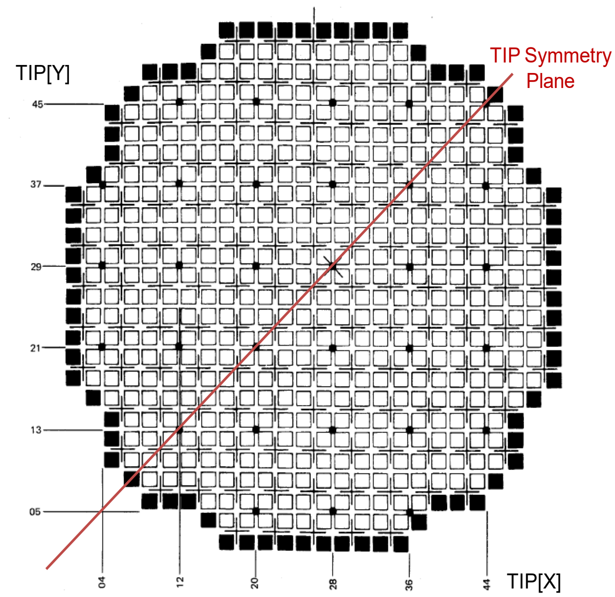

Figure 3-2
A BWR core: fuel assemblies arranged in a pattern to fit inside the circular core shroud and pressure vessel

Table 3-2
1F2 Fuel Rod Specifications

8x8 high burn-up fuel		
	Diameter of fuel pellet	10.4 mm
	Height of fuel pellet	10.0 mm
	Outer diameter of cladding tube (ZrO_2)	12.3 mm
	Thickness of cladding (ZrO_2)	0.86 mm
	Thickness of Zr liner	~0.1 mm
	Gap between Zr liner and pellet	~0.1 mm
	Rod pitch	16.30 mm
9x9 Type B Fuel		
	Diameter of fuel pellet	9.6 mm
	High of fuel pellet	10.0 mm
	Outer diameter of cladding tube (ZrO_2)	11.2 mm
	Thickness of cladding (ZrO_2)	0.71 mm
	Thickness of Zr liner	~ 0.1 mm
	Gap between Zr liner and pellet	~ 0.1 mm
	Rod pitch	14.4 mm
Fuel Assembly 8x8 high burn-up		
	Number of assemblies in core	68
	Number of fuel /rod assemblies	60
	Outer diameter of water rod centered	34 mm
9**x9 type B fuel**		
	Number of assemblies in core	332
	Number of fuel/ rod assemblies	72
	Side length of square water channel	38.5 mm
	Total mass of one fuel assembly	311 kg
	Side length of channel box with 8x8 and 9x9	134 mm
	Material of channel box	Zircaloy-4

Table 3-3
1F2 Core Material Specifications

Control blade	Control material	B_4C	1,150 kg
	Configuration	Cross shape	-
	Number of control blades	97	-
	Pitch	305	mm
Stainless steel structures	Top guide	6,900	kg
	Core Plate	9,300	kg
	Control rod guide tubes	88,680	kg

3.3.1 ISOTOPICS

In order to obtain a reasonable estimate the isotopics present in the core at the time of the accident, it was necessary to perform a depletion calculation for an equilibrium core.

The analysis used of the U.S. NRC codes PARCS/PATHS [Wang, 2013] [Collins, 2011]. These codes calculated the equilibrium cycle isotopics for the Fukushima-surrogate (Hatch Unit I) BWR and processed the data for developing input to a Monte Carlo model. Hatch Unit I is a GE BWR4 reactor power plant and Mark I Containment which is very similar to the Fukushima BWR unit 2. The core design and operating data for Cycles 1-3 of Hatch were documented in EPRI reports [EPRI, 1975a], [EPRI, 1979b], [EPRI, 1984]. The codes PARCS/PATHS have been benchmarked using Cycles 1-3 from Hatch as reported [Yarsky, 2013b]. The equilibrium cycle in the work here will be based on the same GE 8x8 fuel type used in cycles 3 of Hatch. The equilibrium isotopics data from PARCS have been post-processed to determine the assembly-wise isotopics which will then be written into the Monte Carlo model.

Since this report is intended for the public domain, it is not possible to use the exact 1F2 core model, since many of the core design and fuel parameters are proprietary. To overcome this difficulty, Hatch reactor, which is very similar to 1F2, was selected. The information on Hatch is readily available. The Hatch plant and 1F2 have similar number of fuel assemblies, power and flow rates. There are minor differences for fuel design and core management strategies, so the effects are relatively small for the initial analyses presented in this report. In the event that there are configurations with low reactivity margin, i.e. k ~ 1.0, then additional calculations would be required using more precise 1F2 parameters. In general, Hatch appears to be a good surrogate for 1F2. A comparison between the two is presented in Table 3-4.

Hatch Unit 1 is a BWR/4, with 560 fuel assemblies with a startup power rating of 2.436GWt. The fuel loading in cycle one through three was a mixture of 7x7, 8x8, and 8x8-LTA fuels. Although these fuels do not represent the current state-of-the-art BWR reactor fuel, the 8x8 was chosen as a reasonable surrogate. An equilibrium model was taken from the cycle 2 to cycle 3 shuffling, which will be used as the basis for the results given here.

The U.S. NRC core simulator, PARCS, was used to perform the equilibrium core analysis in the work performed here. PARCS supports the development of a core burnup distribution through an equilibrium cycle search. This can be done exactly, using a 1-1 unique bundle movement, similar to a core follow exercise, or the fuel can be "batched" to reduce the complexity of the fuel shuffle. This batching was used here to reduce the complexity of the model and provide some useful data for an initial PARCS and then SERPENT model.

Table 3-4
Comparison of Hatch versus Fukushima Unit 2/3 parameters

	Fukushima 1F2		**Hatch**	
Reactor Type	BWR-4			
Unit / Cycle	Unit 2	Unit 3	Cycle 2/3	Equilibrium
Assembly Type	9x9	9x9 MOX	7x7, 8x8	8x8
Number of Assemblies	332 (9x9) 216 (8x8) 648 Total	516 (9x9) 32 (MOX) 548 Total	560 Total	
Ave. enrichment (w/o U-235)	2.58		2.36 / 2.46	2.5
Power	2381		2436	
Operating pressure (MPa)	7.03		7.13	
Coolant flow rate (t/hr)	33,800		35,600	
Control Absorber	B4C		B4C granules in SS	
Number of Blades	137		137	

In BWR reactors the fuel is arranged in assemblies of fuel rods clad in Zircaloy and moderated by water. The arrangement is referred to as a reactor lattice. The reactor is controlled by cruciform control blades, made of stainless steel encapsulating a neutron capture material, B_4C. These fuel assemblies are arranged in a form of squares arranged in a nearly circular pattern to fit with the cylindrical internals of the pressure vessel. Since the lattice and the fuel assemblies are repetitive structures, the isotopic calculations identify a repetitive element, called a unit cell that comprises a single fuel rod, its cladding and adjacent moderator. For calculation purposes, it is assumed that there is zero net neutron current between cells. A fuel bundle consisting of 4 fuel assemblies of the type used in1F2 is shown in Figure 3-3. A full length assembly is shown in Figure 3-4, and the core arrangement is shown in Figure 3-2.

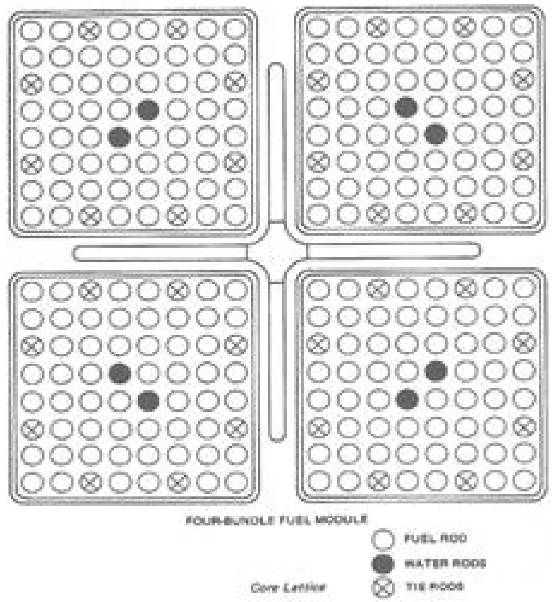

Figure 3-3
A BWR bundle arrangement. Four fuel assemblies form a basic module with a cruciform control blade in the center

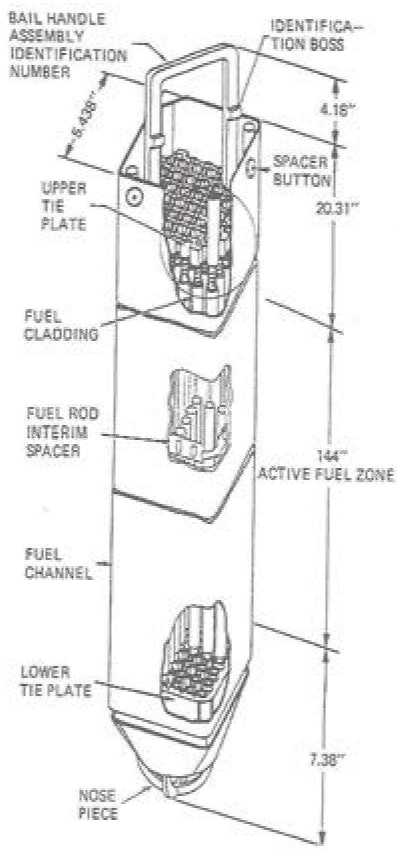

Figure 3-4
A typical complete BRW fuel assembly showing the Zircaloy fuel channel

The following sections describe the deterministic methods used to model the Hatch BWR, and then summarize the equilibrium cycle calculation and results. Finally, the methods will be described which were used to map the equilibrium cycle isotopics from the deterministic model to the Monte Carlo model for criticality calculations.

3.4 Deterministic Coupled Codes

This section describes a coupled neutronics – thermal-hydraulics capability developed for performing a 3-D full core analysis of the Fukushima core. This HELIOS/PARCS/PATHS code system was applied to find the equilibrium cycle core composition based on the Hatch Unit 1 design specifications. The following section describes the HELIOS lattice model and the generation of homogenized multi-group cross sections for PARCS. The next section describes the full core models developed with the US NRC core neutronics code PARCS, as well as the coupling of PARCS to the thermal-hydraulics code PATHS for performing the equilibrium cycle search. The process for determining an equilibrium composition based on the Hatch Unit 1 core along with the results of the equilibrium core calculation with a coupled HELIOS/PARCS/PATHS model are presented. Finally, the methods and mapping used for generating a SERPENT Monte Carlo model are discussed along with a comparison of the deterministic and Monte Carlo results. The overall code system used for this core analysis is summarized in Figure 3-5. Each of the modules in this code system will be described in the following sections.

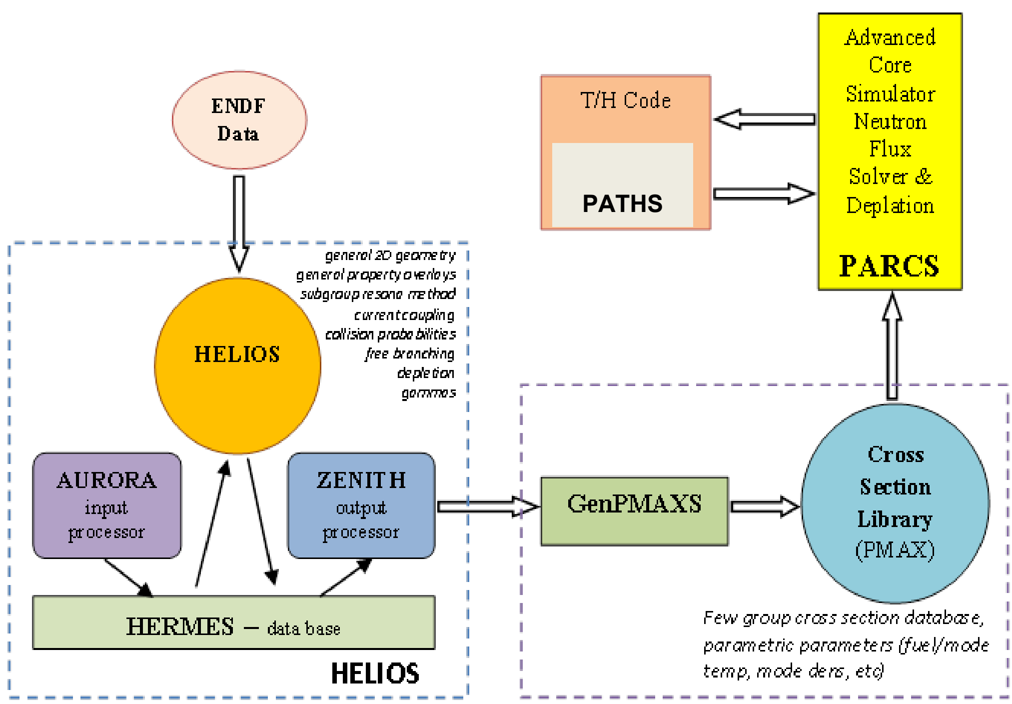

Figure 3-5
Computational Code Package for the Fukushima Unit 2 Full Core Calculation

3.4.1 Lattice Modeling

HELIOS [Stammler, 1994] is a two-dimensional neutron and gamma transport code for fuel assembly calculations in general two-dimensional geometry developed by Studsvik Scandpower, Inc. The particle transport is performed using the current coupling collision probability (CCCP) method in which the space elements are globally coupled using interface currents and local transport with the space elements performed using collision probabilities. The resonance treatment is based on the subgroup method and allows for full interaction of the resonance isotopes. The HELIOS nuclear data libraries are based on ENDF/B-VI and are provided in a 190 neutron energy group structure. Depletion is performed using a predictor/corrector method with 29 heavy isotopes and 114 fission products. The HELIOS input and output processors are the separate codes AURORA and ZENITH. The data flow between the three codes is via a data base that is accessed and maintained by a subroutine package called HERMES. A 2-D assembly model (Figure 3-6) was created to calculate the cross section data with HELIOS.

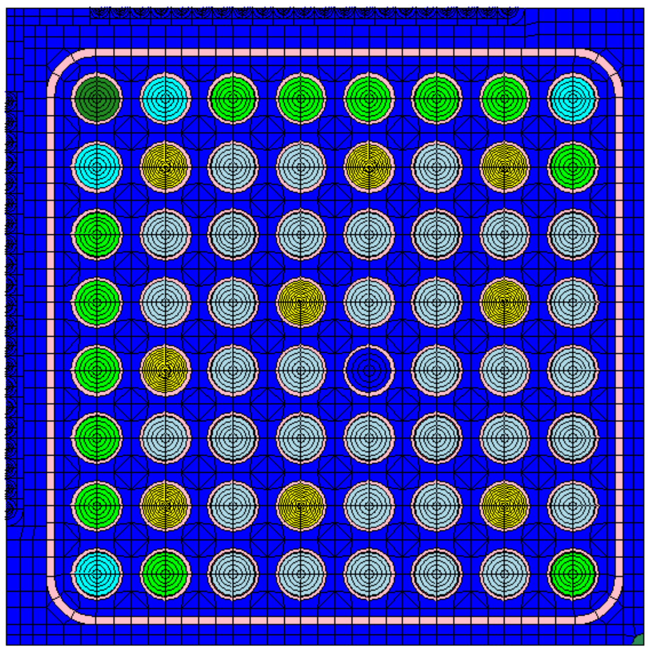

Figure 3-6
HELIOS Assembly Model

3.4.2 Cross Section Formatting with GenPMAXS

The PARCS cross section interface code GENPMAXS processes the cross section data generated by various lattice codes into the PMAXS file format that is used by PARCS. PMAXS provides all of the data necessary to perform core simulation for steady-state and transient applications including the principal macroscopic cross sections, the microscopic cross sections of Xe/Sm, and the group-wise form functions with several different branch states for the appropriate fuel burnup states (see Figure 3-7). In this analysis, the GENPMAXS program was used to generate the PMAXS files from the macroscopic cross section libraries of the lattice code HELIOS. The representation of the cross sections and the major methodologies employed in the PMAXS format are available in the GENPMAXS code manuals. The cross section data for PARCS is produced in GENPMAXS using the history and branch settings as shown in Table 3-1 and Table 3-2 which was taken from the code manual.

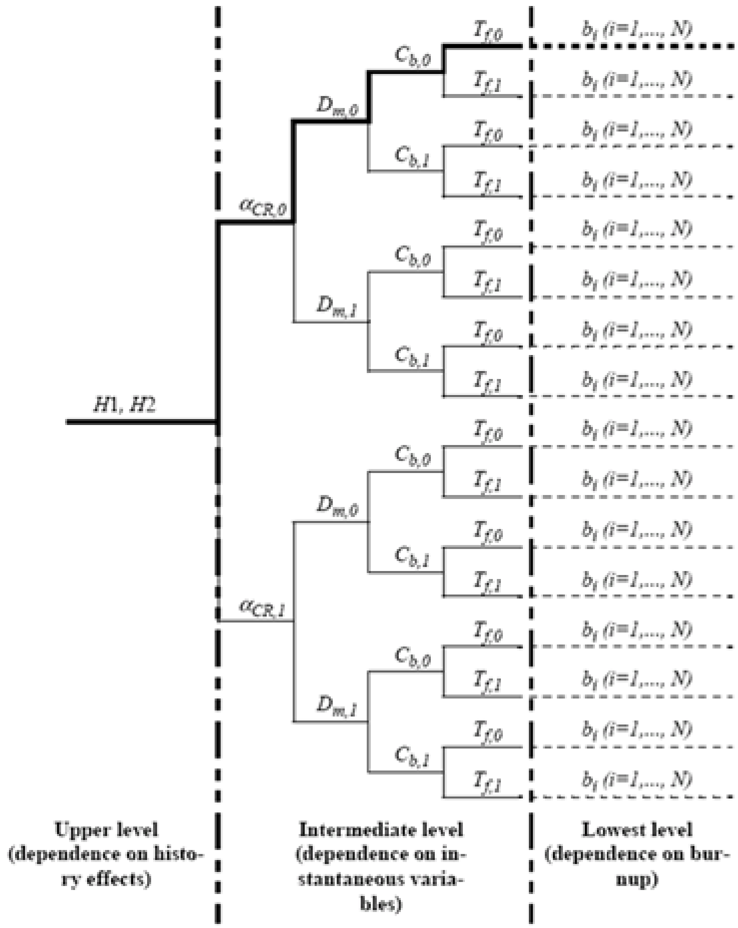

Figure 3-7
Pin Cell Configuration in HELIOS Lattice Model

Table 3-5
History Structure

History	Control Rod State	Moderator Density (g/cc)	Corresponding Void Fraction (%)	Fuel Temperature (K)
1	0	0.10000	80	850
2	0	0.45731	40	850
3	0	0.80000	0	850
4	1	0.10000	80	850
5	1	0.45731	40	850
6	1	0.80000	0	850

Table 3-6
Branch Structure

Index	Control Rod State	Moderator Density (g/cc)	Corresponding Void Fraction (%)	Fuel Temperature (K)
1	0	0.45731	40	850
2	1	0.45731	40	850
3	0	0.10000	80	850
4	0	0.31731	60	850
5	0	0.59730	20	850
6	0	0.80000	0	850
7	1	0.10000	80	850
8	1	0.31731	60	850
9	1	0.59730	20	850
10	1	0.80000	0	850
11	0	0.10000	80	500
12	0	0.10000	80	1500
13	0	0.45731	40	500
14	0	0.45731	40	1500
15	0	0.73617	0	500
16	0	0.80000	0	1500
17	1	0.10000	80	500
18	1	0.10000	80	1500
19	1	0.45731	40	500
20	1	0.45731	40	1500
21	1	0.80000	0	500
22	1	0.80000	0	1500

3.4.3 PARKS Neutronic Core Simulator

PARCS is a reactor core simulator which calculates the neutron flux in a nuclear reactor core [Downar, 2006]. In addition to the standard eigenvalue calculation for a given reactor configuration, criticality searches are also available in which the critical control rod pattern and critical boron concentrations are determined. PARCS has the capability to analyze both short term (kinetics) and longer term (depletion) core behavior. PARCS is also coupled to the PATHS steady state TH code for fuel cycle depletion calculations. In order to provide the depletion capability to PARCS, a depletion module was added to PARCS, as well as a cross section module for calculating a node-wise cross section for its burnup history and current TH state from PMAXS. The depletion module relies on the PARCS calculated neutron flux solution to update history state information (e.g. burnup and control rod history) during the simulation of a fuel cycle. The cross section module calculates cross sections based on the burnup and other history state information, as well as on the current thermal-hydraulic state. Burnup dependent macroscopic cross sections are read from the PMAXS file prepared by the code GenPMAXS, and the PARCS node-wise power is used to calculate the region-wise burnup increment for time, advancing the macroscopic cross sections.

The standard practice in LWR core analysis is to homogenize cross sections in space and in energy over a fuel assembly sized region. For Light Water Reactors, this cross section homogenization is customarily performed in two energy groups. The cross-sections are functionalized for the expected range of conditions in the nodal calculations for variables such as fuel temperature (TF), coolant temperature (TC), control rod position (CR), coolant density (DC), and soluble poison concentration (PC). These branch states are perturbed from history conditions through which the node would experience exposure during the cycle. Since the macroscopic cross sections are strongly dependent on history effects, especially for BWR analysis, up to five history variables can be employed, with the cross sections represented as functions of burnup and history.

The cross section libraries in the work performed here were developed using methods consistent with practices specified by the U.S. NRC. The branch conditions and history states were determined using the guidelines in NUREG/CR-7164 [Wang, 2013], and the history variables considered were control rod history and moderator density. The unrodded void histories were 0%, 40%, and 80%, and the fully rodded history corresponded to 0%, 40% and 80% void fraction. All six histories were carried out at a hot fuel temperature (850K). Instantaneous state branching calculations were performed at five void fractions in combination with four changes in fuel temperature (0%, 20%, 40%, 60%, and 80% with 500K, 950K, 1500K, and 2500K). These branchings were completed for both controlled and uncontrolled conditions.

The tabulated data in the PMAXS file includes the macroscopic cross sections, the microscopic cross sections of Xenon and Samarium, the group-wise form functions with several different branch states for the appropriate fuel burnup states, and all of the appropriate kinetics data. PARCS currently employs a macroscopic depletion method, and, with the exception of Xenon and Samarium (which are represented by their microscopic cross sections and number densities), the microscopic cross sections and number densities are not specifically tracked during core depletion. Further details on the cross section modeling in PARCS are provided in the GenPMAXS manual [Ward, 2015].

3.4.4 PATHS Core Thermal-Hydraulics Simulator

The PATHS code was originally developed to solve for the steady state thermal-hydraulic state parameters (void fraction, moderator density, pressure, and temperature) of the BWR, in order to provide state data to PARCS for cross section feedback [Collins, 2011]. PATHS is based on an incompressible flow, drift flux formulation of the two-fluid TH equations with a subcooled boiling model. This makes it possible to perform fast running calculations with one-to-one neutronic/TH mapping to calculate the void fraction for each neutronic node, which improves the fidelity of the coupled neutronics/TH solution of the system. The hydraulic conservation equations consist of continuity in each channel, core wide continuity, and momentum in each channel using the finite volume method. These hydraulic equations are combined into an equation set for parallel, heated channels, with the boundary conditions being specified as the total flow into the channels, the enthalpy at the inlet, and the pressure at the outlet. Currently, the core bypass is not treated in PATHS and therefore was not included in the current analysis and assessment. A separate study was performed to evaluate the impact of bypass treatment and it was found to be negligible for the purposes of the study performed here [Yarsky, 2013c].

In PATHS, the energy balance is formulated through the enthalpy, and the nonlinear relationships between mass, momentum, and enthalpy are resolved by using the previous iteration data in solving the hydraulics equations. PATHS solves the energy equation with the pressure and velocity distributions from this previous iteration and convergence between the fields is achieved when the error residuals become smaller than a specified tolerance. The solution algorithm in PATHS is shown in Figure 3-8 below.

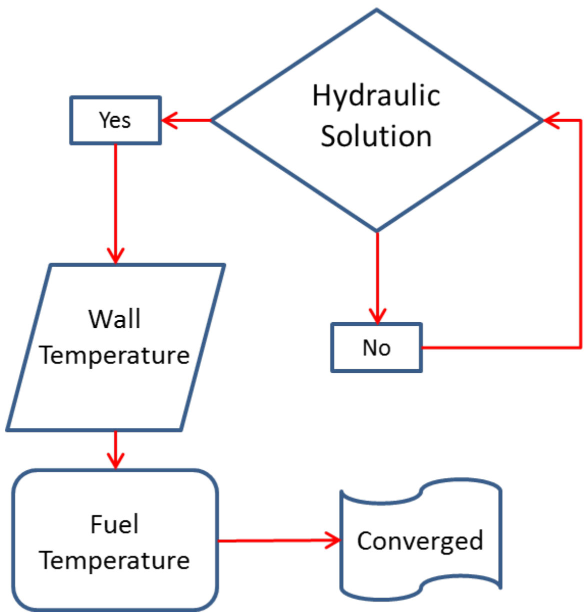

Figure 3-8
PATHS Solution Flow Diagram

3.5 Analysis of HATCH Cycles 1-3 with PARKS/PATH

PARCS and PATHS had previously been assessed through detailed comparisons of code predictions to the full-scale plant data [Yarsky, 2013b]. This operational data had been collected during Cycles 1 through 3 at Hatch Unit 1, and the results were published in a series of Electric Power Research Institute (EPRI) reports [EPRI, 1979a], [EPRI, 1979b], and [EPRI, 1984]. However, the results published in [Yarsky, 2013b] were performed using cross sections calculated by HELIOS [Stammler, 1994]. The data from the EPRI reports include core flow, vessel pressure, and traversing in-core probe (TIP) measurements, all evaluated at critical operating state points. In addition to providing these measurement data, the EPRI reports also provide details about the plant thermal-hydraulic and core nuclear design. This section will present revised results over what has previously been published. The model development approach, calculation methodology, and comparison of the calculation results to data are presented.

The model development of the H1C1, H1C2, and H1C3 was performed in three stages. In the first stage, a PATHS thermal-hydraulic model of the core was developed. In the second stage, a PARCS neutronic model of the core was developed and integrated with the PATHS thermal-hydraulic model. In the third stage, TRITON was used to generate nuclear cross-sections for the coupled calculation. There are three fuel types present in the first core (H1C1), but a total of five types present in the third core (H1C3). The Type 1-3 fuel bundles are 7x7 array types while Types 4 and 5 are 8x8 designs with internal water rods. The core hydraulic model is a combination of individual channel components based on the five unique fuel types that comprise Cycles 1, 2, and 3 (7x7 arrays and 8x8 arrays with internal water rods). The core is modeled with 560 parallel channels each with 24 axial nodes (15.24 cm in height). The details of the fuel thermal-hydraulic design data used in the PATHS model are provided in [Yarsky, 2013b].

The nuclear model was coupled to the thermal-hydraulic model with the same radial and axial nodalization. The PARCS description of the nuclear model specifies the axial "stacking" of individual lattice types for each fuel bundle design. Once the fuel bundles are stacked, radial arrays specify the layout of the nuclear nodes within the core. In addition to specifying the core fuel loading in this manner, the PARCS nuclear model also accounts for the positioning of control rods and TIPs within the core. The neutronics solution in PARCS was performed using the nodal expansion method (NEM) with two energy groups and coarse-mesh-finite-difference (CMFD) acceleration method.

Once the basic models were developed, the cycle depletions were performed sequentially since the predicted exposure patterns of the H1C1 core at end-of-cycle (EOC) impact the initial core loading at the beginning-of-cycle (BOC) for H1C2; and so on. The cycle exposure history was simulated using the standard quasi-static depletion method in which the core state (i.e. power, flow, pressure, and control rod pattern) is held constant between depletion points. The detailed depletion steps and operating histories used in the analysis are provided in [Yarsky, 2013b].

The k_{eff} predicted by PARCS/PATHS at each known critical point during the simulated exposure of each of the three cycles is depicted in Figure 3-9. H1C1 was subdivided into two portions; the first part of H1C1 indicates poorer agreement when compared to the second part. In the first part of operation of H1C1, the core support plate included holes to promote bypass flow in the inter-assembly region. Part way into H1C1, the reactor was shut down, and these holes were plugged. As can be seen in Figure 3-9, the agreement between the calculations and the plant data is much better following this plugging operation. While the specific cause of the change in code performance has not been thoroughly studied, there are two factors that contribute to the change. First, early during H1C1 operation, the plant operating history is somewhat sporadic owing to initial cycle testing and other operational factors. Therefore, one can expect that the use of a constant power, step-like approximation for a depletion interval would lead to some error. Second, the presence of the core support plate holes promotes bypass flow, which is not explicitly treated in the PARCS/PATHS model. Given that the effect of bypass flow is expected to be more pronounced for higher flow rates, it is conceivable that the poor agreement between calculation and measurement is related to the high bypass flow rate during the early part of H1C1. An explicit bypass flow treatment will be incorporated into a future version of the PATHS code.

Following core support plate hole plugging, the PARCS/PATHS calculation results show consistently good agreement in the predicted k_{eff} for the remainder of H1C1 as well as for all of H1C2 and H1C3. The average bias over all of H1C1 is quite large (~ -982 pcm), but this is reduced in H1C2 to -719 pcm and to -568 pcm in H1C3. Considering all of the k_{eff} data after plugging, the mean bias is -649 pcm and the standard deviation is 226 pcm. The k_{eff} agreement with the plant measurement is reasonable and indicates adequate performance of the PARCS/PATHS codes for BWR cycle depletion.

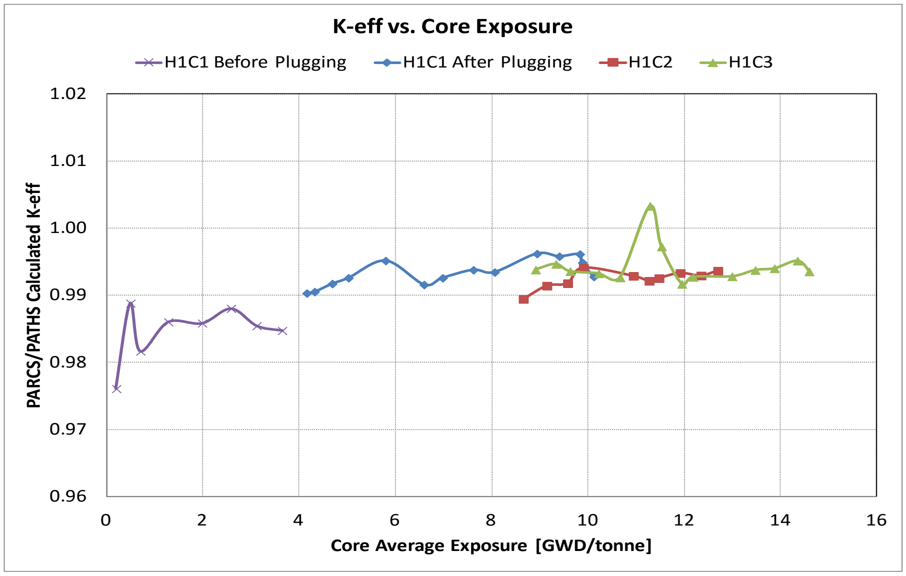

Figure 3-9
Predicted Core Multiplication Factor vs. Exposure

3.5.1 Equilibrium Cycle Calculation

Using the validation of cycles 1-3 as a basis, an equilibrium cycle was then determined for the Hatch Unit I core as a basis for the Monte Carlo criticality calculations. This section will first describe the search methodology and the specific shuffling pattern and reload scheme used for the equilibrium cycle, and the next section will then provide results of the equilibrium cycle search.

3.5.2 Search Methodology

The equilibrium cycle calculation was performed using the design parameters based on the Hatch core design. This included the assembly shuffle pattern and control rod position. The technique involves depleting the core over a given timeframe while moving the control rod positions at specified intervals. Once the core is depleted to the user specified amount, the fuel is shuffled and the core depletion is repeated. The average burnup is compared to the previous cycle and the process is repeated until the burnup difference falls below a set convergence criteria. The convergence criterion for this simulation was set to 0.5 GWD/T for the infinite norm of node-wise burnup at the End of Cycle (EOC). The overall flowchart is shown in Figure 3-10.

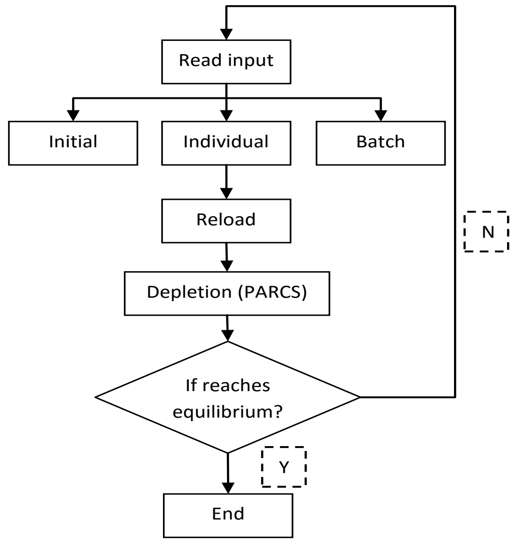

Figure 3-10
Equilibrium Cycle Search Flowchart

The control rod sequence pattern is used for core reactivity control during the cycle. The fuel shuffling pattern is used for multi-cycle fuel loading. An iterative algorithm has been developed to provide nested iterations to determine the equilibrium core configuration using the HELIOS/PARCS/PATHS code system. It takes into consideration explicit treatment of fuel bundles shuffling and control rod scheduling and the core was depleted with PARCS by steps defined by the specified control rod sequence. Some information regarding the Fukushima operating strategy was assumed. A three batch core was analyzed, considering an eighteen month cycle. Initially, batch reloading was used to model the fuel shuffling, where each batch is radially averaged at the end of each cycle before shuffling to the new positions. This approach was used to speed up core loading analysis, but an individual sequence was used for the higher fidelity calculations.

3.5.3 Equilibrium Cycle Results

A three batch fuel loading with a 18 month burnup cycle was assumed to capture both the given core burnup for Fukushima and also emulate the conventional operation of most operating BWRs. Design variables included the fuel enrichment, number of Gd rods, Gd enrichment in the rods, and also the core loading. The control rod pattern was adjusted to reduce the relative power peaking in the equilibrium cycle. The final fuel design was actually a variation of the original Hatch type 4 fuel in which the number of Gd rods was increased to nine, and the enrichment was also increased slightly.

The shuffle pattern is given in Figures 3-11 and 3-12. As can be seen, the fresh fuel is generally near the core periphery, with a checkerboard of once and twice burned fuels in the center of the core. Although this was higher leakage, the power peaking was more manageable. The resulting shuffle is rotated to allow a full core symmetric calculation.

1	95	97	2	49	6	52	12	55	20	60	30	114
96	48	3	99	7	101	13	105	21	109	31	66	73
98	4	50	8	102	14	56	22	61	32	115	42	120
5	100	9	53	15	106	23	110	33	67	74	121	84
51	10	103	16	57	24	62	34	116	43	79	85	128
11	104	17	107	25	111	35	68	75	122	86	129	
54	18	58	26	63	36	69	44	80	87	130		
19	108	27	112	37	70	45	81	126	92	134		
59	28	64	38	117	76	82	127	93	135	138		
29	113	39	71	46	123	88	94	136	139			
65	40	118	77	83	89	131	137	140				
41	72	47	124	90	132							
119	78	125	91	133								

Figure 3-11
Shuffle Index

-1	94	91	-1	46	-1	43	-1	40	-1	35	-1	74
92	47	-1	89	-1	87	-1	83	-1	79	-1	29	22
90	-1	45	-1	86	-1	39	-1	34	-1	73	-1	68
-1	88	-1	42	-1	82	-1	78	-1	28	21	67	11
44	-1	85	-1	38	-1	33	-1	72	-1	16	10	60
-1	84	-1	81	-1	77	-1	27	20	66	9	59	
41	-1	37	-1	32	-1	26	-1	15	8	58		
-1	80	-1	76	-1	25	-1	14	62	3	54		
36	-1	31	-1	71	19	13	61	2	53	50		
-1	75	-1	24	-1	65	7	1	52	49			
30	-1	70	18	12	6	57	51	48				
-1	23	-1	64	5	56							
69	17	63	4	55								

Figure 3-12
Shuffle Patterns

The total number of control rod “notches” inserted versus the core burnup is shown in Figure 3-13. As indicated, during the initial phase of the burnup cycle (core burnup 14-20 Gwd/tHM) the Gadolinia burnable absorber in the fuel controls the core very well and less than 700 steps of control rods are inserted in order to hold down core excess reactivity. The most highly reactive state of the fuel occurs at about 22 Gwd/tHM and is used as the core state for the analysis here.

The core relative power peaking is shown in Figure 3-14. As indicted, the power peaking generally is less than 2.25, with some cases slightly higher. An optimum rod management would likely reduce the power peaking, however, the core conditions shown here are reasonable given that the primary goal of this effort was an approximation of Fukushima Unit II for purposes of core criticality calculations which depends primarily on the core mass, enrichment, and reactivity control poisons.

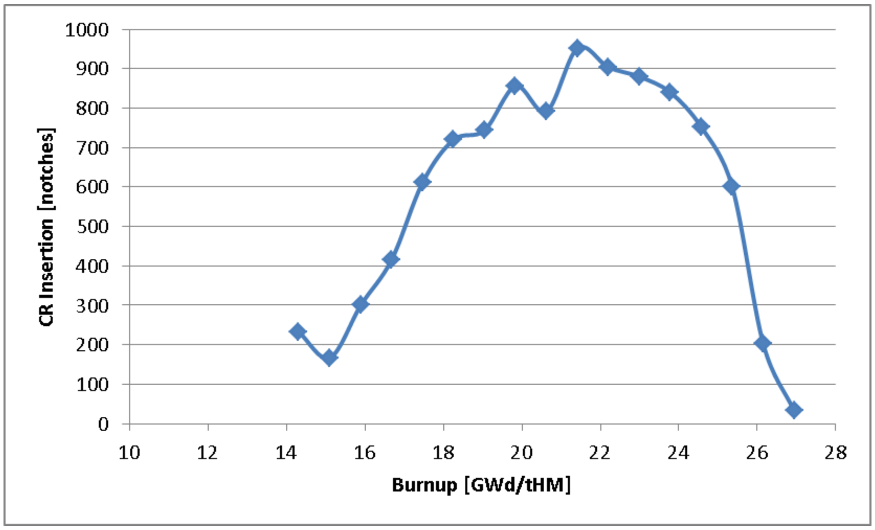

Figure 3-13
Control Rod Insertion Depth During Burnup

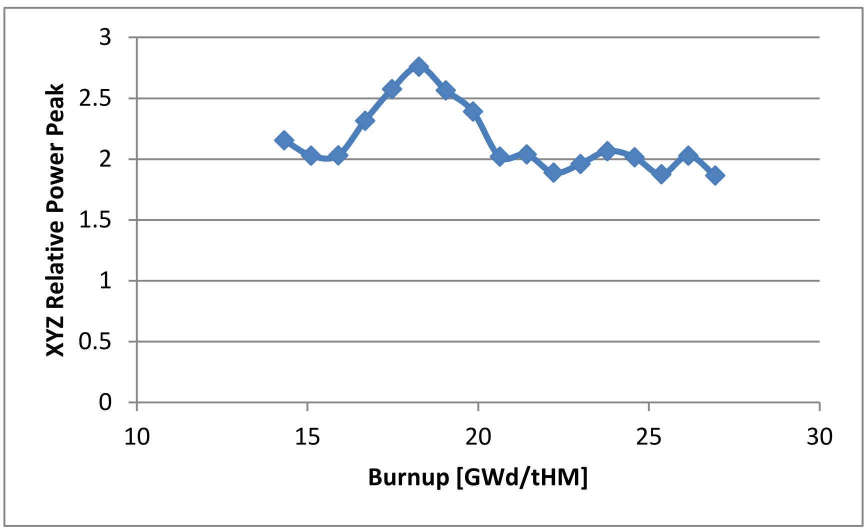

Figure 3-14
Core Relative Power Peaking During Burnup

Some of reactor cold cases were studied as well with HELIOS/PARCS as summarized in Table 3. These cases were intended to ensure realistic core shutdown performance, but also to provide comparison to SERPENT. The most important case is the cold shutdown conditions. In order to ensure the reactor is subcritical at the most reactive state, the thermal-hydraulic state of the core is assumed to be at room temperature and the k_{eff} must be below 0.99. As shown in Table 3, the equilibrium core here provides a k_{eff} at "cold" TH with all rods inserted of 0.985. This was intentionally chosen to be close to the acceptable conditions to provide a "conservative" condition for the Monte Carlo criticality calculations.

3.6 Monte Carlo Criticality Calculations

3.6.1 Preparation of PARCS Equilibrium Cycle Data for Serpent

The preparation of the Serpent input deck required several quantities to be calculated and transferred from the PARCS/PATHS equilibrium cycle solution. These included the isotopics, fuel temperature, and coolant temperature and coolant density. The fuel temperature, coolant temperature and coolant density are all saved for each node in the PARCS depletion file. These values are read in through a MATLAB script that converts each value to the correct units used in SERPENT. Once each value is converted, it is placed in its corresponding node inside the quarter-core geometry. All of the isotopic information was calculated based on the PARCS equilibrium cycle calculation using macroscopic depletion and the HELIOS lattice calculation. PARCS provides the node specific burnup and state conditions (coolant density, control rod history, fuel temperature) which are used to interpolate over the history/branch structure provided in the HELIOS input. This provides the pin-by-bin number densities for Serpent based on the PARCS node averaged quantities. Figure 3-15 shows an example.

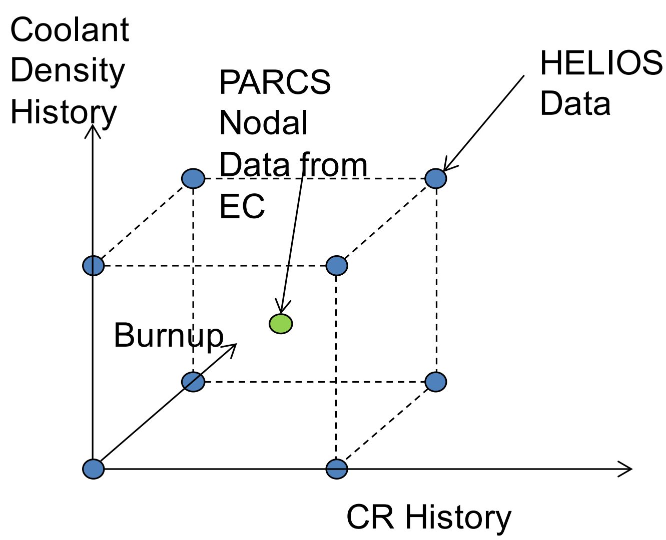

Figure 3-15
Isotopic interpolation based on PARCS and HELIOS depletion

Once all the number densities have been extracted for each pin type, then a separate script is used to generate the Serpent input file. This script creates a quarter-core geometry similar to the PARCS core layout by separately generating each assembly and filling each node with the appropriate number densities calculated from PARCS and HELIOS. Each assembly is then placed into its corresponding location. Control rods can be placed in the various core positions based on STL files created from CAD software. To avoid computer memory issues, some simplifications were made to the quarter-core Serpent model. For each pin, up to 160 isotopes were tracked from the HELIOS calculation. With a total of 24 axial regions, 140 assemblies and 64 fuel pins per assembly, this meant that there was up to a total of 34,000,000 isotopes. To reduce the total number of isotopes, a cut-off was placed that removed any isotope below a concentration of 1E-10. An additional simplification was made to the fuel temperatures. Serpent has a built-in Doppler broadening routine that allows the user to specify exact material temperatures. However, Serpent must generate a table for each separate material region which drastically increases the memory needed along with the initial time to load all of the materials. Using the explicit node temperatures provided from the PARCS equilibrium cycle calculation, SERPENT required more than 4 days plus an excess of 500 GB of memory just to load the materials. Instead, each nodes temperature was rounded to the nearest 5 degrees Kelvin. This reduced the total run time to one day while also significantly decreasing the total memory.

3.6.2 Mapping of PARCS Equilibrium Cycle to Serpent

In order to generate the SERPENT input file, the data from PARCS/PATHS was mapped to SERPENT. The input structure was divided into 140 assemblies (quarter-core), 24 axial regions and 64 pins per assembly. This produced a total number of 215,040 discrete regions within the Serpent geometry compared to 3,360 discrete regions in PARCS due to smearing the various pins during the homogenization process. Because the pins are smeared within the PARCS model, when the state information is transferred from that node to the Serpent model, all pins within that node are based on the same state conditions. The isotopics within those pins are different due to the pin-by-pin information produced from the HELIOS calculation. For the initial comparisons, a 1-to-1 mapping was used between the nodal schemes in PARCS to SERPENT as shown in Figure 3-16. Two separate radial reflector treatments were created based on a realistic cylinder of water surrounding the core and a set of “reflector assemblies” similar to the PARCS methodology. Comparisons of the two reflector treatments showed differences of less than 50 pcm, and therefore the realistic reflector model was used for all of the calculations.

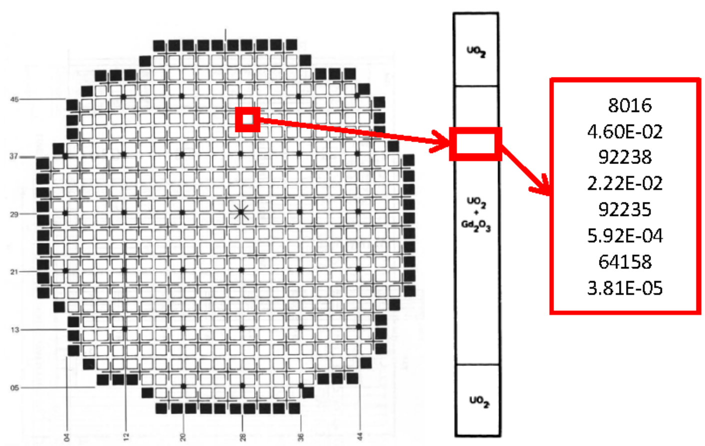

Figure 3-16
Comparison of PARCS and Monte Carlo Results

A sequence of cases was analyzed using the methods and cores described in the previous sections. The various core configurations are briefly described followed by a table comparing the PARCS and SERPENT results. The first case (Case 0) was performed to verify the consistency of modeling between the PARCS and SERPENT. This involved using all fresh fuel with uniform 300K fuel and coolant temperatures and 0.8 g/cc coolant density. In addition, all of the control rods were removed from this system. To test the fuel temperature profile, the 5K interval temperatures were then introduced using the burned isotopic composition based on PARCS depletion. The coolant temperature was maintained at 550K and the coolant density was kept at 0.8 g/cc while keeping the control rods out of the core. This case (Case 1) is closest to the operating conditions of the core but with all of the control rods removed. To test the core eigenvalue immediately after a scram, control rods were inserted into the previous case and state conditions were modeled that are similar to full power core operation. This case is shown in the Table as Case 2. The last two cases involved testing the eigenvalue at cold shutdown with the control rods both in and out of the core. The fuel and coolant temperatures were both set to 300K and the water density was increased to 1.0 g/cc in Case 3. The control rods were then removed at cold shutdown, this is expected to be the most reactive state for the core and is shown as Case 4.

Table 3-7
Comparison of Serpent and PARCS Results

Cases	Fuel	Rods	Temp	Coolant Density	PARCS	SERPENT	Delta-k (Serp-Parcs)
0	Fresh	ARO	300	0.8	1.19245	1.20097 ±16pcm	852
1	Burned	ARO	Real	0.8	1.14349	1.14080 ±16pcm	-269
2	Burned	ARI	Real	0.8	0.93544	0.93416 ±16pcm	-128
3	Burned	ARI	300	1	0.98595	0.97564 ±17pcm	-1031
4	Burned	ARO	300	1	1.15296	1.13202 ±15pcm	-2094

As indicated in the table there is reasonably agreement (<1% k) in cases 0 which provides confidence in the consistency of the PARCS and SERPENT models and the mapping of the isotopic data. There is also good agreement in cases 1-2 which are models the actual core operational state. The discrepancy between PARCS and SERPENT increases (>1%) for cases 3 and 4 which is to be expected since the core is at cold conditions for which the deterministic results are not as reliable. In general, however, the results shown in Table 3-3 provide confidence in the core geometric modeling and isotopics mapping, and therefore confidence in the application of SERPENT to criticality calculations with MAAP temperature / fluid conditions.

3.6.3 Mapping of MAAP conditions to SERPENT

In addition to mapping the isotopic information from PARCS to SERPENT, the data produced from the MAAP calculation (fuel temperature, coolant temperature, coolant density, etc.) was transferred to the SERPENT input. Because the MAAP calculation was performed with a different nodalization scheme from the PARCS simulation, the state information had to be mapped accordingly. Within MAAP a total of five equal-volume rings and 10 axial layers were used for the simulation as shown in Figure 3-17. Since each radial ring is equal-volume, a total of 28 assemblies from Serpent were mapped to each radial section's information. Similarly, 2 to 3 axial regions from Serpent were mapped to each axial region from the MAAP calculation.

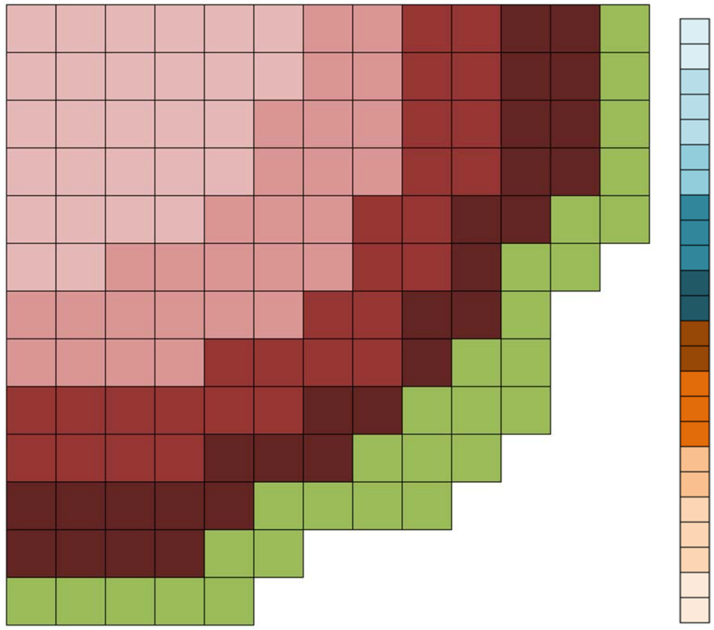

Figure 3-17
Radial map (left) and axial map (right) from MAAP to SERPENT

4
MONTE CARLO CRITICALITY ANALYSIS OF FUKUSHIMA DAIICHI UNIT 2

The initial core configuration employed in this analysis corresponds fairly closely to the average properties of the core as a whole. The objective of fuel cycle design is for the core to reach an equilibrium cycle—which is what was assumed for 1F2. In the initial phases of the event, before significant fuel melting occurred, the fuel assemblies from different fuel cycles can be actually approximated by the same bundle at different burnups. Therefore initial isotopics can be obtained from a single depletion calculation with a lattice physics code.

Previous work has estimated the control rod worth of an unrodded BWR to be approximately k ~1.07, resulting in a prompt critical core. For a fully rods inserted case k was ~0.81. Other estimates given in the prior study (for comparison purposes), estimated a Hot Full Power core without control rods at 40% void to be k~1.05, and one at Room Temperature and Pressure, rods inserted, to be ~0.96. These results were the result of a generic study using much more 'homogenized' assumptions due to lack of specific information as to a core configurations such as was available to this study using information from the 1F2 reactor.

For the period following the 1F2 event, the level of Xe-135 increases in the fuel rods because the half-life of Xe-135 (~9.1 hr) is longer than its I-135 precursor (~6.7 hr). At shutdown the reactivity worth of Xe-135 is ~ -0.02 $\Delta k/k$ at normal full power operating condition. The reactivity worth will become more negative for some hours post shutdown. This decreases the likelihood of recriticality during this period. In addition, he core or 1F2 started to relocate at approximately 3 days after the earthquake, so any Xe-135 built up in the initial few hours would have been considerably depleted. This is not the case for a core melt, as happened at 1F1, where core dislocation occurred after a few hours.

If, however, there is significant core damage or cladding failure, the Xe-135 which is in a gaseous state, can leave the core area, and therefore no longer act as a neutron poison. Not all of the Xe-135 will escape, since much of it will be retained in the fuel matrix or melt. Only the Xe-135 trapped in the interior cladding volume, or that which migrates out of the fuel matrix during melting will be lost. Our understanding of this process is limited, and consequently estimates of this are very rough.

A small, but still significant amount of reactivity is contributed by Sm-149. Its precursor, Pm-149 has a relatively long half-life, that little decay will occur for most of the accident, and is certainly the case for the 1F2 event.

Another consideration is the change in the density of moderator in ambient conditions, i.e. at 298°C (77°F) and 0.1 MPa (14.7 psi). Under these conditions the density of water is almost exactly 1.0 g/cm3. Conditions at 1F2 during the accident were much different: pressures and temperatures were much higher. MAAP calculations for these parameters are given in Figures 4-1 and 4-2.

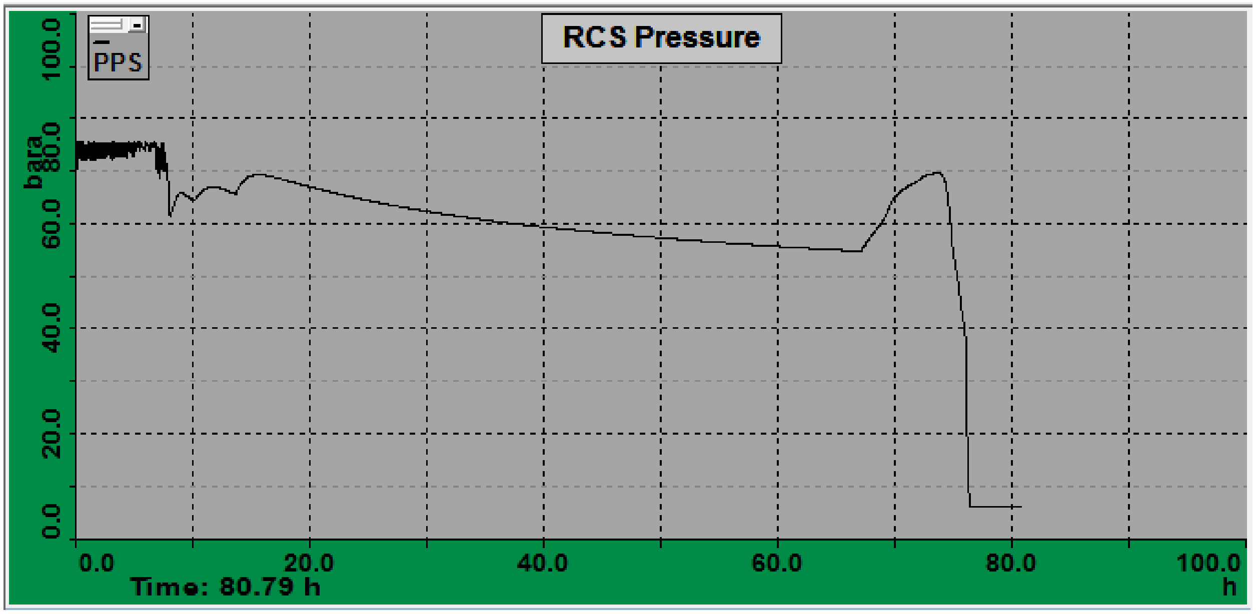

Figure 4-1
Reactor Coolant System pressures during the accident

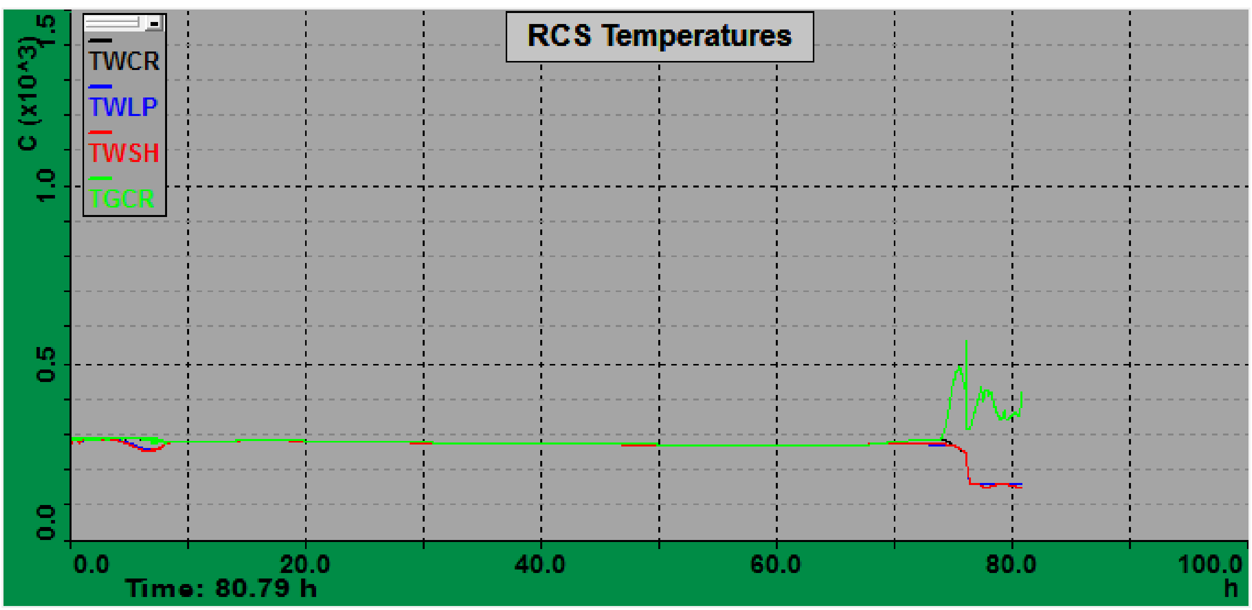

Figure 4-2
Reactor Coolant System temperatures during the accident

A key, but still unanswered, question concerns B-10. The experiments show that all, or nearly all, the boron is likely to be lost from the melted core region. But on the other hand if some remains behind, it can have an important effect on the reactivity. Mosteller showed that a core with melted control rods and still remain subcritical if as little as 5% of the boron remains behind dispersed into the moderator regions of the still more or less intact fuel assemblies. This is because of two effects, both of which introduce negative reactivity: reduced self-shielding of the

B-10, and more equal neutron flux in the fuel and poison. It was found that the negative reactivity worth of dispersed boron is a strong function of the penetration into the fuel assemblies, and moderator availability and location. Well dispersed boron can easily result in subcritical $k_{eff.}$

Retention in the core region of very small amounts of boron control rod material corresponds to high soluble boron concentrations in reflooding water. For instance, if 5% of the boron stayed behind post control rod eutectic formation, this would have the same negative reactivity worth as between 1600 and 2600 ppm of soluble boron, depending on pin cell geometries.

4.1 SERPENT Calculation at Predicted MAAP Conditions

Two major cases were simulated with the SERPENT Monte Carlo code. These are:

- **Case 1:** When the water level first reaches the top of active fuel (TAF), at 73.40 hours after scram
- **Case 2:** Core voided, blades intact, at 75.55 hours after scram.

Effect of Decreased Retention of Control Rod Material

The impact on reactivity on the redistribution of control rod material was also studied. The exact distribution of the control materials throughout an accident is highly speculative. However insights on this subject can be obtain via some simplified assumptions. In this study, it was assumed that during the early melt phase of the accident, the eutectic of SS-B_4C materials will start to melt, and candle down the relatively intact fuel assemblies and canisters. Some of this material is likely to remain behind in some form, plated out or otherwise sequestered in the core region. This study assumed that ~5% of the B_4C behaved in this manner, and made the further simplifying assumption that this could be modeled by assuming that the number density of boron in the core was 5% of its nominal value.

Effect of Adding Water and Soluble Boron in Reflood Stage

Severe Accident Management Strategies use reflooding the core as soon as possible if it becomes uncovered. To provide insights on this strategy, several series of calculations were performed. One assumed that the core became completely or partially voided, and that water reflooding was initiated. The assumption was that the reflooding would be relatively slow (perhaps 100 gal/min), with water entering the core region from the bottom. The intent was to study reactivity as a function of water height. As similar set of calculations was made using the assumption that the water was borated to ~2000ppm, a value that is close to that used in PWR for the boron letdown during normal operation.

Effects of Reflectors

Also studied was the effect of reflectors in the vicinity of the core, namely any water remaining either below the core or in the downcomer regions. The reflectors can have a significant effect on reactivity, and may or may not have been present in various phases of the 1F2 accident.

Case 1 – Water at Top of Active Fuel

The first case, the water level first reaches the top of active fuel (TAF). This occurs at approximately 73.40 hours after scram. At this time the core is still adequately cooled and the control blades and guide tubes are completely intact. This would be a 'normal' shutdown condition. This represents a base case for the recriticality investigation. Figure 4-3 is a snapshot of the core at this moment in time. There are 5 radial regions. In each radial region there is (going left to right) fuel, cladding, water, channel box, control blade material, and a second water channel. The status of the Balance of Plant at this time is given in Figure 4-4.

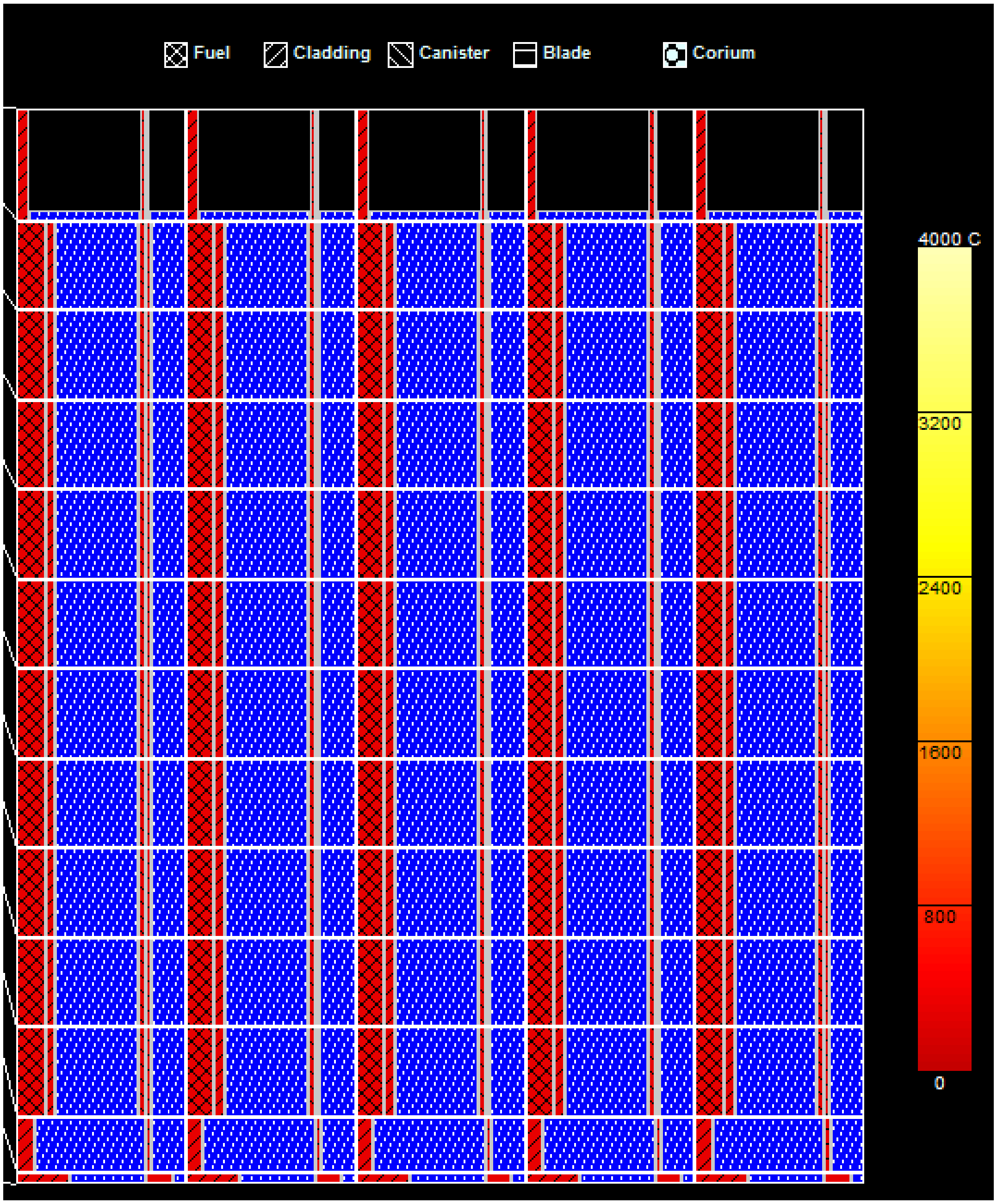

Figure 4-3
A cross section of the core region at 73.40 hours as calculated by MAAP5. The centerline of the core is at the left hand side of the figure. The temperature of each of the core materials is color code with temperature, see legend at the right side of the figure.

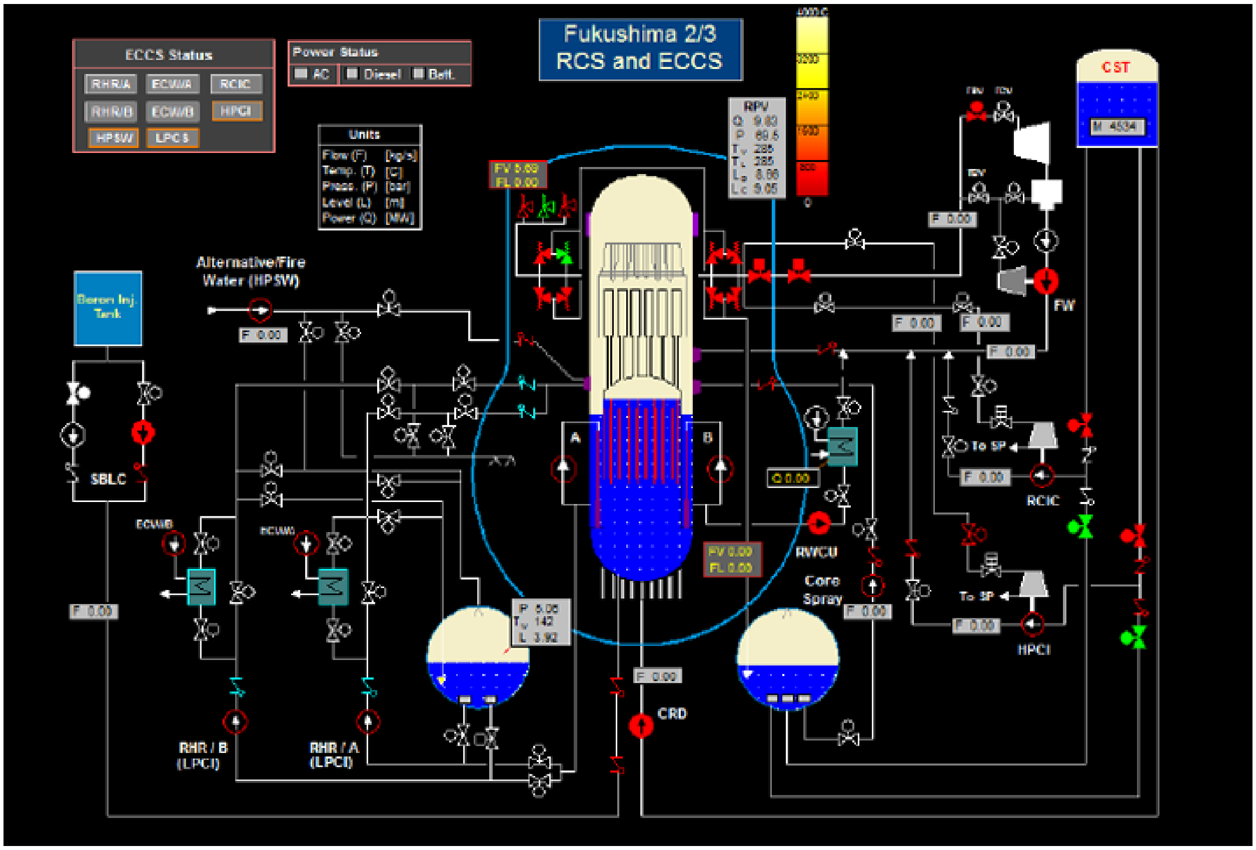

Figure 4-4
Schematic of the Reactor Coolant System at 73.40 hours. The figure shows the status of the core and water in the vessel. No significant core damage has occurred.

Case 2 – Core Voided, Blades Intact

By 75.55 hours, the water level has reached the lower nodes of the core region. Core cooling provided by steam from the lower reaches of the core has become insignificant and rapid core damage due to core heatup from decay heat is about to occur. Conditions for Case 2 are summarized in Figures 4-5 and 4-6.

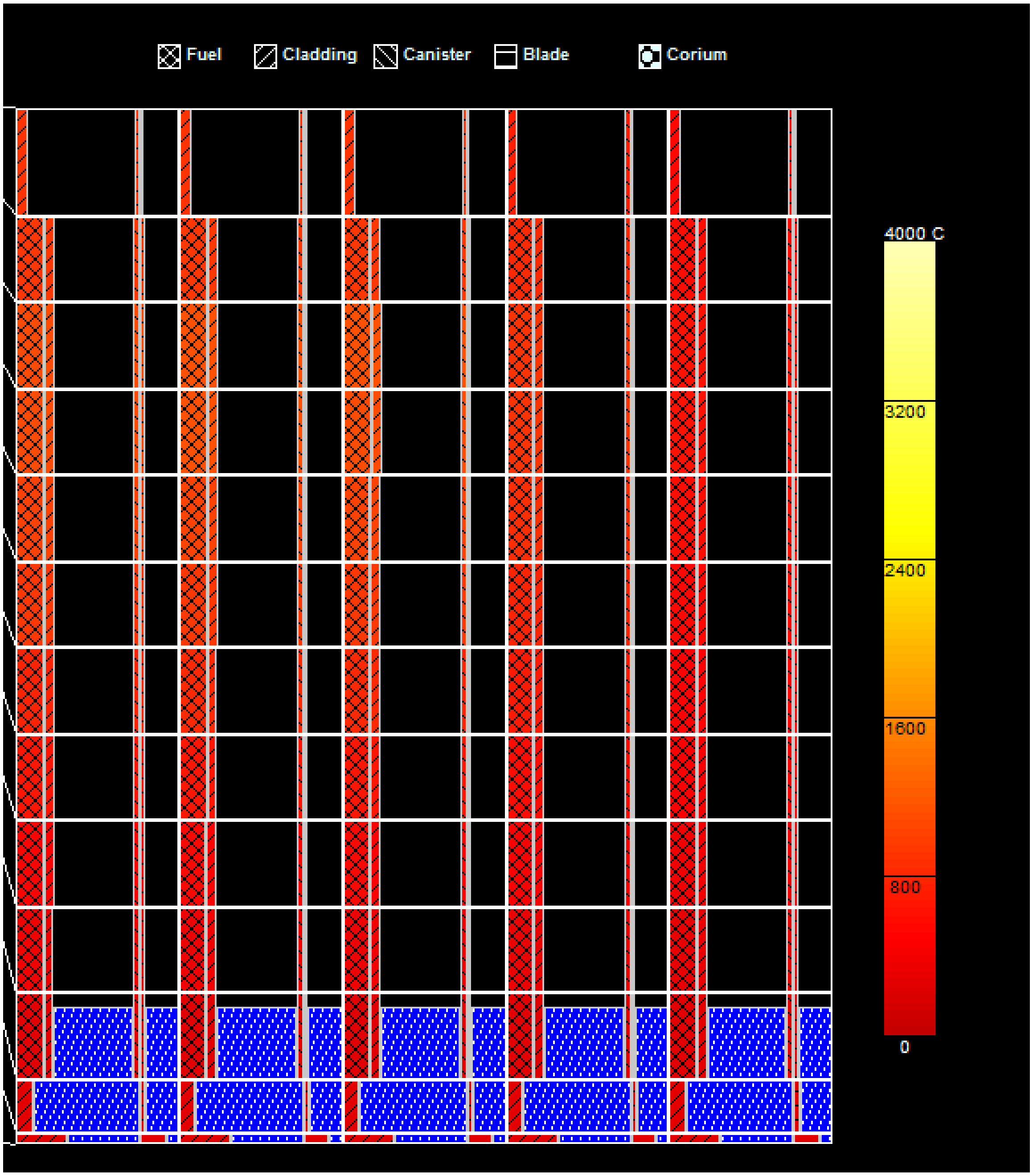

Figure 4-5
A cross section of the core region at 75.55 hours as calculated by MAAP5. The centerline of the core is at the right hand side of the figure. The core is nearly completely voided of water, except for the bottom nodes. Control materials have formed a eutectic material and are starting to disappear from the core.

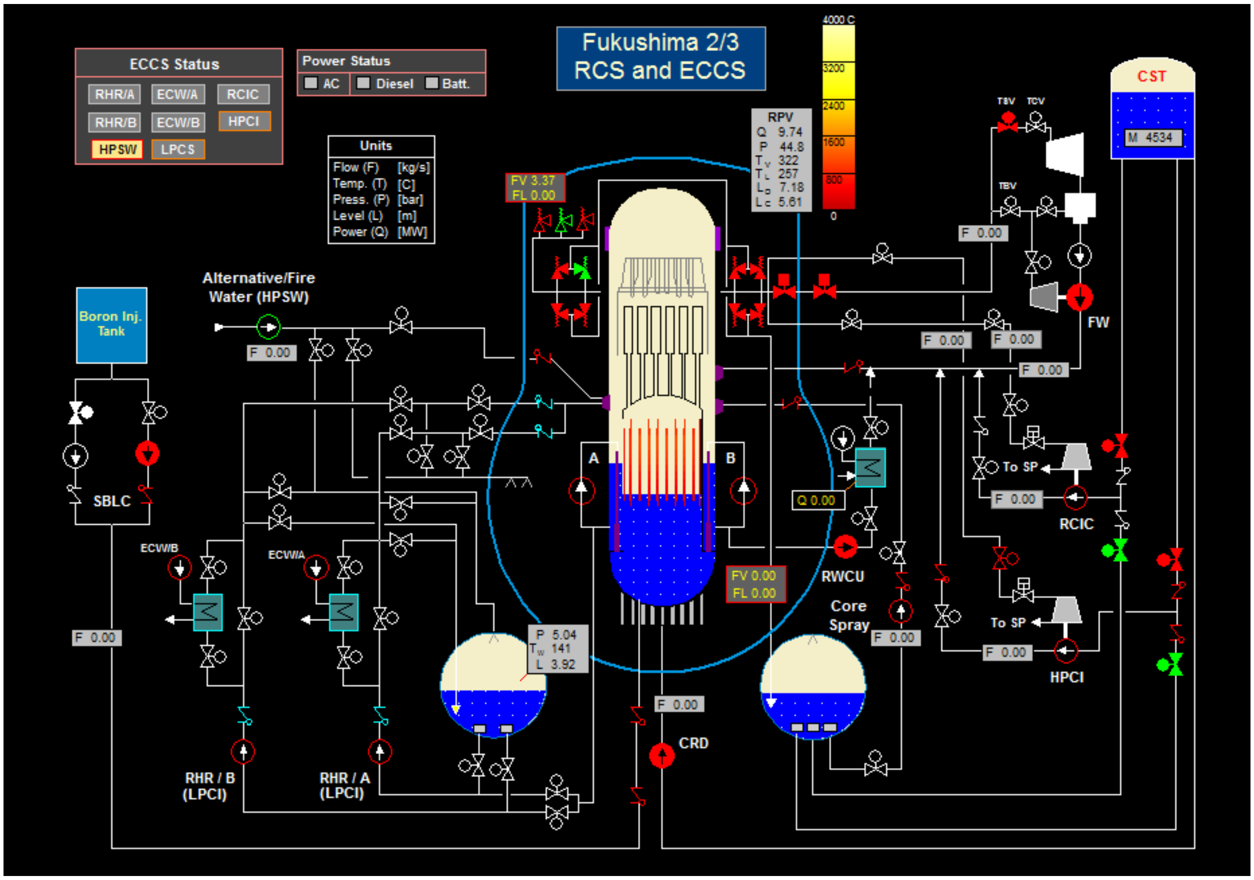

Figure 4-6
Schematic of the Reactor Coolant System at 75.55 hours. The figure shows the status of the core and water in the vessel. Core is nearly completely uncovered.

From the pressures and temperatures given in Figures 4-1 and 4-2, two simulation cases were developed: Case 1 was based on a pressure of ~70 bars and temperature of ~280C and Case 2 based on a pressure of ~6 bars and temperature of 160C. The SERPENT simulation for Case 1 resulted in an eigenvalue of 0.91032 (±16 pcm), while the second case with the all the coolant in the reactor assumed to be steam, the reactor is strongly subcritical with an eigenvalue of 0.46642.

As a further study of case 2, a sequence of cases was performed to simulate the boiling away all the coolant from the reactor. The simulation is performed by starting with reactor fully covered with saturated water and then the water level is reduced by voiding reactor in 25% increments as shown in the Table 4-1 below. The SERPENT model has 24 materials for the coolant in the assembly corresponding to 24 axial levels. There is a separate material for coolant between the assembly walls and the fuel pins (a thin layer) and a separate material for water between assembly walls. There are also separate materials for the top, bottom and side reflectors of the core.

In summary, the six conditions corresponding to the results in Table 4-1 are modeled in SERPENT as:

C 1: Full saturated water at 6 bars.

C 2: 25% void. The coolant at the top 6 axial levels + top reflecting region (i.e. above the core) is steam, below is water.

C 3: 50% void: The coolant at the top 12 axial levels + top reflecting region is steam, below is water.

C 4: 75% void: The coolant at the top 18 axial levels + top reflecting region is steam, below is water.

C 5: 100% void: All axial levels + top reflecting region is steam.

C 6: 100% void*: All axial levels + top reflecting + all the water in downcomer regions around the fuel assembly region is steam.

Each of these cases was performed with 4 core conditions:

1. All the control rods in
2. All the control rods out,
3. All the rods are in but plated out (simulated w/ 5% boron in the control rods)
4. All the rods are out but the coolant with 2000 ppm boron.

Table 4-1
SERPENT Monte Carlo Results with MAAP Thermal-Hydraulics Conditions

Coolant Condition	All Rods In	All Rods Out	Rods Plated Out (5% Boron left)	Borated Coolant 2000 ppm
full water	0.92859	1.10904	0.99931	0.80733
25% void	0.92714	1.10725	0.99752	0.81128
50%void	0.91782	1.09882	0.98818	0.82482
75%void	0.89098	1.07082	0.95966	0.83066
100%void	0.70235	1.02919	0.84469	0.83138
100%void*	0.46642	0.99420	0.50631	0.84234

When all the control rods are in the reactor, regardless of the coolant conditions the core is subcritical. When rods are out, the core is critical until all the water is boiled away. Even when the rods are 95% removed, they are still capable of maintaining the core subcritical for any coolant conditions. When borated water (2000ppm boron) is used to cool the core, the core is significantly subcritical for all water conditions. These results are shown in Figure 4-7.

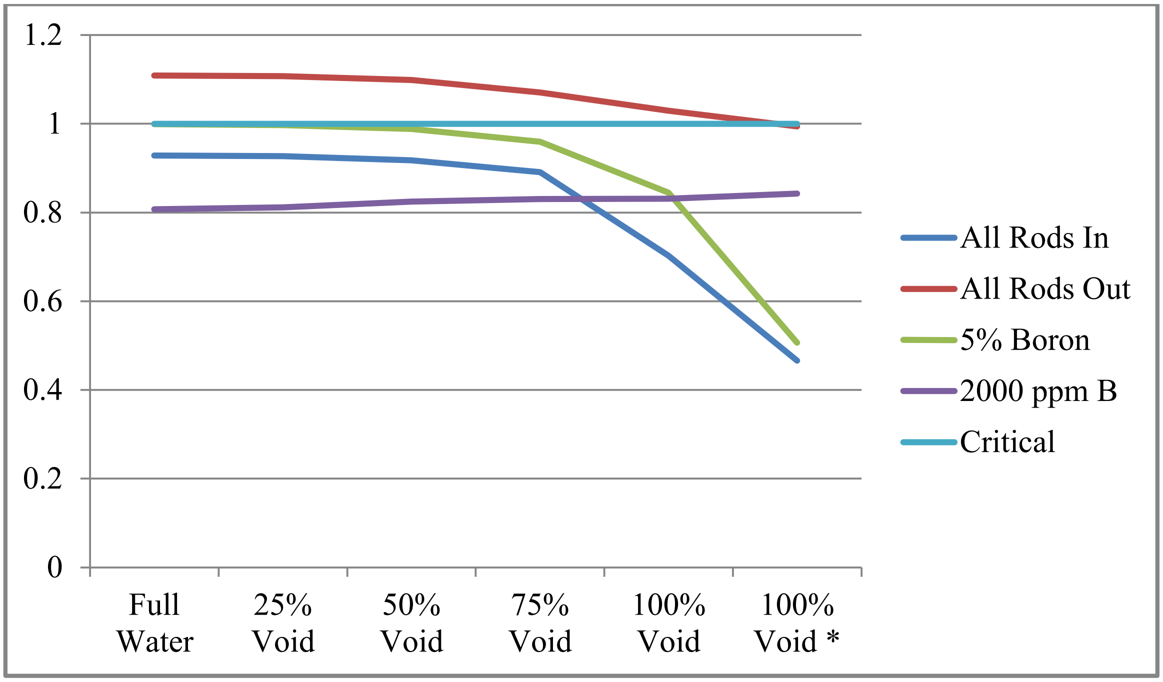

Figure 4-7
Results of the SERPENT Monte Carlo Calculations

5 CONCLUSIONS

Based on a conservative bounding analysis, this report concludes that there was a potential for recriticality in Fukushima 1F2 reactor if core reflood occurred after control blade melting has begun but prior to significant fuel rod melting. However, the MAAP analysis showed that there was insufficient time between the melting of the control materials and the collapse of the core for recriticality to have occurred, Even if water had been injected during this period the rate of injection during the accident probably would have precluded recriticality from occurring.

Although not in the scope of this report, other studies have shown that once core melting occurred, the state of the reactor became highly subcritical due to the disrupting the geometry into unfavorable configurations. This holds true even if all the core mass melted and accumulated in the bottom reactor head.

Although evolution and final location of the damaged core materials has not yet been ascertained, it is very unlikely that recriticality occurred any time during the event.

5.1 Insights of Significance to Reactor Safety

In this study it is clear that in situations where the control blade remained intact within the core, k_{eff} remained well below critical for all water levels inside the pressure vessel. However water that remained in the downcomer regions, and below the core, has an important effect. This is because water in these regions can reflect escaping neutrons back into the core regions. For the case when the core was under accident pressure and temperature conditions, the maximum k_{eff} was ~0.93 for a completely covered core with intact control rods.

For a hypothetical intact core that contained no control material at all, k_{eff} exceeded 1.0 (i.e. critical) for all water levels studied. Fortunately there does not appear to be a credible scenario that would lead to this situation. More likely is the scenario where a small fraction (~0.05) of the boron would remain behind in some areas of the core—and these scenarios lead to subcritical conditions. It should be noted, however, that the reactivity of the retained B^{10} is sensitive to the porosity and homogeneity of the eutectic materials formed during the accident and their degree of dispersal in the core region. This was not investigated in this study.

Of special value for accident management strategies, this study confirms that borated water would maintain the core in highly subcritical state.

5.1.1 Core Configurations and Conditions Subject to Recriticality

Qualitatively, previous studies of more-or-less intact cores concluded that recriticality is possible during reflooding with water of a partly degraded core with missing control materials. Under large additions of cold, unborated water recriticality could lead to sharp power peaks – possibly including prompt power excursions. More likely, quasi steady-state power generation would result. In such cases, the stabilized power seems to be below 20% of the nominal power. This would be about 20 times the decay heat rate, posing a severe heat removal problem. It is likely that the reactor power would adjust itself to a level corresponding to the power necessary to evaporate all water entering the core.

Recriticality phenomena are sensitive to thermal-hydraulic modelling, the specifics of accident scenario, such as system pressure and distribution of boron-carbide in the core, and the importance of multi-dimensional neutron kinetics for the determination of local power distributions in the core. The MAAP studies of 1F2 modeled the entire BWR primary system as realistically as possible in order to capture the reactor decay heat power - primary system behavior feedback effects.

5.1.2 Likely Effects if Recriticality were to Occur

If recriticality were to occur in a damaged reactor, a quasi-steady power level might result. In such a scenario the containment pressure and temperature would increase until the containment failure pressure is reached, unless actions are taken to terminate the event.

5.2 Prevention of Recriticality

While our current study shows a low probability for recriticality in the 1F2 event, other circumstances may not be as benign. If recriticality is of concern, certain steps could reduce its probability. In particular:

- If possible highly borated water should be used to cool the reactor. If borated water is not available, either seawater, or service water can be used (in that order of preference).
- Consider limiting the reflooding flow rate whenever control rod melting might be expected. The normal feed water should not be started.
- Limiting the maximum injection mass flow rate to less than ~25 gal/s in order to avoid the risk of fuel fragmentation and melting.
- Carefully consider depressurization of the primary system, under some circumstances, in order to limit relocation of control rod.

Developing Accident Tolerant Fuel (ATF) materials to replace materials that undergo rapid oxidation at low temperatures (specifically Zircaloy) with others (such as SiC, Molybdenum). This would extend the time before control elements melted and allow more time for boron injection. If ATF is used then the control materials must incorporate similar AT features.

6
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