EPRI MAAP5 FUKUSHIMA DAIICHI ANALYSIS

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ABSTRACT

The study presented in this paper summarizes work conducted as part of the EPRI Fukushima Technical Evaluation project. This effort is designed to develop a representation of the core damage events that occurred at Fukushima Daiichi Units 1, 2 and 3 during March 2011 using the analytical capabilities provided by the EPRI Modular Accident Analysis Program, Version 5 (MAAP5). The analytical investigations of Fukushima Daiichi performed with MAAP5 indicate that core melt progressions at Units 1, 2 and 3 likely span a range of core damage conditions. The core status at Unit 1 is likely consistent with a large fraction of core debris having relocated into the containment. By contrast, the MAAP5 evaluations indicate that there is a reasonable potential for a significant fraction of core debris to be retained inside the RPV at Unit 2. The corresponding Unit 3 simulations, however, highlight the important role that degraded HPCI operation at low RPV pressure may have played in promoting some relocation of core debris out of the RPV and into containment. The detailed containment evaluations conducted as part of this study also highlight the critical role played by thermal stratification phenomena (either in the suppression pool or in the drywell) in influencing the magnitude of containment pressure and thermal challenges. These simulations highlight the potentially critical role that thermal stratification in the upper drywell may have played in accelerating the onset of leakage through the drywell head flange due to thermal degradation of the drywell head gasket. Finally, these simulations of reactor and containment response provide good representations of the occurrence of flammable conditions in the Units 1 and 3 reactor buildings, supporting the nature and timing of the observed reactor building combustion events.

KEYWORDS
Fukushima Daiichi, severe accident analysis, core melt progression, MAAP

1. INTRODUCTION

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 were operating at full power at the time of the seismic event. Units 4, 5, and 6 were in shutdown. Unit 4 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 had been in shutdown 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 Unit 5, Unit 6 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.

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 tsunamis 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 Unit 1, Unit 2, Unit 3, and 1F4. The flooding also resulted in all direct current (dc) power being lost at Unit 1 and Unit 2. Some dc power sources survived at Unit 3. 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 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 (Unit 1, Unit 2, and Unit 3). With the Unit 1 and Unit 2 control room and associated reactor buildings in darkness and operators at Unit 3 attempting to maintain core cooling with limited battery power, the capability to identify and maintain the condition of the reactor cores was severely compromised. Because of limited means to cope with the most severe challenge to a nuclear power plant's critical safety functions, the conditions at Unit 1, Unit 2, and Unit 3 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 Unit 1, Unit 2, and Unit 3 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 reactor buildings, 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.

The subsequent discussion of severe accident progression at Fukushima Daiichi Units 1, 2 and 3 is based on analytical investigations using the MAAP5, Version 5.02, computer code. This study is based on, and extends the work, from the first phase of the Fukushima Technical Evaluation [1]. The reported analysis identifies key features of core melt progression, containment response and the development of reactor building flammability. It should be noted that discussion of reactor building atmospheric conditions is not provided for Unit 2. The open blowout panel on the refuel floor of the Unit 2 reactor building would have prevented the development of flammable conditions inside the reactor building. Therefore, unlike the Units 1 and 3 reactor buildings, flammable gas combustion did not occur at Unit 2. The reactor building response at Unit 2, therefore, does not provide a meaningful signature of its accident progression.

2. MAAP5 MODEL FOR FUKUSHIMA DAIICHI ANALYSIS

This section describes the MAAP5 models developed to represent the Fukushima Daiichi plant, including the reactors, containments and reactor buildings. The overall nodalization scheme adopted is the same for Units 1, 2 and 3. Identical containment and reactor building models are used for Units 2 and 3—these units are essentially the same from the perspective of a lumped volume approximation for the containment and reactor building. However, the Unit 1 model is distinct from the Units 2 and 3 models. This is due to the differences in volume between these units.

3. UNIT 1 BEST ESTIMATE EVENT PROGRESSION ANALYSIS

3.1 Core Damage Progression

Following the earthquake on March 11, 2011, the Isolation Condenser (IC) became the primary means of removing decay heat generated within the Unit 1 reactor core. The IC is a heat exchanger that helps cool the reactor system while isolating it from the outside environment. Following the seismic event, electrical power was available to operators to control the IC. The IC Train B was initially operated but was taken out of service because of concerns over the observed rapid cooldown rate exceeding operating procedures (i.e., in excess of 100°F/h). The IC Train A was then operated to maintain reactor pressure by periodically closing and opening an outside-of-containment isolation valve (MO-3A).

Immediately prior to the loss of all ac and dc power at Unit 1, one of the IC isolation valves was closed. Closure of this isolation valve was done as part of the controlled depressurization of the RPV being performed by Unit 1 operators. Without any indications available to the operators in the control room, the status of core cooling could not be confirmed after the loss of instrumentation caused by the loss of all electrical power. However, by 3½ hours after the earthquake, some electrical power had been restored to the control room. Operators were then able to confirm that the IC had most likely been isolated from the RPV from the time power was lost. Subsequent attempts were made to operate the IC. Some functionality was restored, indicated by limited steaming from the IC Train A condenser tank. This also likely indicated a reduction in the total amount of heat that could be removed by the IC A train. Such a reduction in heat removal capability can occur if hydrogen gas plugs the horizontal condenser tubes. However, it is not known to what extent the heat removal from the isolation condenser was degraded at this time; the performance of the IC is still under investigation. In this study, no credit for IC operation is taken after the loss of all electrical power.

It was not until 15 hours from the time of the seismic event that water injection into the Unit 1 RPV was established. Using fire engines, water was injected into the RPV through the fire protection system connection to the core sprays. Because the reactor vessel and containment drywell pressures were quite high (twice containment design) by this point, actual water injection from the fire engine pump would have been limited by system backpressure. Not until approximately one day after the loss of electric power—when operator-initiated venting occurred—did the drywell pressure drop to a level that allowed higher rates of injection.

The following assumptions used with MAAP5 provide the most reasonable calculations of the observed reactor and containment thermalhydraulic conditions during the Unit 1 event. The core went without cooling from the time the IC A train was isolated from the RPV just prior to the loss of all electric power. The fire water cooling restored approximately 15 hours after the earthquake was insufficient to arrest significant core damage. The in-core instrument tubes may have failed, providing a small leak path into the containment. This failure is assumed to occur when the core reaches temperatures in the regime of steel melting. Around 5 hours into the event, RPV depressurization could have occurred. It is assumed tha that the RPV depressurized through the seizure of a safety relief valve (SRV) under severe core damage thermal-fluid conditions. It is possible that other means of RPV depressurization contributed; for example, main steam line creep rupture or enhanced RPV seal leakage. This minimal set of plausible assumptions represents well the observed Unit 1 reactor conditions, as shown in Figure 1 depicting the simulated and observed RPV pressures.

Detailed evaluation of the RPV water inventory transient is not discussed further here because of the difficulties interpreting measurements under conditions when the RPV pressure has either decreased significantly or high temperatures prevail in the containment. Such conditions tend to result in the reference leg of the water level instrumentation becoming depleted of water, with measurements reflecting unknown conditions in the instrumentation piping rather than the actual RPV.

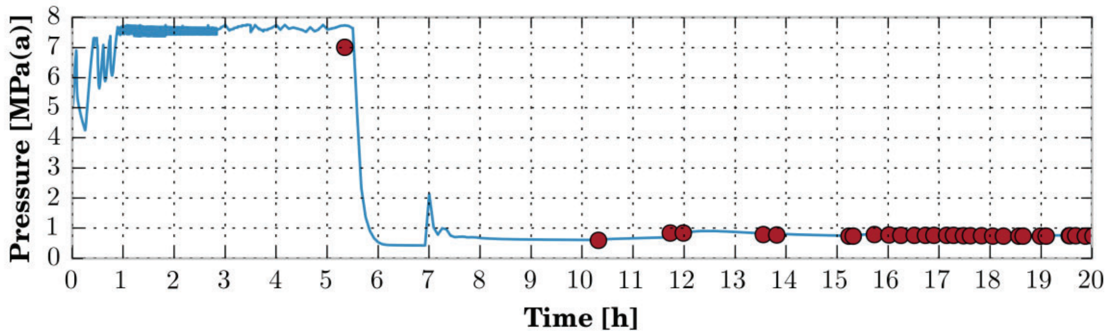

Figure 1. MAAP5 Simulation of Unit 1 RPV Pressure Transient

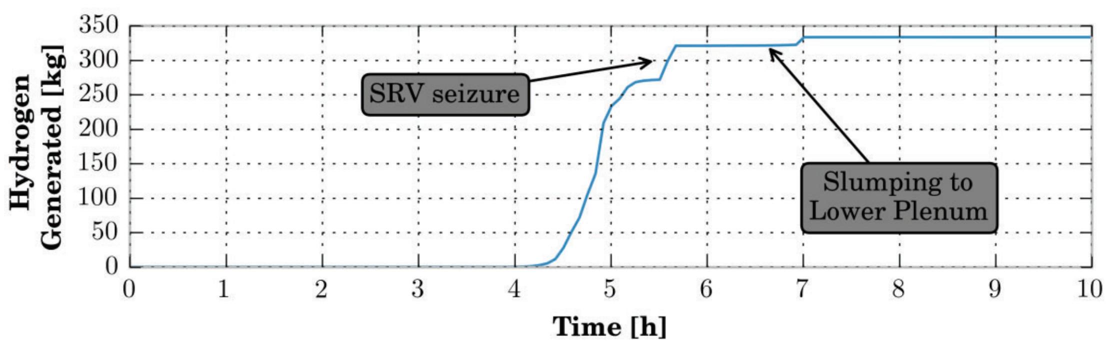

Figure 2. MAAP5 Unit 1 Simulation of In-Vessel Hydrogen Generation

The corresponding in-vessel hydrogen generation transient is shown in Figure 2. This corresponds to a relatively low hydrogen generation (about a total of 325 kg generated in-vessel). The failure of the RPV lower head just after the 12-hour mark, however, leads to substantial hydrogen (and carbon monoxide) generation due to unmitigated CCI. The generation of hydrogen due to CCI ultimately ensures that there is substantial hydrogen available to leak into the reactor building, independent of the amount simulated to be generated in-vessel. Thus, the likely relocation of core debris ex-vessel at Unit 1 prevents further assessment of whether or not current models of in-vessel hydrogen generation are truly representative of what occurred during the Fukushima Daiichi Unit 1 event progression.

3.2 Evolution to Containment Impairment

The following accident progression characteristics provide a representative simulation of the observed Unit 1 containment response.

- A steam leak from the RPV into the drywell after 5 hours, sufficient to depressurize the RPV
- Drywell head lifting starting around the 12-hour mark

- Drywell head impairment around 13 hours, inducing a small leak in the drywell (greater than about 10 cm2)—this lifting of the drywell head would have corresponded to an observed drywell pressure around twice the design pressure
- The drywell head may not have completely resealed following its initial lifting when operators reduced containment pressure by venting; however, it is assumed to reseat in this analysis
- RPV lower head breach around 12 to 13 hours
- Low water injection rates into the RPV beginning at about 15 hours (significantly less than 10 gpm)
- Wetwell venting around 23.8 hours for about 30 minutes

The overall containment pressurization is well-represented based on these event scenario assumptions. The simulation of the drywell pressure transient, compared against the observed drywell pressure, is shown in Figure 3.

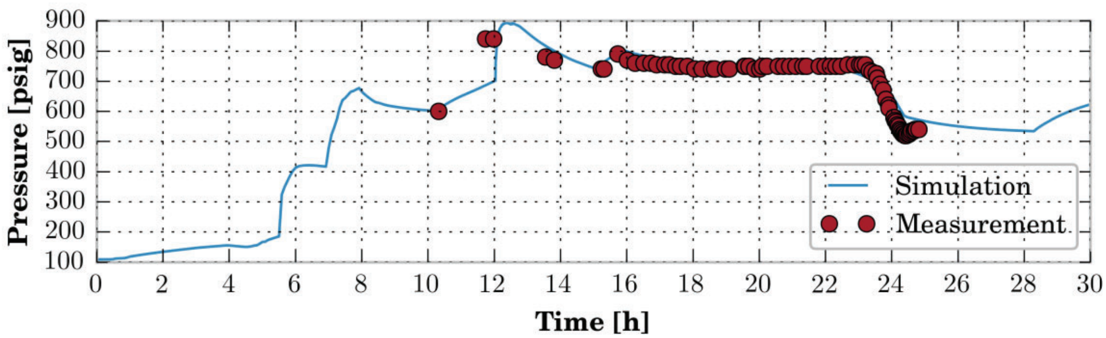

Figure 3. MAAP5 Simulation of Unit 1 Containment Pressure Transient

The corresponding drywell temperature distribution is shown in Figure 4.

Prior to the onset of containment impairment between the 12-hour and 13-hour marks, the upper drywell temperature is found to be below about 500°F. Temperatures at or above this level are necessary to induce degradation in silicone rubber seals exposed to a steam environment [7]. This type of seal is used for the drywell head gasket and would have been exposed to a steam environment beyond about the 12-hour mark as a result of drywell head lifting. The simulation of a relatively slow build-up of temperature in the upper drywell of Unit 1 tends to support the possibility of limited degradation of the drywell head gasket at the time drywell head lifting could have started due to high drywell pressure.

As a result of the slow increase in simulated upper drywell temperature, the Unit 1 event progression appears to have been characterized by conditions in which drywell head leakage may not have commenced immediately upon drywell head lifting. The response of the Unit 1 drywell head gasket, thus, could have been initially governed by seal spring back, in which the still elastic seals filled-in gaps in the head created by lifting of the drywell head. This would have delayed the onset of leakage despite drywell head lifting. Such a response has been noted in experimental tests of drywell head flange systems (e.g., the work of Hirao et. al. [7]) as well as in past Chicago Bridge and Iron (CBI) studies of Mark I containment response [8]. This type of drywell response would explain key Unit 1 observations: the containment pressure reaching twice design pressure approximately an hour prior to an increase in site boundary dose rates [1].

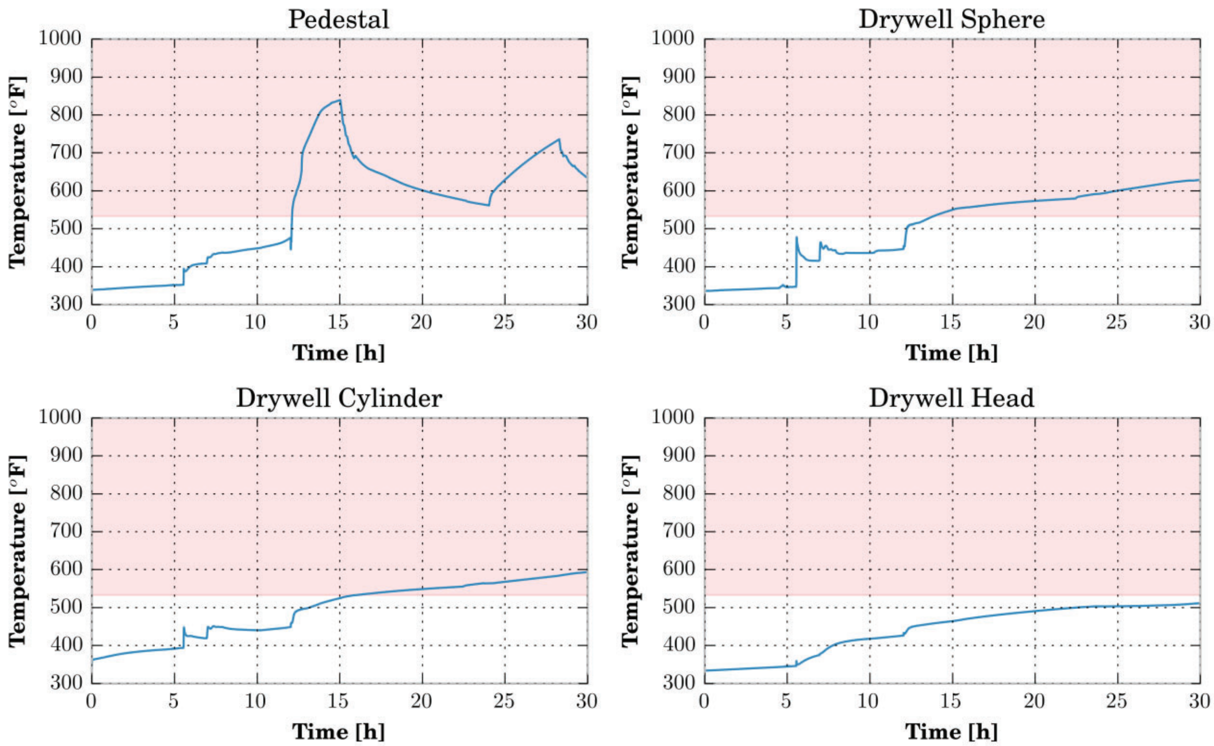

Figure 4. MAAP5 Unit 1 Simulated Drywell Temperature Distribution

3.3 Flammable Gas Build-up in Reactor Building

The simulated Unit 1 flammable gas build-up in the reactor building is shown in Figure 5. The accumulation of hydrogen on the refuel floor is relatively slow. After the onset of drywell head lifting and leakage, approximately 2 hours elapses before the concentration of hydrogen exceeds that sufficient to support an energetic combustion event. By the time of the energetic combustion event at Unit 1 (24.8 hours), the hydrogen concentration on the refuel floor is about 15%.[1]

Based on the results shown in Figure 5, an energetic combustion event could have occurred at any point between 15 and 24.8 hours. The energetic combustion event, however, may not have occurred until 24.8 hours due to the lack of an ignition source. It may not have been until this point in time that efforts to restore power to the plant generated the necessary spark to ignite the refuel floor atmosphere.

Figure 5 also shows a slow build-up of hydrogen at lower elevations. The pressurization of the refuel floor due to the combined steam and hydrogen leakage from the drywell head flange promotes flows off the refuel floor through the large openings provided by open stairwell doorways. The Fukushima Daiichi reactor buildings have openings from each reactor floor into the reactor building stairwell that runs the height of the building. By approximately 24.8 hours, the concentration of hydrogen at lower elevations is not found to exceed the lower limit for flammability (about 4% in dry air).

This distribution of hydrogen is consistent with the type of damage that occurred to the Unit 1 reactor building—the damage to the structure was localized to the refuel floor. The drywell head leakage scenario is thus a relatively plausible characteristic of the Fukushima Daiichi Unit 1 event progression.

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1 The combustion of hydrogen is artificially suppressed in these simulations to mimic the absence of an ignition source. The concentration of hydrogen beyond T+24.8 hours is an artifact of the simulation and not reflective of the hydrogen concentration in an open refuel floor after this time.

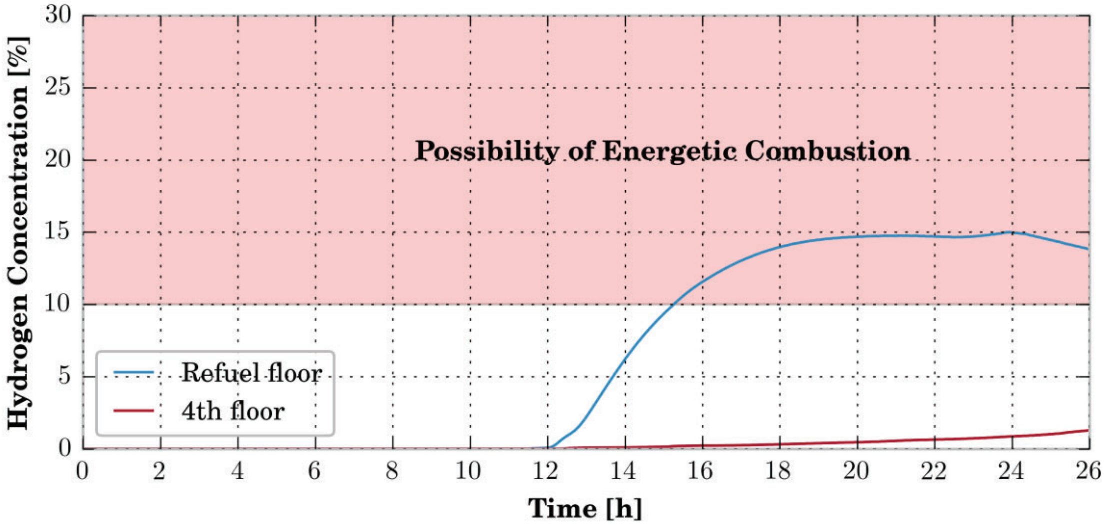

Figure 5. MAAP5 Unit 1 Simulation of Hydrogen Build-up in Reactor Building

4. UNIT 2 BEST ESTIMATE EVENT PROGRESSION ANALYSIS

4.1 Core Damage Progression

Unlike the accident progression at Unit 1, core cooling was not lost at Unit 2 immediately following the loss of power across the station. Unit 2 is a BWR equipped with a reactor core isolation cooling system (RCIC). It uses 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 itself.

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 RCIC is operating normally. However, if the height of water in the vessel rises too high—at or above the level of the main steam line (MSL)—water could flood into the RCIC turbine. Damage to the RCIC turbine could result, 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, and 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 electrical power at Unit 2, 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. When dc power is available, 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. In this manner, operators can achieve a desired RCIC injection rate when dc power is available for control.

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 the level of the MSL penetrations.

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 Unit 2 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 established following the depressurization of the vessel when operators opened an SRV over 5 hours later.

The MAAP5 computer code simulations described in this paper and the first phase of the EPRI Fukushima Technical Evaluation [1] 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 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 (see Figure 6). This would have allowed both water and steam to be discharged from the vessel into the RCIC turbine. As a result, the rate of loss of coolant mass (water and steam) from the vessel would have increased relative to the rate of mass loss had only steam been discharged to the RCIC turbine.

RPV pressures are available 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 (see Figure 7). In addition, to capture the RPV water level holding around the level of the MSL penetrations, the rate of RCIC water injection was adjusted 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 (Figure 5 and Figure 6).

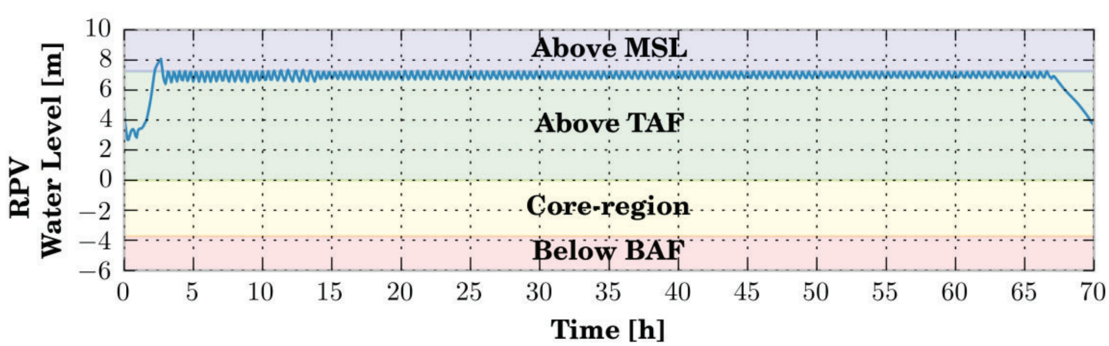

Figure 6. MAAP5 Unit 2 Simulation of RPV Water Level

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 produced in the fuel. The potential for water discharged from the fire engine pumps bypassing the RPV has been noted by TEPCO post-accident investigations [6].

This set of physically plausible assumptions represents the observed Unit 2 reactor conditions very well.

Following the loss of the RCIC system, core cooling was not restored until after 75 hours and following depressurization of the RPV by deliberate opening of an SRV. The RPV partially re-pressurized 3 times between 75 hours and 85 hours most likely due to SRVs re-closing and progressive core damage (see Figure 8). The re-pressurization of the RPV during this period likely resulted in degradation of fire engine injection due to high back pressure. As a result, the MAAP5 simulation predicts progressive core melting and in-vessel hydrogen generation over this period (see Figure 8). Of particular importance during the second re-pressurization period shown in Figure 8 is the enhanced hydrogen generation found in the MAAP5 simulation. This likely could have resulted due to relocation of some core debris (e.g., molten metals from dissolved core structures) into the lower plenum, causing increased steam generation and concomitant in-vessel hydrogen generation.

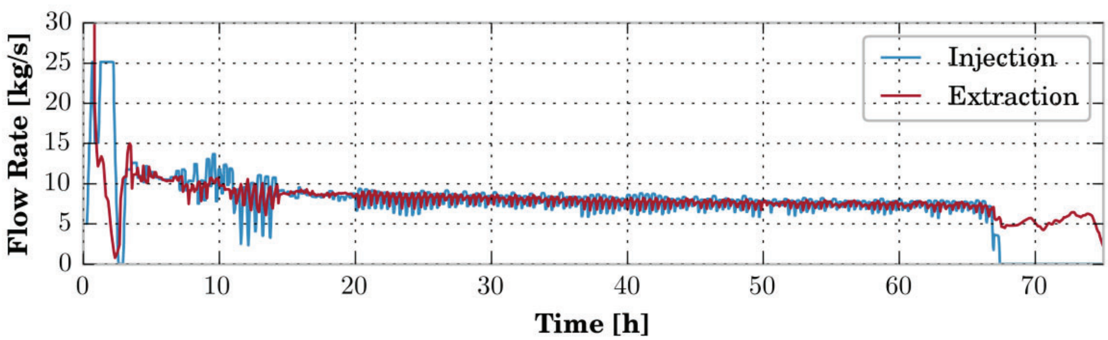

Figure 7. MAAP5 Unit 2 Simulation Assumed Operation of RCIC System

Core melt progression was relatively stable until about T+94 hours when core melt relocation either into the lower plenum or reactor pedestal may have occurred. The MAAP5 simulation for Unit 2 finds core melt relocation to the lower plenum is a possible explanation of the rapid rise in RPV pressure around this time. A rise in containment pressure at this time also occurred. This is discussed further below.

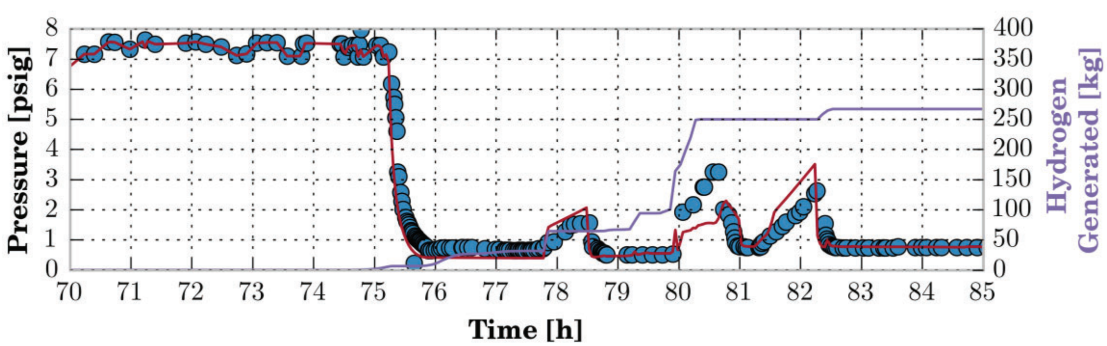

Figure 8. MAAP5 Unit 2 Simulation of Late-Phase RPV Pressure Transient

4.2 Evolution to Containment Impairment

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 (see Figure 9). During RCIC operation, the MAAP5 simulation assumes that torus room flooding occurred. This has been hypothesized as the primary reason for the control of containment pressure below design during the nearly three day period of RCIC operation [5, 6].

Such torus room flooding would have enhanced heat rejection through the steel walls of the torus, providing a means by which heat could be rejected from the suppression pool water. In this MAAP5 simulation, this heat rejection from the suppression pool is simulated through an energy removal term in the simulation. To achieve the pressurization observed during the three day period over which the Unit 2 RCIC system operated, a heat removal of about 5 MW is required to maintain the suppression pool water temperature sufficiently low. Enhancements to this modeling are currently underway, based on implementation of a multi-zone representation of the suppression pool in the MAAP5 computer code.

These modeling enhancements are also necessarily to mechanistically capture the containment pressure response observed between 70 hours and 75 hours (see Figure 9). During this period in the event, SRV actuation actually resulted in a decrease in the containment pressure. Separate calculations, using a generalized buoyant jet model and multi-zone suppression pool, indicate that SRV discharge into a thermally stratified suppression pool can result in a decrease in the temperature at the surface of the pool. These calculations indicate that a possible mechanism to explain this can occur due to the pool circulation induced by the buoyant jet emerging from the lower elevation SRV sparger and rising toward the pool surface. As this buoyant jet propagates through the suppression pool, it entrains colder fluid from below the thermally stratified layer formed above the RCIC sparger. The buoyant jet carries this entrained fluid toward the upper surface of the suppression pool. This motion has the effect of mixing colder fluid into the hotter fluid of the stratified layer formed above the RCIC sparger. The net effect can be a reduction in the temperature of the upper surface of the suppression pool.By 80 hours into the event, containment pressure escalated by about 40 psig to approximately 1.5 times design pressure, as shown in Figure 9. This corresponded to a period of enhanced hydrogen generation in the Unit 2 simulation, which is shown in Figure 8. To completely represent the magnitude of containment pressurization at this time, however, it was necessary to assume a direct leakage pathway between the RPV and drywell. This type of leakage pathway could have been established by failure of Transverse In-core Probes (TIPs). Such a failure likely occurred during the TMI-2 event [9]; it is found to occur in this MAAP5 simulation due to the progressive nature of core damage. It is also important to note that this period of containment pressurization corresponded to the onset of an increase in the measured drywell dose rate.

The potential for drywell head lifting after T+80 hours cannot be excluded. All operator attempts to control Unit 2 containment pressure after the loss of RCIC failed. Based on the available measured drywell pressure data, as well as TEPCO post-accident investigations, venting through the wetwell or drywell did not occur. There are, however, no clear signatures in the site boundary dose rate measurements that could account for appreciable leakage from the Unit 2 drywell head over this period. Low levels of leakage from the Unit 2 drywell head, however, are plausible over this period since the magnitude of radiological release would not necessarily have been detectable given the already high level of background radiation levels at the site boundary by this point in the vent.

The Unit 2 containment subsequently depressurized at about 88 hours; this was not the result of successful operator initiated venting. The potential exists for a gross loss of sealing capability at this point due to prolonged exposure to a drywell atmosphere with high steam content at high temperature (see Figure 10). The notable increase in the site boundary dose rates at this time lends further support to gaseous leakage through an impaired containment. Given the fact that significant post-accident radiation levels have been observed above the drywell head at Unit 2, the drywell head flange is a plausible location for containment impairment at this time.

The Unit 2 containment subsequently experienced increases in containment pressure that were limited to approximately the containment design pressure; one such period of re-pressurization occurred at about 94 hours. As discussed above, this period of drywell re-pressurization is explained in the MAAP5 simulation by slumping of core debris into the lower plenum. The pressurization also occurred coincident with an increase in the measured drywell dose rate. This provides a further indication of this period of Unit 2 containment pressurization occurring due to molten core debris relocation into a water pool.

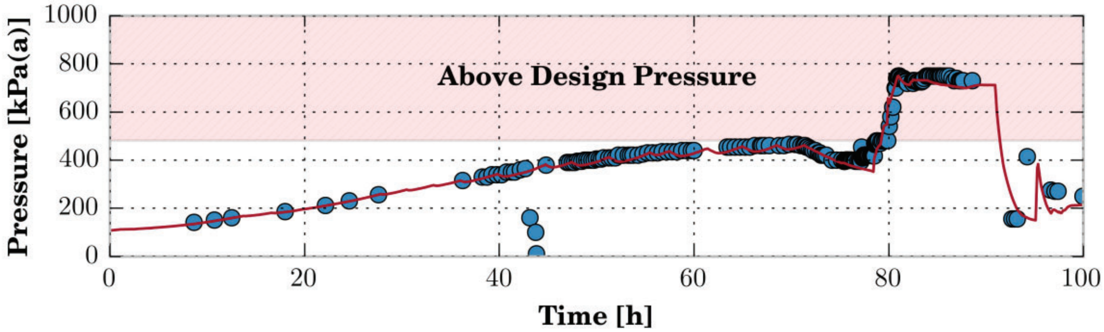

Figure 9. MAAP5 Unit 2 Simulation of Containment Pressure Transient

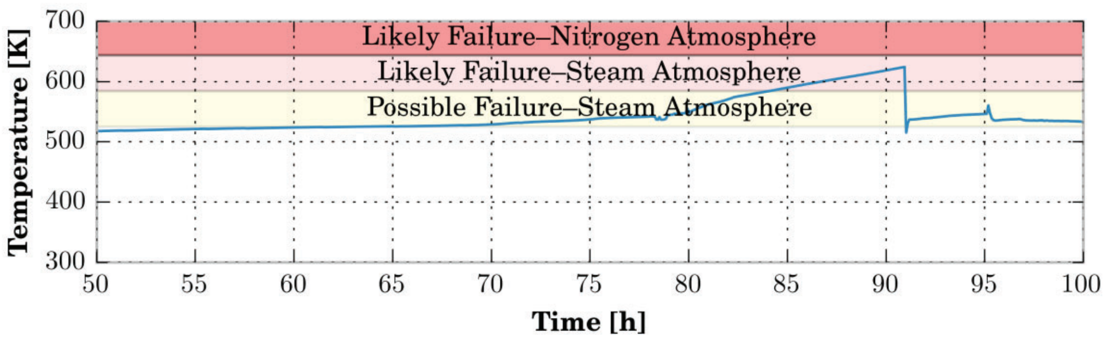

Figure 10. MAAP5 Unit 2 Simulation of Upper Drywell Temperature Transient

5. UNIT 3 BEST ESTIMATE EVENT PROGRESSION ANALYSIS

5.1 Core Damage Progression

Following the loss of power at Unit 3, core cooling was maintained by the RCIC system. The tsunami flooding at Unit 3 did not lead to the loss of all dc power, unlike the complete loss of power that occurred at Units 1 and 2. As a result, the operator control of the Unit 3 RCIC system was maintained. This RCIC system is identical to that described in the summary of the accident progression at Unit 2.

During the 20 hours from the time of shutdown over which RCIC operated, operators were able to control the amount of water added to the RPV by the RCIC system. Therefore, the water level in the vessel never rose to the level of the MSL penetrations. The RCIC system at Unit 3 operated under conditions for which it was designed.

The steam extracted from the vessel drove the RCIC turbine throughout this period, providing continuous injection of water from the CST into the vessel. If the RCIC system automatically stopped because of high water level in the vessel, operators would have to restart it—which would have more quickly exhausted the batteries needed to control it. To preserve battery life, the operators maintained the water level in the vessel by diverting a portion of the RCIC pump discharge back to the CST and adjusting the amount of steam drawn from the vessel to drive the RCIC turbine and power its pump. Despite these attempts to ensure continued RCIC system operation, it abruptly failed at about 20 hours and could not be restarted.

By about 21 hours, core cooling was restored when the high-pressure coolant injection (HPCI) system automatically started in response to a drop in the water level in the vessel. Similar in principle to the RCIC system, the HPCI system takes steam generated inside the vessel and uses it to drive a turbine that powers a pump. As with RCIC, the HPCI pump draws water from either the CST or the suppression pool to add water to the vessel. During the event at Unit 3, the HPCI system took water from the CST.

The HPCI system has a turbine that draws significant steam off the RPV, providing more power to the HPCI pump to rapidly add water to the vessel. The operators controlled the amount of water added by the HPCI system to the vessel by 1) diverting a fraction of the water pumped through the HPCI pump back to the CST and 2) adjusting the amount of water drawn by the pump from the CST by changing the rate of steam extraction from the RPV to drive the HPCI turbine.

However, a significant amount of steam must still be drawn from the vessel to maintain adequate rotation of the HPCI turbine. This means that a large amount of energy—more than that generated by decay heat inside the fuel—is taken out of the vessel in order to keep driving the turbine. This caused the pressure in the vessel to start dropping just after HPCI operation began. By about 28 hours, the vessel had depressurized to the point at which the amount of energy carried by the steam was significantly reduced. The rate of turbine rotation therefore dropped low enough that sufficient power to the HPCI pump could not be provided. The rate of water addition to the vessel would be significantly reduced as a consequence, and likely was insufficient to make up for the steam generated by the decay heat.

Because of issues with operating the HPCI system at such low vessel pressures, operators took the HPCI system off-line at about 36 hours. Attempts to provide core cooling with diesel-driven fire pumps after HPCI was taken off-line were not successful, with the RPV pressure rapidly increasing above the fire pump shutoff head. Core cooling was not restored until the pressure in the vessel suddenly decreased at about 42 hours, likely due to actuation of the Automatic Depressurization System (ADS) [5, 6]. Injection with fire engine pumps restored some amount of core cooling, with brief interruptions, after the vessel pressure was reduced.

The MAAP5 simulations described in this report have assessed the Unit 3 sequence of events against the observed thermal-hydraulic signatures of accident progression (see Figure 11). These simulations indicate that the following assumptions provide the most reasonable representation of the observed reactor and containment thermal-hydraulic conditions during the event.

- The amount of steam drawn and water added to the vessel during RCIC and HPCI operation is varied. This simulates operator control of these systems to maintain the water level in the vessel at a fixed level above the fuel.
- During HPCI operation, when the pressure in the vessel drops to about 1 MPa(a), it is assumed that the injection rate to the vessel is not sufficient to remove all of the decay heat. The rates of water addition to the vessel that are most representative of the observed pressure and water level in the vessel are about 2–3 kg/s. Operation of the HPCI system at low RPV pressure after about 28 hours likely resulted in a reduction in water injection rate and an inability to maintain water level—by 36 hours, it is likely that RPV water level had reached TAF.
- From 36 hours, RPV pressure was controlled by SRV cycling and no water was injected due to the high pressure.
- Consistent with the observed vessel pressures, the pressure in the vessel is assumed to drop to the pressure in containment resulting from ADS actuation just prior to 42 hours [5, 6].

- After 42 hours, the rate of water addition by the fire engine pump is assumed to be at or below the minimum rate required to remove all decay heat by boiling of the injected flow [1].[2]

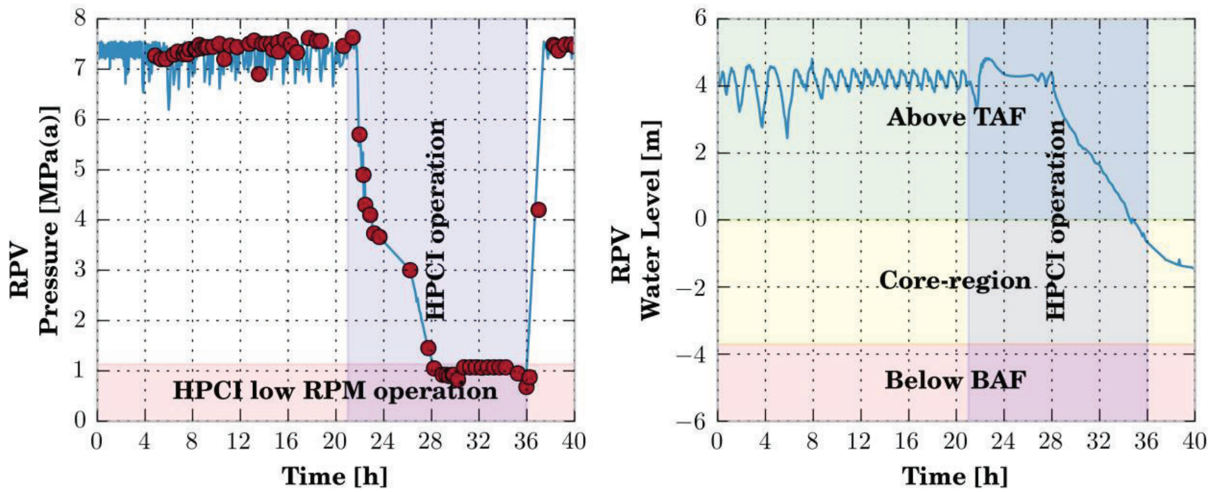

Figure 11. MAAP5 Unit 3 Simulation of Reactor Response

These simulations of core melt progression are not capable of uniquely distinguishing between in-vessel and ex-vessel core melt progression. Both conditions are plausible representations of the available information regarding reactor response. There is a very large sensitivity to the potential for RPV lower head breach identified in these simulations, influenced primarily by the assumed effectiveness of degraded HPCI water injection from about 28 hours until 36 hours. The resulting very small change in the RPV water level at 36 hours, either slightly below or slightly above TAF depending on the assumed HPCI function, can either promote the potential for in-vessel retention or RPV lower head breach.

5.2 Evolution to Containment Impairment

The key assumptions made in performing this best estimate representation of the Unit 3 containment response are as follows. These assumptions have been found necessary, as part of the first phase of the EPRI Fukushima Technical Evaluation [1], to capture the observed Unit 3 containment pressurization transient. The MAAP5 simulation of the Unit 3 containment pressurization is shown in Figure 12.

Either gas-phase leakage from the cycling SRV or recirculation pump seal leakage is assumed to occur after the first couple of hours of RCIC operation. Either of these mechanisms provides a means of representing the strong pressurization of the containment during the 20-hour period of RCIC operation. It is important to note that some of the contribution to the observed pressurization could have arisen as a result of thermal stratification in the Unit 3 suppression pool.

GOTHIC analyses conducted as part of the overall EPRI Fukushima Technical Evaluation effort, however, have indicated that the combination of RCIC and SRV sparger discharges into the Unit 3 suppression pool during this period would not have generated sufficient stratification to explain the observed pressurization [10]. The SRV discharges would have induced circulation currents in the suppression pool that had the effect of disturbing the more pronounced thermal stratification that tends to develop during periods of RCIC operation [10]. Such strong thermal stratification during RCIC operation was observed during a Browns Ferry Nuclear Plant (BFNP) Unit 2 test, against which the GOTHIC model has been benchmarked [10].

2 Fire engine injection was likely degraded due to flow bypass through the condensate transfer pump [6].

These GOTHIC analyses have identified the important contribution of a steam leak from the RPV into the drywell in explaining the observed Unit 3 containment pressurization [10]. These more detailed analyses support the assumption of a direct discharge from the RPV into the drywell made in the MAAP5 simulations reported in this paper. Work is presently ongoing to enhance the MAAP5 model to capture in more detail the thermal response of the suppression pool to this type of discharge transient, incorporating a generalized buoyant jet model together with a multi-zonal representation of the water pool. This type of modeling will tend to modify the magnitude of leakage from the RPV into the drywell, but not necessarily the potential for this leakage to have occurred in the first 20 hours of the Unit 3 event.

After the failure of the RCIC system, the containment pressure is influenced by the actuation of torus and drywell sprays, at varying points from the 20-hour mark to the 42-hour mark. This is represented in the MAAP5 simulation. As shown in Figure 12, the modeling of the spray system is not capable of capturing the extent of containment pressure decrease observed just after 20 hours. In the absence of a more detailed suppression pool model, however, it is not possible to capture the effect of the HPCI turbine discharge into the suppression pool on its surface temperature. Any discharge into the pool after the period of RCIC operation and SRV discharge can have a critical effect on the surface temperature of the pool; for example, due to disturbance of a stratification profile by motion of colder fluid from below a stratified layer toward the pool surface (as discussed above related to the Unit 2 suppression pool response).

The containment venting operations that were performed after 42 hours aided in maintaining drywell pressure around or below design. During this period, it is also assumed that discharge from the RPV is partially directed into the drywell, bypassing the suppression pool. This assumption is required to capture the rate of containment pressurization observed beyond 42 hours (i.e., following the onset of core damage).

Between 60 hours and 67 hours, however, the drywell pressure escalated and held at design. It is likely that an attempted venting operation to prevent this pressure increase from occurring failed at this time [5]. In the MAAP5 simulations presented in this paper, it is assumed that drywell head leakage initiated and was capable of maintaining the pressure at about design for this period. The onset of leakage through the drywell head flange over this period is ultimately consistent with the occurrence of flammable conditions in the Unit 3 reactor building by 67 hours, which is the time of energetic combustion in the reactor building. This assumption of leakage from a drywell head flange around design pressure is further supported by the simulated upper drywell temperatures shown in Figure 12. The simulation results indicate temperatures approaching 700°F, well in excess of the 500°F at which silicone rubber seals begin to degrade in a steam environment.

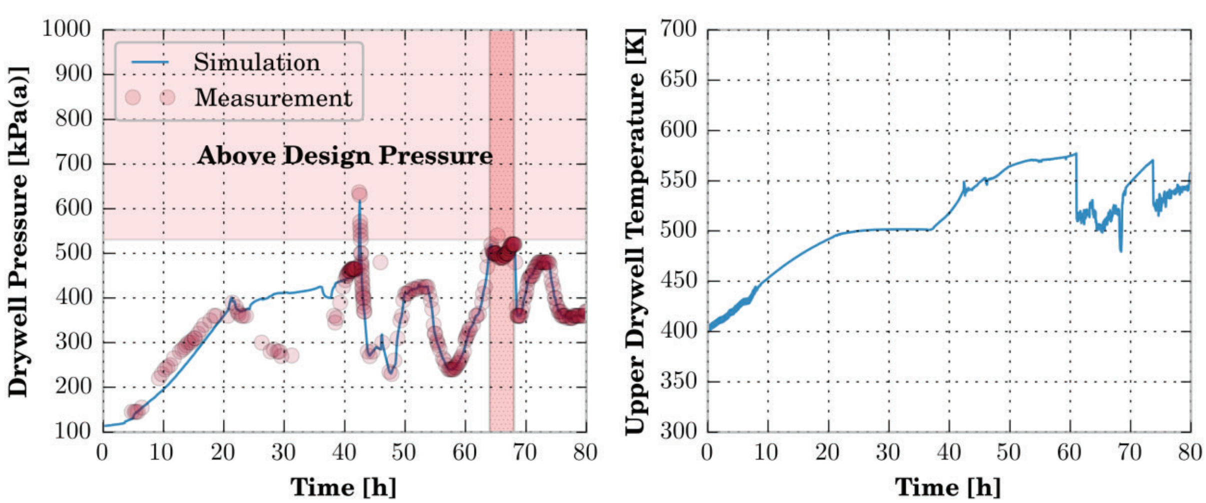

Figure 12. MAAP5 Unit 3 Simulation of Drywell Response

It is important to note that the simulated containment pressure transient shown in Figure 12 can be captured by either in-vessel or ex-vessel core damage states. This was discussed in more detail as part of the first phase of the EPRI Fukushima Technical Evaluation [1].

5.3 Flammable Gas Build-up in Reactor Building

The build-up of flammable gases in the Unit 3 reactor building is influenced by a number of factors not relevant to Units 1 and 2. These reflect the overall uncertainty in the accident progression at Unit 3. Uncertainties arise because of the over 2-day period after 42 hours during which the core melt would not have been stabilized. The core status (i.e., the potential for RPV breach) can have a significant impact on the magnitude of hydrogen generated—MAAP5 simulations tend to indicate that continued hydrogen generation is limited once core melting compacts the debris (and reduces the exposed surface area to participate in oxidation). Furthermore, leakage points from the Unit 3 containment will influence the distribution of flammable gases throughout the reactor building. In addition, the nature of leakage from the containment influences the potential for combustible mixtures of flammable gas to build-up inside the reactor building.

The comparison of in-vessel and ex-vessel scenarios provides one possible analytical clue into the extent of core damage at Unit 3. These MAAP5 simulations indicate that some degree of CCI may have been required to generate sufficient amounts of hydrogen approximately one day after the onset of core damage (around 40 hours) to support the development of flammable conditions inside the reactor building.

6. CONCLUSIONS

The MAAP5 simulation code has been successfully exercised to analyze the events at Fukushima Daiichi Units 1, 2, and 3. The agreement obtained between the simulations and the available plant data provides a level of confidence that the challenges to the plant resulting from a potential severe core damage event are adequately understood and represented in the computational tools. Given this validation of the tools, insights gained from this and similar analyses can be used to further enhance the safety of nuclear power reactors and provide valuable input into the consideration of hardware and procedural changes to the plants. Furthermore, simulations like those documented in this paper can help to prioritize future research and development activities.

The results of the analyses presented in this paper indicate that plant personnel were able to eventually stabilize conditions in the Fukushima Daiichi reactors with varying degrees of damage across the affected units. In particular, it is highly likely that a large fraction of the Unit 1 core ultimately relocated out of the RPV and into containment. The exact status of the Unit 2 core is less certain; however, the simulations presented in this paper indicate that there is a higher potential for a reasonable fraction of the Unit 2 core to have been retained in the RPV. These simulations indicate that some fraction of the core likely slumped into the lower plenum; however, further relocation of debris into containment over the long-term cannot be precluded. By contrast, the status of the Unit 3 core is much less certain. The MAAP5 results presented in this paper indicate that uncertainties in the event “boundary conditions” (e.g., the operation of HPCI at low RPV pressure) make it difficult to conclusively identify either in-vessel or ex-vessel core damage conditions. Simulation of the build-up of flammable gases in the Unit 3 reactor building, however, provides one clue that CCI likely occurred at Unit 3—this scenario ensures that flammable gases are being generated at the same time that drywell head leakage into the reactor building is occurring.

Despite the ability to simulate the Fukushima Daiichi event progressions, there are still areas where epistemic uncertainty prevails, preventing refined estimates of core damage status. As noted in the recent MAAP-MELCOR Crosswalk study [11], key differences exist in the modeling of in-vessel core melt progression. These differences can affect the simulation of in-vessel hydrogen generation. Such differences are thus relevant to a key conclusion regarding core damage status presented in this paper. In this manner, information gained from Fukushima Daiichi decommissioning can provide critical input to reducing epistemic uncertainties in computer models of in-vessel core melt progression.

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