EPRI Fukushima Technical Evaluation—Evaluation of Flammable Gas Leakage from Fukushima Daiichi Containments using the MAAP5 Computer Code

David L. Luxat[a], Donald A. Dube[a], Andrew S. Dercher[a], Richard Wachowiak[b], Rosa Yang[b] and Jeff R. Gabor[a]
[a] ERIN Engineering and Research, Inc., West Chester, PA, 19380, United States of America
[b] Electric Power Research Institute, Palo Alto, CA, United States of America

Abstract: This paper presents initial results from the investigations of flammable gas transport from the Units 1, 2 and 3 containments into their respective reactor buildings. This study is being conducted as part of the Phase 2 effort of the EPRI Fukushima Technical Evaluation, which is an extension of Phase 1 evaluation (Reference [3]). It builds upon the existing event evaluations conducted by TEPCO (References [1] and [2]) and Sandia (Reference [4]). The analyses are conducted using EPRI's Modular Accident Analysis Program (MAAP), version 5.01. The analyses identify the potential for high temperature conditions in the drywell head region of Units 2 and 3 to contribute to the onset of leakage from each drywell—at drywell pressures below twice design. It is not likely that high temperatures in the drywell head region developed at Unit 1 prior to the onset of leakage from the drywell head flange (at about twice design pressure). The leakage at all units through the drywell head flange has been found to enhance the build-up of flammable gases on the refuel floor. Unit 1 may have experienced flammable conditions on its refuel floor for 10 hours prior to the combustion event. Unit 2 likely did not develop flammable conditions on its refuel floor due to the open blowout panel. At Unit 3, leakage from the hard pipe vent into the Standby Gas Treatment System soft ducting may have allowed hydrogen to build-up at lower elevations—this could have contributed to more damage to the reactor building structure.

Keywords: PRA, thermal-hydraulics, MAAP, Fukushima Daiichi.

1. INTRODUCTION

The combustion of hydrogen and carbon monoxide in the Fukushima Daiichi Units 1, 3 and 4 reactor buildings has become a key signature of the event—both from the perspective of public awareness and the overall course of accident management and remediation. Initial work to enhance safety and accident management procedures/guidelines following the event has identified a number of key actions to be performed. However, the detailed technical basis for specific implementations of actions, such as ventilating a reactor building, is still under development. Insights related to the conditions that gave rise to the different combustion events at Fukushima Daiichi thus serve an important role in establishing the technical basis for on-going and future safety enhancements.

The thermal-hydraulic conditions underlying the transport of hydrogen from the Reactor Pressure Vessel, to the wetwell and drywell, and ultimately to the reactor building and environment, are key to the full understanding of the course of the accident at Fukushima Daiichi. The MAAP5 computer code, used for evaluation of integral plant response, is used in this study to:

a) Analyze the containment response at Units 1, 2 and 3;
b) Identify the thermal-hydraulic conditions and types of challenges to containment integrity which may have played a role in the onset of flammable gas leakage into the reactor building; and
c) The transport of flammable gases in the respective reactor buildings.

2. FLAMMABLE GAS LEAKAGE PATHWAYS FROM MARK I CONTAINMENTS

This section describes the assessment of potential hydrogen leakage pathways from containment to the reactor building and environment that was performed as part of this overall effort. The assessment identified the containment failure location, failure mode, and applicable nuclear units at Fukushima Daiichi. The likelihood of the failure mode is categorized in terms of HIGH, MEDIUM AND LOW based on whether the failure mode and associated phenomena are consistent with observed accident behavior at Fukushima Daiichi, and supported by most analyses and separate effects tests. Likewise, the consequences of the failure mode are categorized based on the potential magnitude of the release pathway (e.g., design leakage (LOW consequence) up to 100 volume % per day or greater (HIGH)).

Drywell head flange leakage is the failure location given most attention in the scientific literature. While overpressure failure via straining of the flange bolts is a predominant failure mode as discussed in a number of studies, this failure mechanism alone cannot explain all of the phenomena. A combination of high pressure, high drywell atmospheric humidity and high temperature conditions leading to elastomer seal degradation best explains the leakage phenomena.

References [7] and [8] indicate that the capacity of head flange against leakage for the Mark I containment is above 0.9 MPa (gauge) (130 psig) at normal temperature, well above observed peak drywell pressures at the three units. Either the pre-load of the bolts was particularly low at Fukushima Daiichi, or some other failure mechanism was occurring. Hence the assignment of LOW to MEDIUM likelihood is made for overpressure alone as the containment failure mechanism. Potential leakage rates via the head flange above 100 volume %/day are possible (HIGH consequence).

The NISA report [8] states that a combination of high pressure and high temperature in the drywell may be the cause of leakage via the head flange. A positive feedback mechanism might explain the behavior. The flange bolts elastically stretch upon high pressure and temperature. Tests [8] indicate that some amount of high temperature gas flow in excess of 350°C (660°F) results in degradation of the elastomer gaskets. This in turn causes greater distortion and opening, resulting in still higher flow rates. The containment system is then in a self-relieving mode, sustaining just enough leakage at a given pressure to equal the rate of gas generation (water vapor, non-condensable gases) and pressure rise with increasing drywell gas temperature. This mechanism may well explain phenomena at several of the units where temperatures in the upper head region in the 600°F range are calculated (see Section 4 below).

Flange distortion caused by a combination of high temperature and pressure that causes a permanent opening in the head flange could be another failure mechanism unlike the self-relieving mode discussed above. Several tenths of a mm opening around the circumference of the flange due to deformation would equate to 10^{-3} to as much as 10^{-2} m^2 opening, the latter equating to thousands of volume %/day leakage (HIGH consequence). This could partially explain the sudden containment depressurization around 90 hours at Unit 2, although it is not the only possible explanation. Therefore, the likelihood that this was the failure mode at Unit 2 is at best a MEDIUM.

Failure of the equipment hatch is next considered. Failure due to overpressure alone is unlikely based on analysis in Reference [7] indicating the capacity to be well above 1 MPa (gauge) (150 psig). A more likely failure mode would be failure of the gaskets around the hatch due to high temperatures. NUREG/CR-4944 [9] and NUREG/CR-5096 [10] indicate from tests that gaskets start to fail as low as 238°C for neoprene, 299°C for ethylene propylene, and 370°C for silicone rubber. Some or all of these temperatures are believed to have been exceeded at each of the units. However, the actual material used at Fukushima Daiichi is unknown at this point, so more definitive assessment is not possible. The likelihood that this leakage pathway existed at one or more units is MEDIUM to HIGH. Based on the potential leakage area, the consequence is MEDIUM to HIGH. However, the release location would be into the lower floors of the drywell, resulting in different hydrogen concentration profiles than head flange leakage.

The personnel airlock is next considered. No design information for Fukushima Daiichi is available as of this writing. NUREG/CR-5118 [11] and SAND90-0119 [12] describe testing which indicate no significant leakage below 427°C with pressure up to 2.07 MPa (300 psig). Above 454°C the inner gasket was degraded and the inner door effectively bypassed. Because of the thermal inertia, the outer door seal remained intact. Thus the leakage path likelihood is LOW to MEDIUM. Based on the potential leakage openings given degraded seals, the consequences are LOW to MEDIUM as well.

For electrical penetration assemblies, NUREG/CR-5334 [13] tests indicate no failures up to pressures of 0.51 to 1.07 MPa (75 to 155 psia) and temperatures from 180 to 370°C (360ºF to 700ºF) for up to 10 days. The outer seal did not experience harsh conditions during the tests. Because the temperatures at some of the Daiichi units likely reached or exceeded the maximum test temperature of 600°F, it is difficult to extrapolate the test results. The likelihood that leakage via this pathway was experienced is judged as LOW to MEDIUM. From test data [8] and given the possibility of multiple EPA failures, the consequence in terms of leakage area and rate is MEDIUM.

Much attention has also been given to the vent bellows connecting the wetwell to drywell. With regard to failure by overpressure, the NUREG/CR-6154 [14] series of tests showed that bellows can be stretched to two to three times their undeformed length without leaking. Reference [7] indicates low probability of leakage below 1.2 MPa (gauge) (175 psig). Hence, this failure mode is assigned LOW likelihood. With regard to consequence, Reference [7] describes experiments which confirm that bellows steel is extremely tough and resistant to unstable crack propagation. A bellows tear results only in a “leak”, and that “rupture” will not occur in the bellows. A LOW to MEDIUM consequence is assessed. However, the tests cited in NUREG/CR-6154 did not consider the effects of severe accident elevated temperatures substantially beyond the test condition of about 218 °C (425 °F). This temperature was clearly exceeded at the three Daiichi units. Thus, the potential for a combination of high temperature and high pressure to have caused deformation of the bellows is assessed as MEDIUM.

Two failure pathways that could have caused containment bypass are melt-through and ejection of molten debris at the bottom of the RPV into the transverse instrument probe tube(s), and melt-through in the reactor cavity sump into the recirculation cooling water (RCW) piping. Based on high dose rates in areas of the reactor building of Unit 1 with RCW components, a HIGH likelihood is assessed for Unit 1 and LOW for Units 2 and 3. While some amount of hydrogen could have been ejected along with the corium debris, based on the restricted openings and low elevation for release in the drywell, LOW to MEDIUM consequence has been assessed.

Finally, based on a series of structural analyses documented in Reference [7], the likelihood of failure by overpressure has been assessed as LOW for the following locations:

- Wetwell access hatch
- Global drywell region
- Global wetwell region
- Other penetrations including steam lines, High Pressure Coolant Injection test line, and containment spray.

The capacities of these potential leakage pathways are typically above 1 MPa (gauge) (150 psig). Depending on the size of the breach, the consequences span the entire range from LOW to HIGH.

For Unit 3, there is a MEDIUM to HIGH likelihood that leakage of flammable gases from the hard pipe vent into the Standby Gas Treatment System (SGTS) soft ducting may have occurred, and is supported by elevated dose rates on the SGTS filters. The consequences are also assessed as MEDIUM to HIGH.

3. MAAP5 REPRESENTATION OF FUKUSHIMA DAIICHI CONTAINMENTS AND REACTOR BUILDINGS

This section describes the MAAP5 models developed to represent the Fukushima Daiichi containments and reactor buildings. The 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.

3.1. Enhanced Representation of Mark I Containment

The model of the containment and reactor building follows a common structure for all three units modelled. These models are enhancements to the current representation of a Mark I containment provided as part of sample user guidance (e.g., the sample model for a plant similar to Peach Bottom). The standard model of a Mark I containment which forms the basis for the user guidance represents the containment in terms of four distinct volumes:

- Pedestal representing the region underneath the RPV
- Drywell representing the volume of containment excluding the wetwell downcomers and the pedestal
- Wetwell downcomers
- Wetwell.

As part of the Phase 2 analysis, enhancements to the containment model have been made to represent additional aspects of plant response in order to assess their impact on the event progression. These enhancements facilitate assessment of the variation of two safety-significant parameters:

- Temperature over the drywell
- Distribution of hydrogen gas within the drywell.

This consists of a refined nodalization of the drywell to explicitly represent the sub-volumes:

- Pedestal representing the cavity region underneath the RPV
- Lower drywell for the volume inside the drywell sphere region
- Drywell cylinder volume in which the atmosphere between the drywell wall and RPV cylinder is assumed to be well-mixed due to circulation flows around the bioshield
- Drywell head volume separated from the drywell cylinder by the refuel seal (flows between the two volumes are through a limited number of openings).

Sub-volumes representing the wetwell downcomers and the torus are the same as in the containment model used in the Phase 1 analysis.

The drywell temperature profile is necessary to identify the potential for thermal challenges to containment penetration elastomeric sealing materials. Depending on the stage of core melt progression, it is possible to have different magnitudes of thermal loading of the drywell sphere, cylinder and head regions. Since the natural circulation flow paths through the drywell do not promote strong heat transport flows, it is possible to develop relatively high temperatures in the drywell head region. The elastomeric drywell head flange seal is thus at potential risk of experiencing high temperatures and undergoing thermally induced degradation.

Understanding the thermal challenge to the integrity of containment penetrations is of interest to assess potential reactor building combustion profiles. The location of leakage from containment can have an effect on the distribution of flammable gases inside the reactor building. This in turn has an important effect on the regions of the reactor building in which sufficient hydrogen can build-up to combust.

In addition to the point of leakage, the distribution of hydrogen within the drywell could affect the magnitude of hydrogen available to leak through points of containment impairment. One possibility addressed by this enhanced drywell model is the potential for hydrogen to stratify in drywell head region. A higher concentration of hydrogen in this region can have the following effects on plant response:

- Enhancement of material degradation at the top of the drywell head
- It is not likely that significant hydrogen embrittlement of metal structures could occur over the time frames of interest
    - However, the presence of hydrogen can enhance the degradation of the elastomeric drywell head flange seal when exposed to a high humidity environment
    - An increase in the rate at which hydrogen leaks out through an impaired drywell head flange seal on to the refuel floor.

3.2. Nodalization of Mark I Reactor Building

The standard model of a Mark I reactor building is simpler, consisting of two volumes:

- Torus room
- Remainder of the reactor building.

Some plant models incorporate additional volumes to represent the Standby Gas Treatment System and a stack. These additional volumes were in the Unit 1, 2 and 3 MAAP5 plant models used in the Phase 1 study [3].

The enhanced reactor building model incorporates the following layout:

- Torus room
- Remainder of reactor building basement
- First floor
- Second floor
- Third floor
- Fourth floor
- Lower half of refuel floor
- Upper half of refuel floor
- Refuel cavity (the region above the drywell head but separated from the refuel floor by concrete shield plugs)
- Standby Gas Treatment System
- Stack.

Of primary interest from the perspective of flammable gas combustion inside the reactor building is the distribution of hydrogen through the height of the refuel floor. For all three affected units, there appears to be a high likelihood of enhanced containment leakage developing through the drywell head flange. This leakage path would have displaced hydrogen (and carbon monoxide) directly on to the refuel floor. Thus, the concentration of flammable gases throughout the refuel floor is a critical accident parameter.

However, there is some indication that bypass flows could have developed at Unit 3 directing hydrogen (and carbon monoxide) from the hard pipe vent into the soft ducting of the building ventilation system. This could have displaced flammable gases on to the refuel floor. It also would have displaced flammable gases on to the fourth floor of the reactor building. This is depicted in Figure **1**. This enhanced layout of the reactor building is used below to assess the nature of combustion events for different enhanced leakage scenarios.

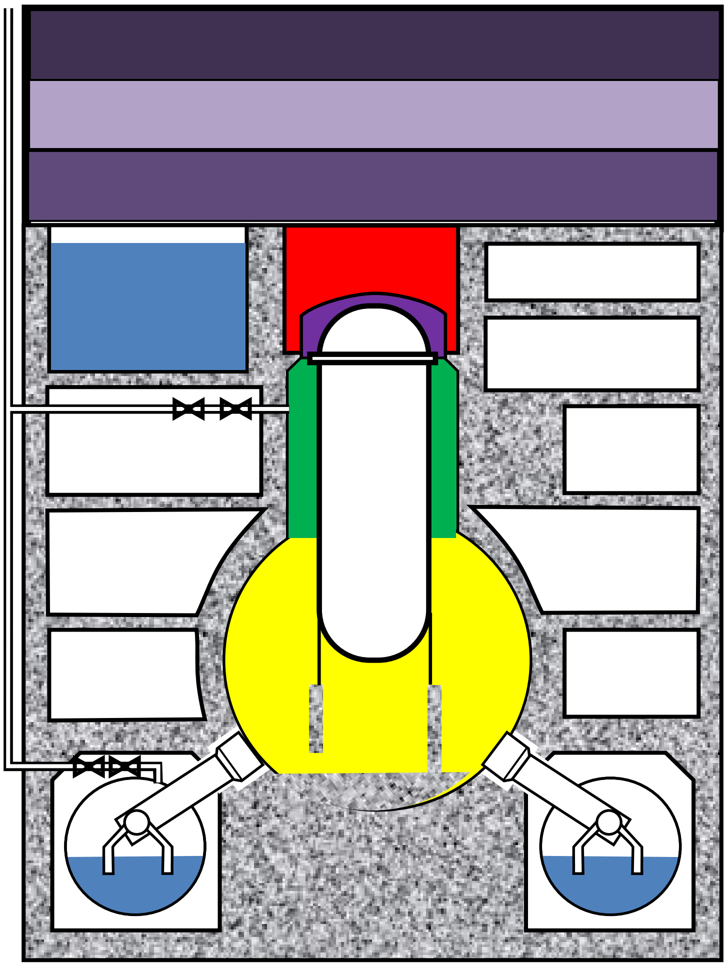

Figure 1: Enhancements to MAAP5 Model of Fukushima Daiichi Containment and Reactor Building

4. MAAP5 SIMULATION OF CONTAINMENT RESPONSE

The release of flammable gases into the Units 1, 2 and 3 reactor buildings depends on the:

- Core melt progression and its influence on total amount of flammable gases generated as well as the rate of generation
- Containment response following the onset of core melting.

This section describes the MAAP5 analyses of the core melt progression and containment response. The analyses are used to identify an estimate of the transient discharge of flammable gases into the Unit 1, 2 and 3 reactor buildings.

4.1. Unit 1 Containment Response

Reference [3] has investigated the alternate Unit 1 accident progression scenarios and compared each against the observed:

- RPV pressure
- Drywell pressure
- Site boundary dose rates.

The following accident progression characteristics are potentially more representative of the Unit 1 event:

- No core cooling with installed systems following the Isolation Condenser being isolated just prior to the arrival of the tsunami
- A steam leak from the RPV into the drywell after T+5 hours, sufficient to depressurize the RPV
- Drywell head impairment around T+12 to T+13 hours, inducing a small leak in the drywell (greater than about 2 square inches)
- RPV lower head breach around T+12 to T+13 hours
- Low water injection rates into the RPV beginning at about T+15 hours (significantly less than 10 gpm)
- Wetwell venting around T+23.8 hours for about 30 minutes.

This type of accident progression is similar to PRA core damage sequences progressing from a Station Blackout (SBO). However, the operation of the Unit 1 Isolation Condenser for about 1 hour after the earthquake delayed the progression of the event (due to the lower decay heat at the time core cooling was lost).

The overall containment pressurization is well-represented based on these event scenario assumptions. The simulation of the drywell pressure transient, compared against observed drywell pressure, is shown in Figure **2**.

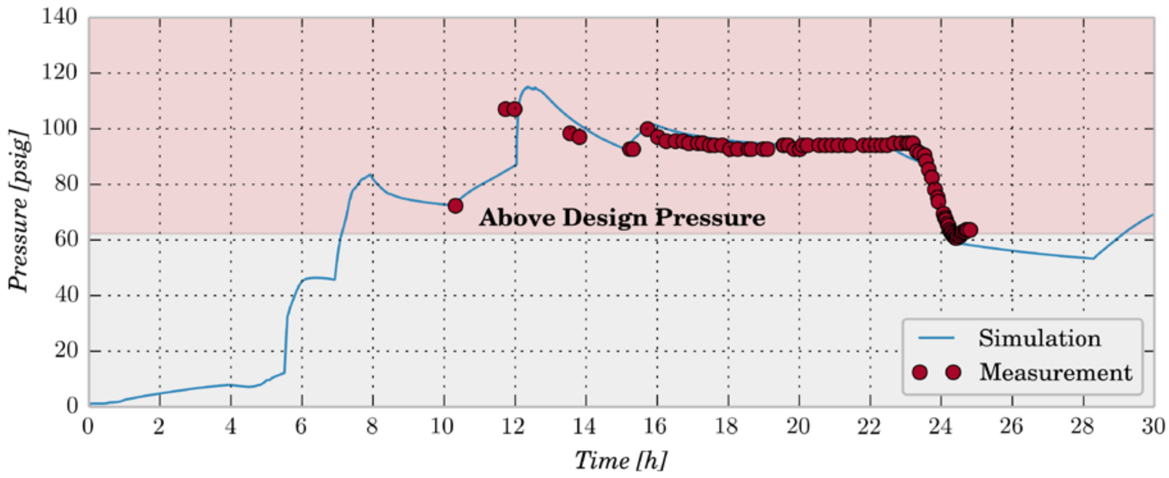

Figure 2: Simulated Drywell Pressure for Unit 1

The impairment of the drywell head is assumed to be due to the rapid rise in containment pressure to nearly 120 psig (around twice design pressure). The drywell head would have lifted against the restraint of the flange bolts. The drywell head seal likely would have degraded over a finite time following drywell head lifting, due to exposure to a steam environment. There may have been a period of nearly an hour before leakage from the drywell head commenced due to the difference between the first observation of pressures around twice design (between T+11 and T+12 hours) and the rise in site boundary dose rates (around T+13 hours).

The temperature in the drywell head region would likely have been relatively low at this time of significant drywell head lifting. Figure **3** shows the simulated distribution of temperatures in the Unit 1 drywell. The relatively short time at which RPV lower head breach is simulated to occur is the primary reason the temperatures in the drywell head region do not exceed 500°F in the simulation. The relocation of core debris into the pedestal around T+12 hours transfers the majority of the heat source to the bottom of the drywell, preventing continued heat losses from the RPV into the drywell cylinder and head regions. Thus, thermal degradation of the drywell head seal would likely not have occurred prior to significant lifting of the drywell head. Leakage from the Unit 1 drywell head flange was thus most likely due to the significant drywell overpressure experienced.

The core melt progression identified through MAAP5 simulations (see, for example, Reference [3]) would have resulted in early generation of hydrogen. Hydrogen generation could have begun around 20 hours prior to the occurrence of the energetic combustion event. Figure **4** shows the in-vessel hydrogen transient simulated using MAAP5.

The magnitude of in-vessel hydrogen generation is quite large in this simulation. Nearly 750 lbm of hydrogen is generated during the in-vessel phase of the core melt progression, based on the simulation. Slow leakage of this amount of hydrogen on to the refuel floor may have resulted in flammable conditions on the refuel floor. However, leakage out of the refuel floor to the environment would likely have prevented flammable conditions being maintained until T+24 hours (i.e., around the time of energetic combustion in the Unit 1 refuel floor).

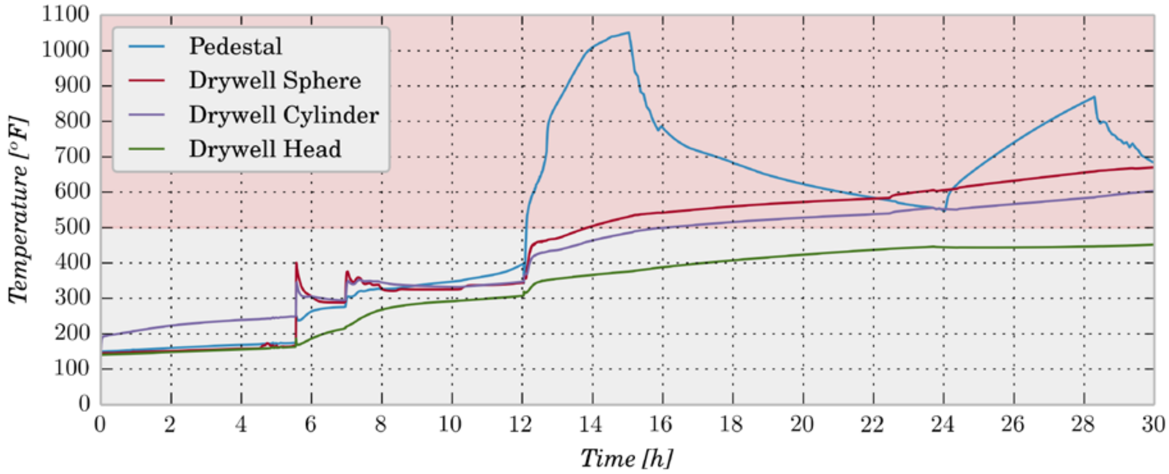

Figure 3: Simulated Drywell Temperature Distribution for Unit 1

The breach of the RPV at about T+12 hours in the MAAP5 simulation initiates core-concrete attack. This process results in the generation of a large amount of hydrogen (and some carbon monoxide*) as indicated in Figure **5**. The rate of generation is relatively slow. During the early phase of core-concrete interaction (CCI), hydrogen is typically generated at a rate of 1 to 2 kg/s (due to the oxidation of the remaining Zr in the core debris) [6]. In the late phase of CCI, hydrogen is generated by Fe oxidation in the core-concrete debris—this typically occurs at a rate of about 4 g/s [6].

This prolonged generation of hydrogen would have maintained a relatively constant leakage, in the long term, of hydrogen on to the refuel floor following drywell head lifting. Higher rates of water injection to the RPV may have been able to quench the core debris on the concrete floor, and terminate CCI. This could have arrested the long-term leakage of hydrogen on to the refuel floor.

* Basaltic concrete is assumed for these simulations based on the type of concrete used for the Fukushima Daiichi units. This concrete type generates significantly lower amounts of carbon monoxide (and carbon dioxide) relative to limestone/common sand concrete.

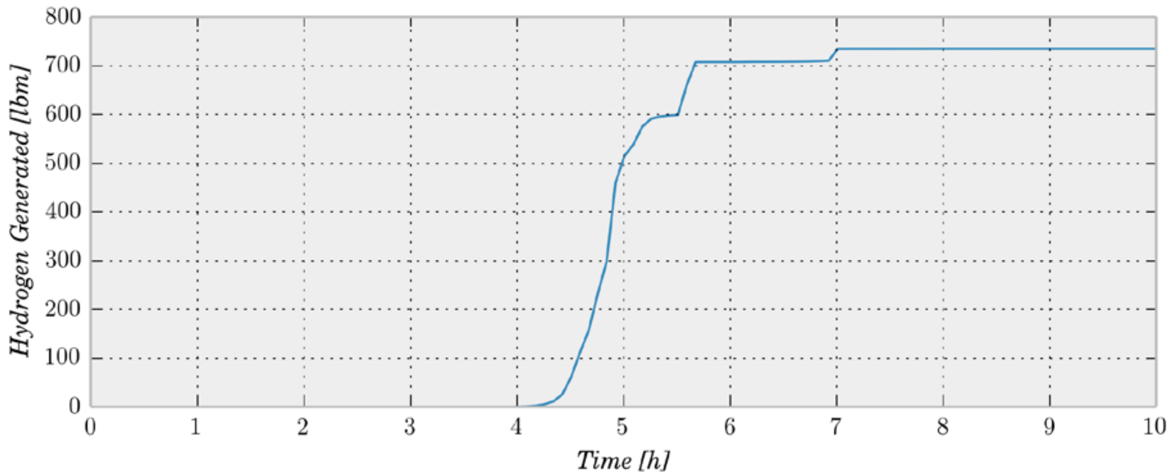

Figure 4: Simulation of Unit 1 In-Vessel Hydrogen Generation

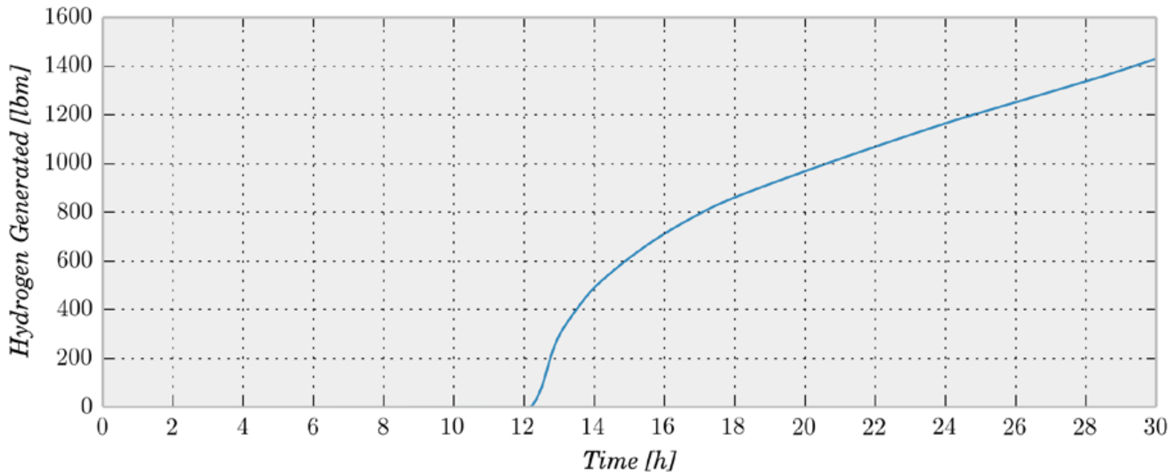

Figure 5: Simulation of Unit 1 Ex-Vessel Hydrogen Generation (Degraded RPV Water Injection Scenario)

4.2. Unit 2 Containment Response

Unit 2 accident progression was investigated in Reference [3] following the same approach adopted for Unit 1. Additional analyses of the Unit 2 accident progression have been performed as part of this Phase 2 effort. These analyses have focused on understanding in greater detail the accident progression after RCIC water injection failed.

The characteristics of the event are as follows:

- Core cooling was likely maintained until about T+67 hours due to operation of RCIC in an unattended mode
- After the loss of RCIC at this time, core cooling was not restored until after T+75 hours and following depressurization of the RPV by deliberate opening of an SRV
- The RPV partially re-pressurized 3 times between T+75 hours and T+85 hours most likely due to SRVs re-closing (see Figure **6**)
- Between T+80 and T+81 hours, the drywell pressure rose by nearly 40 psig due to a combination of enhanced hydrogen generation and steam generation
- 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 containment pressure.

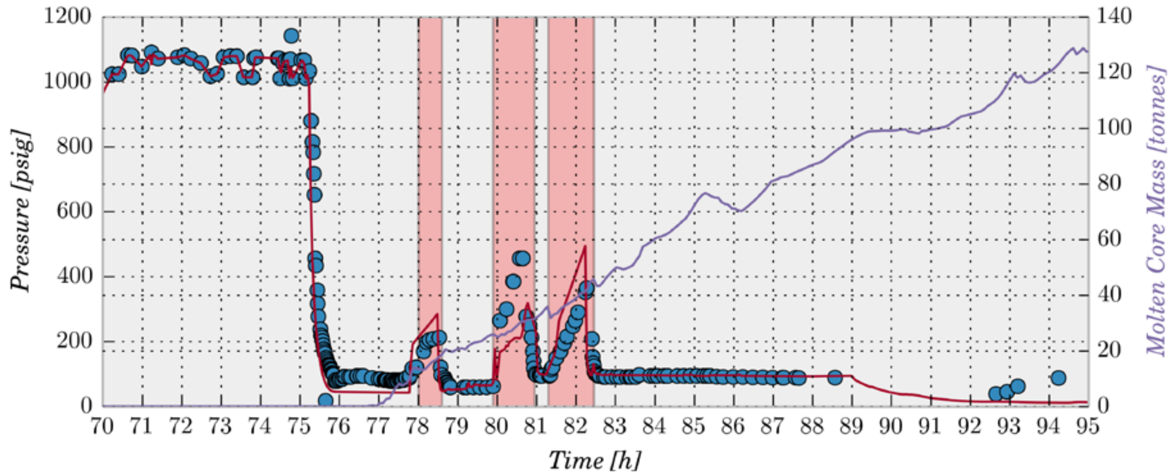

Figure 6: Measured Unit 2 RPV Pressure Transient from T+75 hours to T+85 hours

Cessation of RPV water injection occurred during the periods of RPV partial re-pressurization, shown in Figure **6**. This allowed core heatup to continue, as shown by the continuing melting of core materials. When SRV opening reduced RPV pressure again, RPV water injection restoration would have been to an overheated core. These conditions made hydrogen generation possible. The MAAP5 simulations have indicated sufficient core heatup could have occurred by T+80 hours to promote substantial generation of hydrogen.

The simulated drywell pressure transient is shown in Figure **7**. Superimposed with the drywell pressure transient is the simulated in-vessel hydrogen generation transient. The rapid pressure rise in containment starting around T+80 hours is due to the generation of steam and hydrogen in the RPV. The quenching of overheated core material causes this.

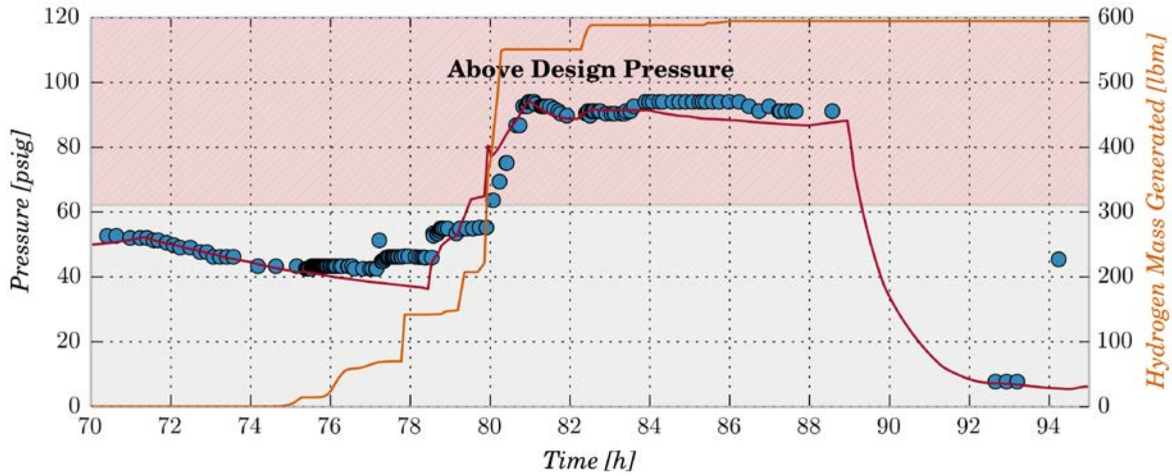

Figure 7: Simulated Drywell Pressure Response for Unit 2

The rise in containment pressure around T+80 hours, however, also indicates the potential for a direct steam leak developing from the RPV into the containment. The rise in RPV pressure around this time (see Figure **6**) coincided with SRV closing. To capture both a rise in RPV pressure and a very sharp jump in containment pressure, MAAP5 simulations have indicated the potential for a steam leak through, for example, a failed in-core instrument tube.

From Figure **7**, the drywell pressure remains relatively constant between T+80 hours and T+89 hours. This is similar to the trend observed between T+15 hours and T+24 hours at Unit 1 (see Figure **2**), which was likely governed by leakage through the drywell head flange. Drywell head leakage at Unit

2 is highly probably based on the observation of high dose rates—of about 100 rem/h—on the refuel cavity seal plugs (above the drywell head).

The cause of drywell head leakage, however, may be different from Unit 1 given the long period of Unit 2 RCIC operation without containment cooling. Figure **8** shows the simulated drywell temperature transient for Unit 2. The heat load to the region of the drywell head is relatively constant prior to T+75 hours, when core damage commences. It is governed by thermal radiation from the RPV upper head into this region. Thus, the prolonged period of RCIC operation, without any forced drywell circulation, causes the temperature in the drywell head region to increase toward 500°F prior to core damage.

The onset of core damage, after T+75 hours, results in a more rapid rate of drywell head temperature increase. This heat load is primarily focused in the area of the drywell cylinder and head.[‡] However, the impact of this enhanced heat load is distinct from that simulated for Unit 1, due to the operation of RCIC until T+67 hours without any forced drywell circulation. By T+78 hours, the simulated temperature in the drywell head region is above 500°F, sustained at these levels for nearly 10 hours. The potential for thermal degradation of the drywell head flange seal is thus possible—by this point drywell head lifting would have started due to containment overpressure (see Figure **7**) and exposed the seal material to a steam environment.

However, the simulation of core melt progression indicates that hydrogen generation from the damaged core may not have occurred over a prolonged period of time. A large amount of hydrogen may have been generated around T+80 hours (about 600 lbm). In the absence of RPV lower head breach, and the occurrence of CCI, current models of in-vessel core melt progression indicate limited hydrogen generation beyond the initial core melting/candling phase.

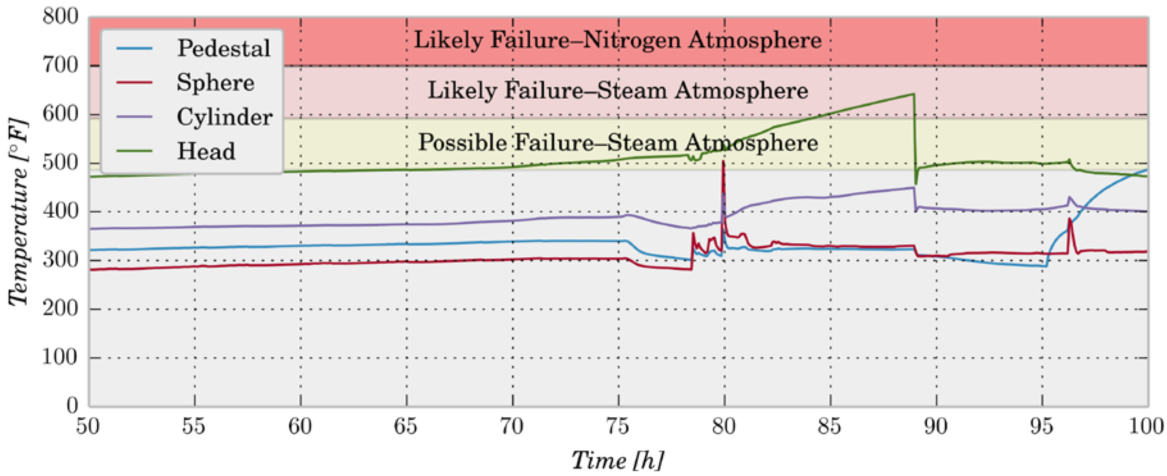

Figure 8: Simulated Drywell Temperature Distribution for Unit 2

4.3. Unit 3 Containment Response

Unit 3 accident progression was investigated in Reference [3] following the same approach adopted for Units 1 and 2. The Phase 2 efforts have focused on evaluating characteristics of accident progression with respect to their potential for leading to a flammable atmosphere in the Unit 3 reactor building.

The key characteristics of the event are as follows:

- Core cooling was maintained after the tsunami until about T+20 hours with the RCIC system

[‡] The temperatures in the lower portion of the drywell are somewhat ameliorated by the assumption of heat dissipation from torus water into the torus.

- From T+21 hours until T+36 hours, operators used the HPCI system in an attempt to maintain water level
- Operation of the HPCI system at low RPV pressure after about T+28 hours likely resulted in a reduction in water injection rate and an inability to maintain water level—by T+36 hours, it is likely that RPV water level had reached TAF
- From T+36 hours, RPV pressure was controlled by SRV cycling and no water was injected due to the high pressure
- After T+42 hours, water injection to the RPV was possible due to the unintentional depressurization of the RPV (it is possible that RPV depressurization occurred due to ADS being triggered [1], [2])
- Fire engine injection was degraded due to flow bypass through the condensate transfer pump (see, for example, Reference [2])—assumed to be at a rate around that required to remove decay heat via boiling of the injection flow
- There is a potential that leakage occurred through the drywell head flange seal starting around T+60 hours—this leakage could have been complemented by some leakage of vent gases into the soft ductwork and discharge on to the fourth and fifth floors of the reactor building.

The drywell pressure response is shown in Figure **9**. The venting of containment after T+42 hours aided in maintaining drywell pressure around or below design. However, between T+60 and T+67 hours (the time of energetic combustion in the reactor building), the drywell pressure escalated and held at design.

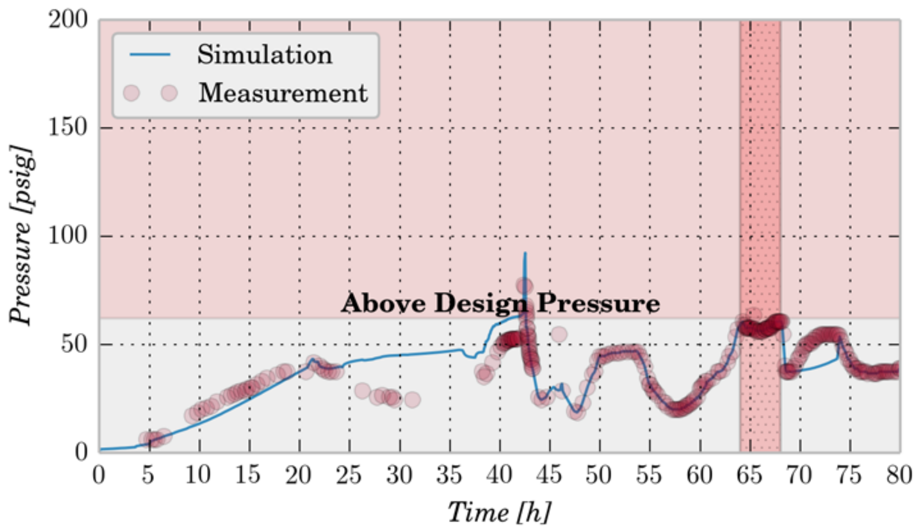

Figure 9: Simulated Drywell Pressure Response for Unit 3

Similar to Unit 2, the heat dissipation from the RPV into the drywell would have been prolonged. Wetwell sprays were used between T+21 hours and T+36 hours to control drywell pressure. For a period of a few hours prior to T+36 hours, drywell sprays were used as well. However, drywell sprays primarily affect the drywell sphere, having minimal effect on temperatures in the cylinder and head regions.

Figure **10** shows the simulated drywell temperature transient for Unit 3. These results for the multi-node MAAP5 drywell have been benchmarked against measured Unit 3 temperatures in the first 20 hours following the earthquake. The simulation beyond T+42 hours is thus judged to provide a reasonable representation of the drywell thermal response. The temperature in the drywell head region exceeds 500°F beyond about T+42 hours, driven higher due to the onset of core damage after T+36 hours. By T+60 hours, the drywell head atmospheric temperature is around 600°F.

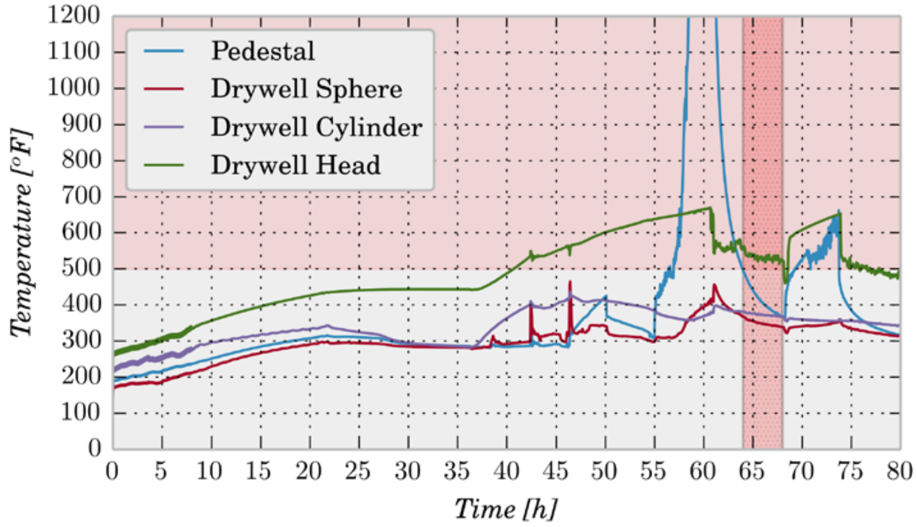

Figure 10: Simulated Drywell Temperature Distribution for Unit 3

There is high likelihood that thermal degradation of the drywell head seal could have occurred by T+60 hours. Leakage from containment is found with MAAP5 simulations to be necessary to explain the drywell pressure holding constant between T+60 and T+67 hours. Thus, leakage from the Unit 3 containment via the drywell head flange could have commenced by T+60 hours.

The MAAP5 simulation of Unit 3 core melt progression has highlighted the potential for RPV lower head breach around T+60 hours. However, this conclusion is sensitive to small variations in the rate of water injection during HPCI operation at low RPV pressure as well as the rate of RPV water injection by fire engine pumps. The potential for some relocation of core debris into containment at Unit 3 appears to be highly likely from the MAAP5 simulations.

The occurrence of CCI is also likely necessary to explain the development of flammable conditions in the Unit 3 reactor building. This is discussed further below. The hydrogen generation transient for Unit 3 is presented in Figure **11**.

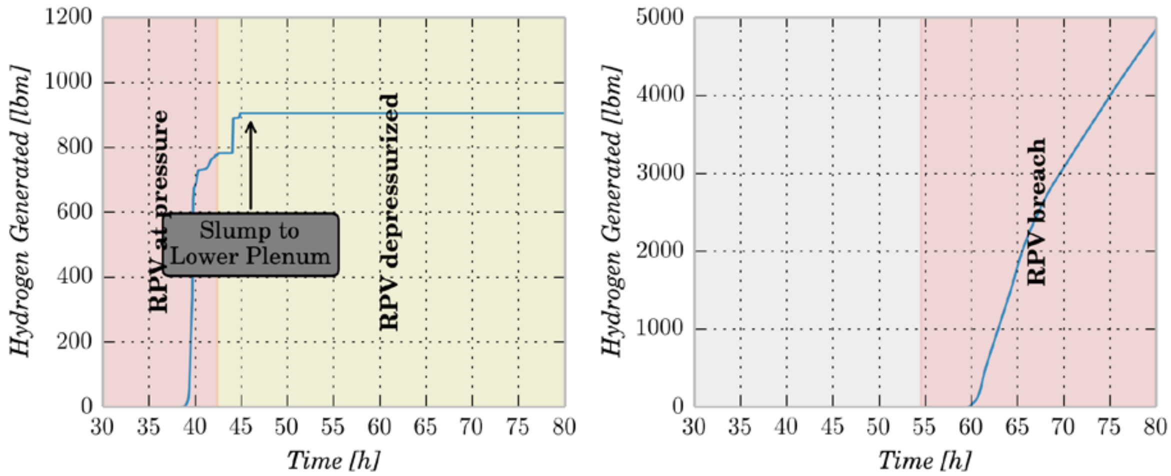

Figure 11: Simulated Unit 3 Hydrogen Generation

5. MAAP5 SIMULATION OF REACTOR BUILDING FLAMMABLE GAS DISTRIBUTION

5.1. Simulation of Unit 1 Flammable Gas Distribution

Figure **12** shows the simulated distribution of hydrogen in the Unit 1 reactor building. 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 (T+24.8 hours), the hydrogen concentration on the refuel floor is about 17%.[§]

An energetic combustion event could have occurred at any point between T+15 and T+24.8 hours. The energetic combustion event, however, may not have occurred until T+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.

There is a slow build-up of hydrogen at lower elevations. This is due to natural convection flow through the building, facilitated by large area openings between each floor and the stairwells. By approximately T+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.[**]

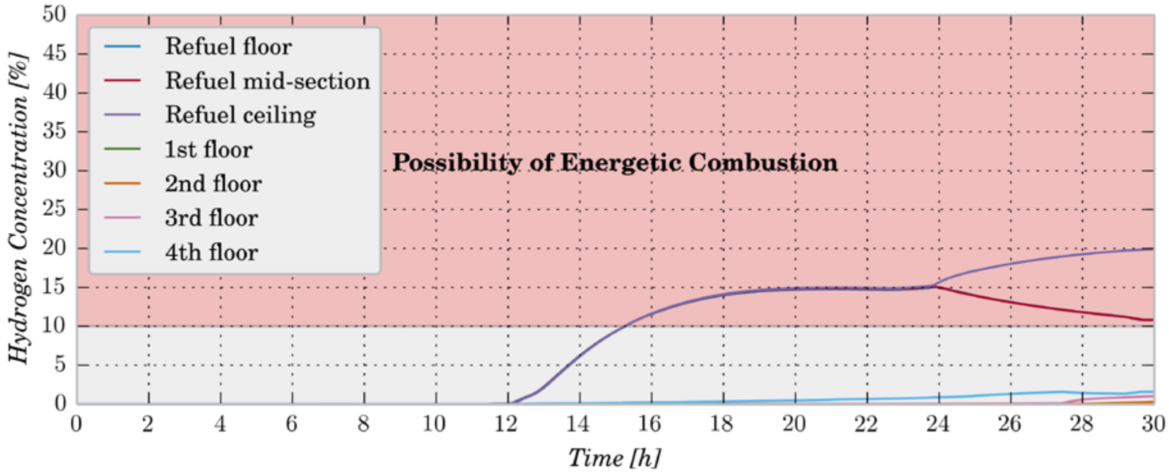

Figure 12: Simulation of Reactor Building Flammable Gas Distribution for Unit 1 (Degraded Water Injection)

5.2. Simulation of Unit 2 Flammable Gas Distribution

The build-up of hydrogen on the refuel floor at Unit 2 would have been significantly different relative to Units 1 and 3. After the energetic combustion of hydrogen occurred on the Unit 1 refuel floor, it is likely that the Unit 2 refuel floor blowout panel was dislodged. The rarefaction phase of the

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§ 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.

** It should be noted that the nature of damage to the Unit 1 structure would also have been influenced by the limited resistance to pressure loading provided by the sheet metal siding at the elevation of the refuel floor. This would have resulted in the combustion pressure wave being vented to atmosphere more readily. By contrast, the Units 3 and 4 refuel floors were part of the concrete super-structure. The refuel floor wall panels for these units would have experienced more over-pressure before yielding and venting the combustion pressure wave to the atmosphere.

compressional wave generated by the combustion event would have resulted in a negative pressure outside the Unit 2 reactor building. The resulting positive differential pressure between the inside and outside surfaces of the Unit 2 blowout panel would have been sufficient to open the blowout panel.

MAAP5 simulations find that the open blowout panel is sufficient to limit the hydrogen concentration to below 4% on the refuel floor. This is despite a relatively large area leakage assumed from the drywell head in order to depressurize containment to atmosphere.

5.3. Simulation of Unit 3 Flammable Gas Distribution

The build-up of flammable gases in the Unit 3 reactor building is influenced by a number of factors not relevant to Unit 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 T+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)
- Leakage points from the Unit 3 containment—there is a high likelihood that drywell head leakage occurred, though the potential for leakage of flammable gases from the hard pipe vent into the SGTS soft ducting is less well understood but supported by elevated dose rates on the SGTS filters.

Figure **13** presents the simulation of reactor building hydrogen distribution. This corresponds to a simulation with partial hard pipe vent bypass (on the order of 10% of the vent flow) commencing around T+55 hours.

Leakage of vent flow into the soft ducting is found to complement hydrogen leakage through the drywell head. This supports flammable conditions developing in the Unit 3 reactor building by T+67 hours. Unlike the Unit 1 energetic combustion event, these results indicate the potential for greater damage to lower elevations, particularly the fourth floor. Flammable gas build-up at this elevation to concentrations supporting energetic combustion may have occurred. The nature of the combustion, with a substantial debris plume directed upward upon combustion, can be partly explained by a significant energy discharge at lower elevations. This would ensure greater damage to the building superstructure.

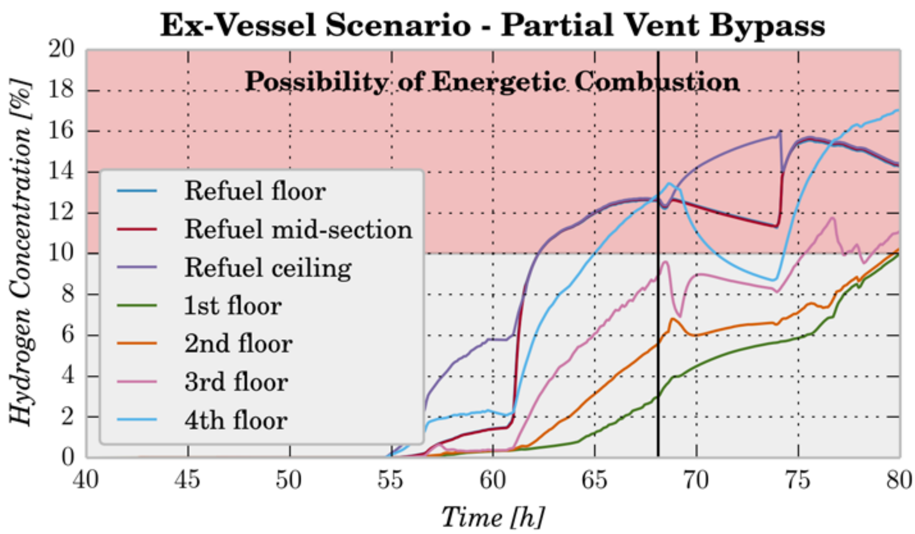

Figure 13: Simulation of Reactor Building Flammable Gas Distribution for Unit 3

6. CONCLUSION

This paper presented initial results from the investigations of flammable gas transport from the Unit 1, 2 and 3 containments into their respective reactor buildings. This study is being conducted as part of the Phase 2 effort of the EPRI Fukushima Technical Evaluation, which is an extension of Phase 1 evaluation (Reference [3]). It builds upon the existing event evaluations conducted by TEPCO (References [1] and [2]) and Sandia National Laboratories (Reference [4]).

The analyses reported in this paper have identified the potential for high temperature conditions in the drywell head region of Units 2 and 3 to contribute to the onset of leakage from each drywell. The drywell pressure at which this leakage would have commenced is around 1.3x to 1.0x design pressure for Units 2 and 3, respectively.

This indicates the potential for a drywell head flange failure mode influenced by degradation of the elastomeric seal at high temperatures. The lifting of the drywell head flange due to high drywell pressures (around design pressure or higher) would have exposed the seal to a high humidity atmosphere. High humidity enhances the degradation of elastomeric seals at elevated temperatures.

The high temperatures developed due to the long time that existed between the loss of active containment cooling and forced circulation and the onset of core damage. Drywell head temperatures at Unit 1, by contrast, were likely much lower when leakage from the drywell head flange commenced. The leakage at Unit 1 is identified through simulations to be caused by very high overpressures developing in the drywell—approximately twice design pressure. These pressures developed very early in the accident (about T+12 hours). At this time, the temperature in the drywell head region was well below temperatures at which elastomeric seal degradation could occur.

Thus, the effect of drywell head temperature may have contributed to the onset of leakage from Units 2 and 3 around, or just above, design pressure.

The leakage at all units through the drywell head flange has been found to enhance the build-up of flammable gases on the refuel floor. Unit 1 may have experienced a flammable refuel floor for a period of 10 hours prior to the actual combustion event. The persistence of flammable conditions likely resulted because of the lack of power, and an ignition source on the refuel floor. Unit 2 most likely did not see flammable conditions developing on the refuel floor because of the effect of the open reactor building refuel floor blowout panel. At Unit 3, flammable conditions developed relatively quickly after leakage from the drywell head may have commenced (at T+60 hours). MAAP5 simulations indicate that leakage from the hard pipe vent to soft ducting may have contributed to the build-up of flammable gases at lower elevations. This would have caused more damage to the Unit 3 reactor building, relative to that observed at Unit 1.

References

[1] "*Fukushima Nuclear Accident Analysis Report*," TEPCO, Tokyo, Japan, (2012).

[2] "*Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and examination into unsolved issues in the accident progression, Progress Report No. 1*," TEPCO, Tokyo, Japan, (2013).

[3] "*Fukushima Technical Evaluation: MAAP5 Phase 1 Analysis*," EPRI, Palo Alto, CA, (2013).

[4] R. Gauntt et al., "*Fukushima Daiichi Accident Study (Status as of April 2012)*," SAND2012-6173, Sandia National Laboratories, Albuquerque, NM, (2012).

[5] "*Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects*," EPRI, Palo Alto, CA, 1025295, (2012).

[6] B.R. Sehgal, ed., "*Nuclear Safety in Light Water Reactors: Severe Accident Phenomenology*," Academic Press, New York, New York, USA, (2012).

[7] B.W. Spencer et al., "*Risk-Informed Assessment of Degraded Containment Vessels*," NUREG/CR-6920, SAND2006-3772P, Sandia National Laboratories, Albuquerque, NM, (2006).

[8] "*Technical Knowledge of the Accident at Fukushima Dai-ichi Nuclear Power Station of Tokyo Electric Power Co., Inc.*," (Provisional Translation), Nuclear and Industrial Safety Agency, (2012).
[9] T. L. Bridges, "*Containment Penetration Elastomer Seal Leak Rate Tests*," NUREG/CR-4944, SAND87-7118, Sandia National Laboratories, Albuquerque, NM, (1987).
[10] D.A. Brinson and G.A. Graves, "*Evaluation of Seals for Mechanical Penetrations of Containment Buildings*," NUREG/CR-5096, SAND88-7016, Sandia National Laboratories, Albuquerque, NM, (1988).
[11] J.T. Julien and S.W. Peters, "*Leak and Structural Test of a Personnel Air Lock for LWR Containments Subjected to Pressure and Temperature Beyond Design Limits*," NUREG/CR-5118, SAND88-7155, Sandia National Laboratories, Albuquerque, NM, (1989).
[12] D.S. Horschel et al., "*Insights into the Behavior of Nuclear Power Plant Containments during Severe Accidents*," SAND90-0119, Sandia National Laboratories, Albuquerque, NM, (1993).
[13] D.B. Clauss, "*Severe Accident Testing of Electrical Penetration Assemblies*," NUREG/CR-5334, SAND89-0327, Sandia National Laboratories, Albuquerque, NM, (1989).
[14] L.D. Lambert and M.B. Parks, "*Experimental Results from Containment Piping Bellows Subjected to Severe Accident Conditions, Volume 2: Results from Bellows Tested in Corroded Conditions*," NUREG/CR-6154, SAND94-1711, Sandia National Laboratories, Albuquerque, NM, (1995).