RELAP5/SCDAPSIM/MOD3.4 Analysis of the Influence of Water Addition on the Behavior of a BWR during a Fukushima-like Severe Accident

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Abstract

Immediately after the accident at Fukushima Daiichi, Innovative Systems Software (ISS), and other members of the international SCDAP Development and Training Program (SDTP) started an assessment of the possible core/vessel damage states of the Fukushima Daiichi Units 1-3. The assessment included a brief review of relevant severe accident experiments and a series of detailed calculations using RELAP/SCDAPSIM. The calculations used a detailed RELAP/SCDAPSIM model of the Laguna Verde BWR vessel and related reactor cooling systems. The Laguna Verde models were provided by the Comision Nacional de Seguridad Nuclear y Salvaguardias, the Mexican Nuclear Regulatory Authority. The initial assessment was originally presented to the International Atomic Energy Agency on March 21, 2011 to support their emergency response team and later to the SDTP Japanese members to support their Fukushima Daiichi specific analysis and model development.

Since the initial calculations were performed and documented in the open literature, a series of related calculations have been performed by ISS and SDTP members. This paper documents the first in a series of assessment calculations performed by the lead author at the Indian Institute of Technology, Kanpur. Specifically, these calculations have looked at the influence of water addition using thermal hydraulic conditions representative of the reactor core isolation cooling (RCIC) system where the water injection is impacted by time and reactor vessel pressure. Although these calculations were extended to the point of likely vessel failure or stable core cooling, this paper focuses on water addition during the initial heating and melting of the core where water addition may be the most effective in limiting the extent of fuel melting. The paper also presents the results of a base case, a station blackout transient without water addition, for comparison purposes. The base case calculations were carried out to 10 hours after reactor scram to a point beyond the point of likely vessel failure.

1. Introduction

As described in [1], immediately after the accident at Fukushima Daiichi, Innovative Systems Software (ISS), and other members of the international SCDAP Development and Training Program (SDTP) [2] started an assessment of the possible core/vessel damage states of the Fukushima Daiichi Units 1-3. The assessment included a brief review of relevant severe accident experiments and a series of detailed calculations using RELAP/SCDAPSIM for a representative BWR vessel and related cooling systems.

Since the initial calculations were performed and documented in the open literature [1], a series of related calculations have been performed by ISS and SDTP members. This paper documents the first in a series of RELAP/SCDAPSIM/MOD3.4(bif) [3-7] calculations performed by the lead author at the Indian Institute of Technology, Kanpur. Specifically, these calculations have looked at the influence of water addition using thermal hydraulic conditions representative of the reactor core isolation cooling (RCIC) system where the water injection is impacted by time and reactor vessel pressure. Although these calculations were extended to the point of likely vessel failure or stable core cooling, this paper focuses on water addition during the initial heating and melting of the core where water addition may be the most effective in limiting the extent of fuel melting. The paper also presents the results of a base case, a station blackout transient without water addition, for comparison purposes. The base case calculations were carried out to 10 hours resulting in {U-Zr}-O_2 melting and relocation and the likely failure of the lower head.

As described briefly in Section 2, the influence of water addition during core heatup and melting has been studied extensively through a wide range of experiments. In particular, two experimental programs, the CORA and Quench programs [8,9], performed at the Karlsruhe Institute of Technology (KIT) in Germany, have explored the influence of water injection for peak cladding temperatures up to 2600 K and thus covered the initial stages of core heat up and melting including the liquefaction and relocation of BWR control blades, structural material, and fuel rod cladding. Two specific experiments in the CORA program, CORA-16 and CORA-17 are particularly germane for these calculations. CORA-16 and CORA-17 used BWR bundles with representative Zircaloy channel boxes and BWR control blade segments. Both experiments used identical thermal hydraulic boundary conditions during bundle heatup and melting. However, the CORA-16 was terminated by a slow cooldown with steam while CORA-17 was terminated by a rapid injection of water.

As described in Section 3, RELAP/SCDAPSIM/MOD3.4(bif) was used to perform the detailed calculations presented in this paper. The RELAP/SCDAPSIM code is designed to predict the behavior of reactor systems during normal and accident conditions including severe accidents up to the point of reactor vessel failure. MOD3.4, the current production version, has been used by member organizations and licensed users to support a variety of applications including the design and analysis of severe accident experiments.

As described in Section 4, the initial Laguna Verde RELAP/SCDAPSIM input models were provided by the Comision Nacional de Seguridad Nuclear y Salvaguardias (CNSNS), the Mexican Nuclear Regulatory Authority. The Laguna Verde BWRs are BWR5 designs with 444 fuel assemblies, a thermal power of 2370 MW, and an average core burnup of 3.57×10^5 MWs/Kg. For comparison, Fukushima Daiichi Unit has a thermal power of 1380 MW from 400 assemblies. Units 2 and 3 have thermal powers of 2381 MW from 548 assemblies. The Laguna Verde BWRs employ the drywell/pressure-suppression features of the BWR/Mark II containments [10,11].

2. Highlights of Relevant Experiments and Phenomena

As described in [1], the existing data base and severe accident code and models developed over the past 40 years since the accident at TMI-2 are considered to be adequate to predict the likely states of the core and vessel of a typical LWR during "Fukushima-Daiichi-like" scenarios. The existing experiments also showed clearly the impact of the addition of water once severe fuel damage has started. For example, at heating rates typical of a station blackout with early loss of cooling, the addition of water resulted in the accelerated oxidation of the Zircaloy cladding (and B_4C when present), significant increases in hydrogen generation, and the accelerated formation of liquefied/molten {U-Zr}-O_2**.** The impact of water addition was strongly dependent on the maximum cladding temperature in the assemblies and the state of the core at the time that the water was added. In general terms, if the water was added when peak assembly temperatures were between 1500 to 2000 K, the shattering or spalling of the protective oxide on the Zircaloy cladding during reflood resulted in accelerated oxidation and heatup of the assembly. If the water was added when peak temperatures were between 2000 to 2600 K, the oxidation and heatup was additionally accelerated by the oxidation of the molten Zircaloy. At temperatures above 2600 K, the influence of enhanced oxidation was diminished due to either the complete consumption of the unoxidized material or the formation of regions of liquefied or molten {U-Zr}-O_2. The dramatic influence of water addition on enhanced oxidation for peak assembly temperatures greater than 1500 K was clearly evident in the Quench 11 experiment performed at KIT [12] as shown in the example presented in Reference [1]. Although Quench 11 used a representative PWR assembly, similar results were observed in experiments with representative BWR assemblies.

Figure 1 shows the results of two such experiments, CORA-16 and CORA-17. Both experiments used identical representative BWR assemblies including a Zircaloy channel box and a B_4C control blade segment and were performed in the CORA facility located at KIT. Both experiments were electrically heated with initial heatup rates controlled by an increase in the electrical power in the assembly as shown in the left graph of Figure 1. A mixture of superheated steam and argon was injected into the bottom of the assembly at a constant rate. The heatup in both experiments was initiated at 50 minutes as the power was increased in a linear fashion. The power was then reduced to zero at ~80 minutes. In CORA-16, the bundle was then slowly cooled by the flow of the mixture of steam and argon. By contrast in CORA-17, after the power was reduced to zero, water was added to the bottom of the assembly. Although the lower, cooler region of the CORA-17 assembly was quickly cooled by the addition of water, the hotter, upper portion of the assembly experienced a rapid increase in temperature. As shown on the right hand side of this figure, this rapid increase in temperature was clearly associated with the sharp increase in oxidation of the upper portion of the assembly (with an associated step increase in the hydrogen generation rate in the assembly). In the figure, the flat top of the hydrogen generation rate for CORA-17 during the rapid spike associated with water addition was due to the upper limit on the hydrogen monitoring instrument.

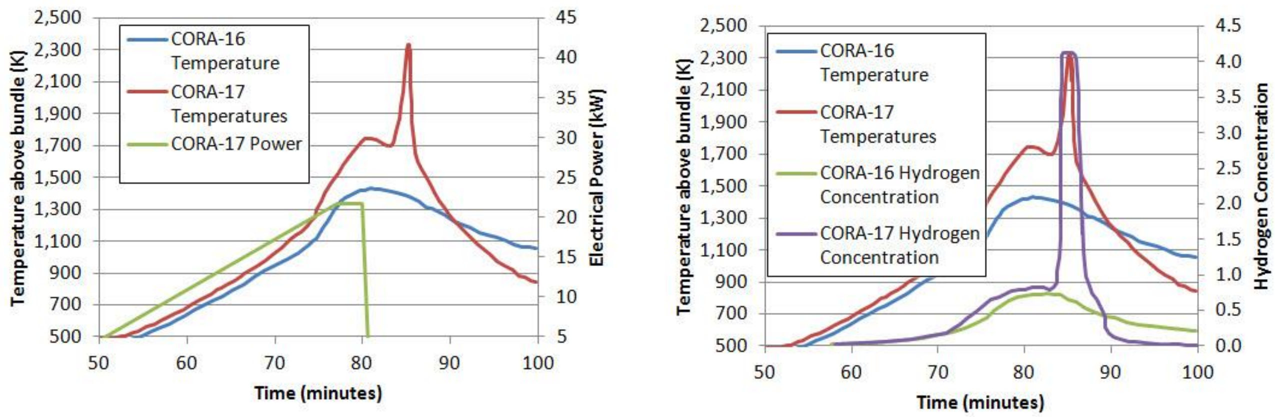

Figure 1 – Assembly power, assembly hydrogen concentration, and upper test train temperature in the KIT CORA-16 and CORA-17 experiments.

3. RELAP/SCDAPSIM/MOD3.4

RELAP/SCDAPSIM is designed to describe the overall reactor coolant system (RCS) thermal hydraulic response and core behavior under normal operating conditions or under design basis or severe accident conditions. The RELAP5 models calculate the overall RCS thermal hydraulic response, control system behavior, reactor kinetics, and the behavior of special reactor system components such as valves and pumps. The SCDAP models calculate the behavior of the core and vessel structures under normal and accident conditions. The SCDAP portion of the code includes user-selectable reactor component models for LWR fuel rods, Ag-In-Cd and B_4C control rods, BWR control blade/channel boxes, electrically heated fuel rod simulators, and general core and vessel structures. The SCDAP portion of the code also includes models to treat the later stages of a severe accident including debris and molten pool formation, debris/vessel interactions, and the structural failure (creep rupture) of vessel structures. The latter models are automatically invoked by the code as the damage in the core and vessel progresses.

RELAP/SCDAPSIM/MOD3.4 is the latest production version in the series of SDTP-developed versions. It is used by SDTP members and licensed user to support the design, simulation, and analysis for thermal hydraulic systems, nuclear power plants, and research reactor systems. The RELAP5 and SCDAP models and RELAP/SCDAPSIM have been validated for a wide range of accident conditions using a variety of experiments and plant data including TMI-2. The unique features of RELAP/SCDAPSIM/MOD3.4, relative to other versions of RELAP5, SCDAP/RELAP5 or earlier versions of RELAP/SCDAPSIM, include (a) special coding and numerics to allow the code to run significantly faster and more reliably, (b) improved or special fuel assembly behavior models, and (c) advanced 3D graphics display options. The improved fuel assembly behavior models include improved fuel rod simulator and shroud models important for the design and analysis of experimental facilities like the CORA and Quench facilities at KIT.

4. Laguna Verde Nuclear Power Plant (LVNPP) Model Description

The LVNPP input model includes reactor vessel, recirculation loops and the main steam supply system. Steam separator and steam dryer assemblies are modeled as equivalent individual components. Two recirculation loops were modeled using generic RELAP components for recirculation pumps, valves and jet pumps. A single jet pump is modeled as equivalent to 10 Laguna Verde jet pumps in both recirculation loops. Physical and thermodynamic properties of both loops are identical. The engineered safety systems included are ECCS (Emergency Core Cooling System) and RCIC (Reactor Core Isolation Cooling). The RELAP/SCDAPSIM thermal hydraulic nodalization diagram developed by CNSNS for LVNPP is shown in Figure 2. See references [1, 11, 12] for more information on the CNSNS-developed nodalization and input model.

The flow in the core is described using 8 vertical channels with four channels representing the flow area within four representative groups of fuel assemblies. The other four channels represent the flow area within the bypass regions outside the individual fuel assembly channel boxes where the B4C control blades are located. Each representative group of fuel assemblies are described using detailed SCDAP core components. The fuel rods within each representative group of fuel assemblies are described using the SCDAP fuel rod component. The fuel assembly channel boxes and control blades are described using the SCDAP BWR control blade/channel box component. The groups of fuel assemblies are grouped by fuel assembly power level and burnup. The power levels in the four representative groups vary by approximately a factor of two from the high power to lower power assemblies. The axial power profiles are dependent on the burnup level with the higher powered groups having a bottom skewed profile and the lower power assemblies a cosine shaped profile. (See reference [1] for a more detailed description.)

The lower plenum region is described using the SCDAP two-dimensional, finite element “COUPLE” model. This model describes the behavior of the lower head as well as any debris relocating into the lower plenum. This model also accounts for the natural circulation heat transfer within the liquefied or molten portion of the debris in the lower plenum.

5. Results and Discussion

A series of transients were run to look at the influence of water addition during the initial core uncovery and heatup. All of the transients were high pressure sequences with the station blackout transient starting at 510 s with reactor scram, pump trip, and main steam isolation valve (MSIV) closure. The transients included a base case transient with no water addition. This base case resulted in the destruction of the fuel assembly channel boxes and control blades around 1500 K, the liquefaction of the unoxidized Zircaloy cladding material around 2000 K, and formation of regions of liquefied or molten $\{U\text{-}Zr\}\text{-}O_2$ above 2800 K. The liquefied metallic materials, Zr, stainless steel, and B_4C, started relocating into the lower plenum at approximately 2.7 hours (9,700 s). The protective ceramic crust and outer core structures supporting the molten $\{U\text{-}Zr\}\text{-}O_2$ pool in the core region were predicted to fail at approximately 3.3 hours (11,900 s) resulting in the molten ceramic material relocating into the lower plenum through the core

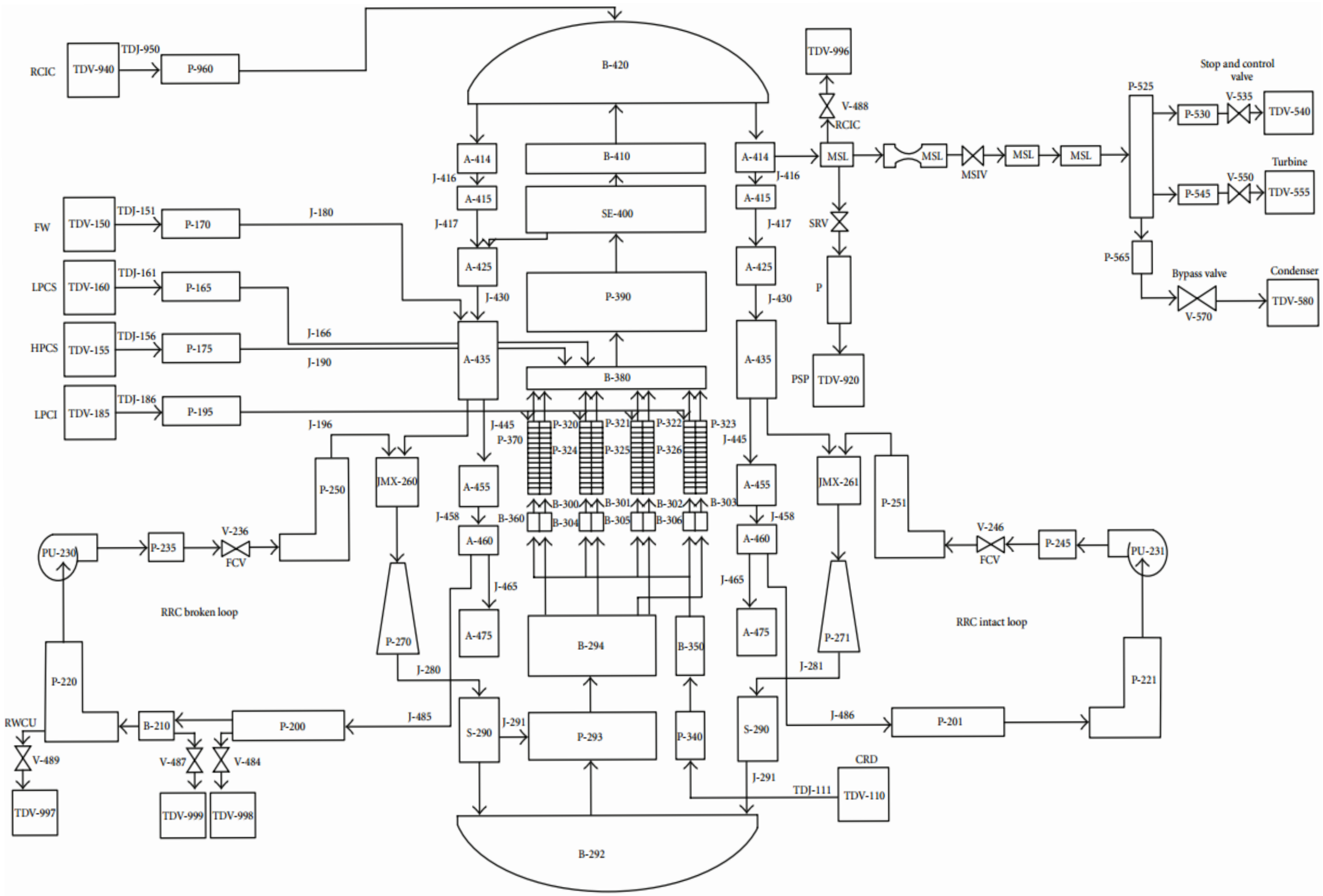

Figure 2 RELAP5/SCDAPSIM Nodalization of LVNPP

bypass region (similar to TMI-2). The base case transient was run out to 10 hours to a point well past the expected failure of the lower head.

Figure 3 shows the thermal hydraulic boundary conditions used in the transient calculations. The water injection rate was initiated at different times to give a range of core uncovery scenarios with the earliest injection starting before the collapsed water level dropped below the top of the fuel. The latest water injection scenario included in this paper was initiated after the water level had dropped to a point about midway into the core and peak core temperatures had exceeded 1500 K. It should be noted that the collapsed water level shown in the figure is measured relative to bottom of the reactor vessel. In addition, the water level shown on the figure only includes the fluid up to the top of the fuel and is thus limited to a maximum value of 9.0 m even though actual maximum water level may be higher. The top of the fuel is approximately 9.0 m. The figure also shows the integrated water addition and resulting upper plenum pressure. The integrated water addition results from the injection of subcooled water at a temperature of 333 K and 7.9 MPa. The water is injected at the top of the vessel through the RCIC connection noted on Figure 2 using a RELAP5 time dependent junction (number 950). The water is injected in a stepwise fashion varying intermittently from 25 kg/s or 0 kg/s so the integral water injection is increasing linearly when the flow is active. The times shown in the legend of 1850, 2850, 3850, 4350 and 4550 s correspond to the initial start of water injection.

Following the bundle water level in the left graph of Figure 3, the influence of the integrated water injection at varying times is easy to see. As the addition of water is delayed, more of the core is uncovered with the most extreme case shown for the base calculation with no water addition. It is also possible to see the influence of a temporary termination of water addition in the case of the water injection starting at 4550s. As the water injection is terminated, the water level starts to drop, increasing only once the water injection is restored. The influence of the water injection on the pressure in the upper dome of the vessel, the right graph in the figure, is a little more difficult to see clearly. However, there are two clear trends. First, during periods of vigorous steam generation, the pressures in the upper dome of the vessel are controlled by the set

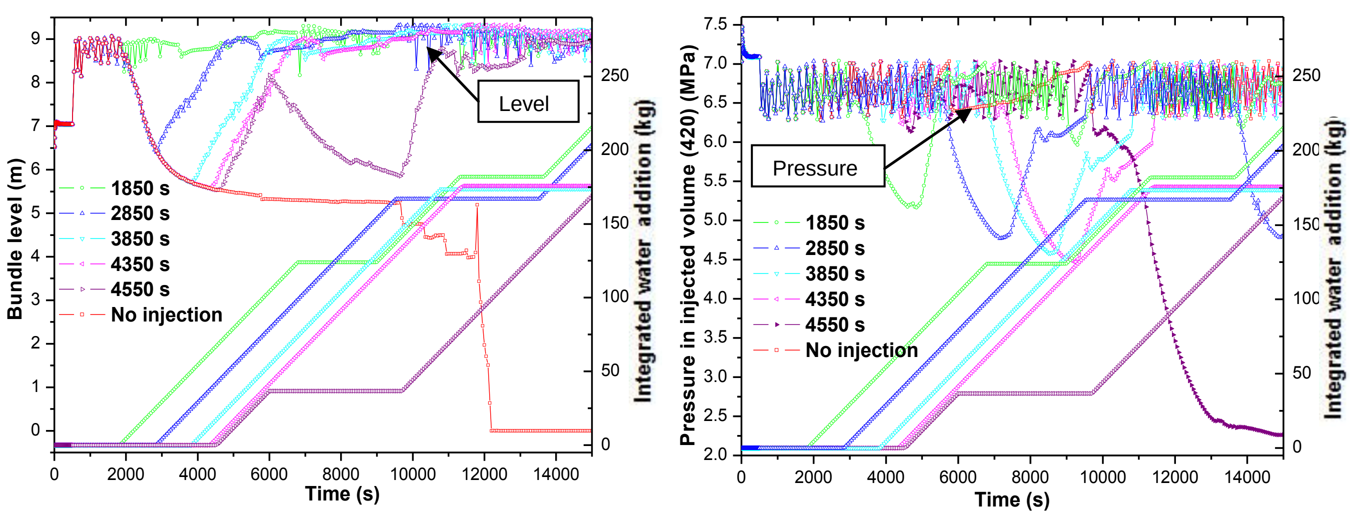

Figure 3 - Water level in the core and pressure in upper plenum with integral RCIC flow.

points on the relief valves with the pressure varying between the opening and closing set points. The second trend is the temporary drop in pressure starting with the injection of the subcooled water while the water in the vessel is heating back up to saturation conditions. It is also possible to see the change in water level in the base calculations without water addition associated with (a) the relocation of metallic core material into the lower plenum around 9700 s and (b) the relocation of a portion of the molten ceramic core material around 11,900 s.

The maximum core temperature and hydrogen generation rate are presented in Figure 4 along with the core collapsed water level. The graph on the right hand side of the figure clearly shows the influence of the drop in core water level on the maximum core temperature. The influence of the water addition as peak core temperatures reach a point between 1500 K and 2000 K can be seen most clearly in the left side of the figure showing the temperatures and hydrogen generation rate. In this case the hydrogen generation rates are shown only for the three cases including the base case with no water addition and water addition starting at 4350 s and 4550 s. Here it is clear to see the same type of behavior described for the experiments for the case of water addition at 4550 s when the peak core temperature is approaching 2000 K. This case results in a rapid generation of heat and hydrogen due to the cracking and spalling of the protective oxide film, a rapid increase in the maximum surface temperature in the core to a temperature above 2800 K, and the formation of a region of liquefied/molten {U-Zr}-O_2. Like the molten pool in TMI-2, the region of molten {U-Zr}-O_2 continues to grow even though the water level gradually moves to the top of the core region. The dip in the temperature for the case of no injection at 11800 s is due to the molten material slumping down into the lower plenum and a corresponding spike in the bundle water level shown in Figure 4.

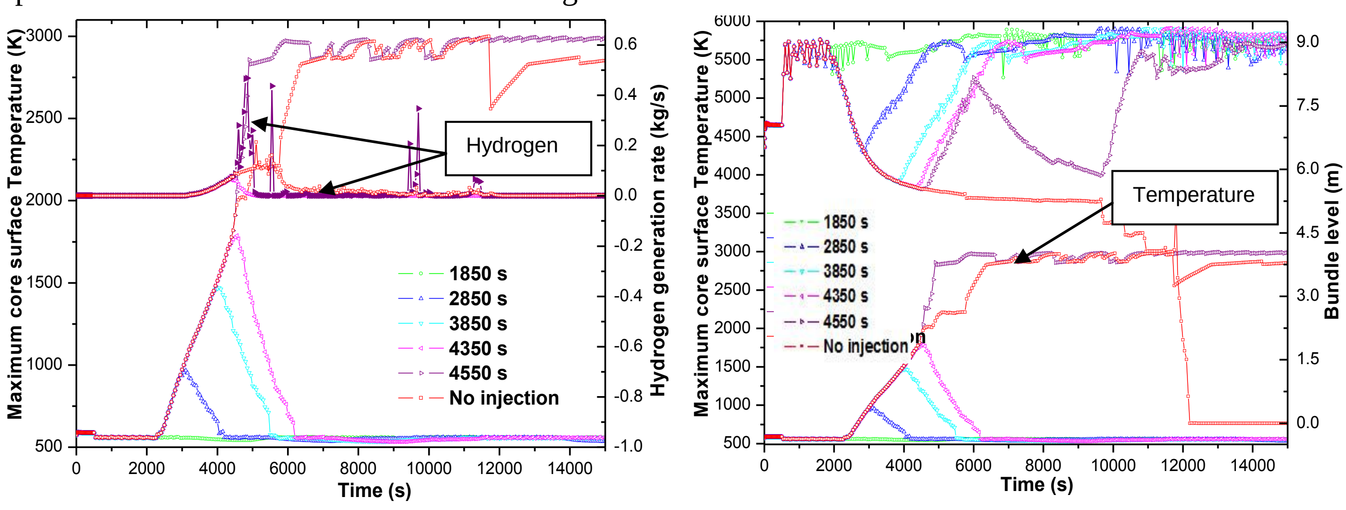

Figure 4 - Maximum core surface temperatures, hydrogen generation rates, and water level in the core.

As shown in Figure 5, which compares the total hydrogen and rate of hydrogen generation for all of the calculations during the initial core uncovery and heatup phase through 4.2 hrs (15000s), the maximum amount of hydrogen is produced in the calculations without water addition (about 300 kg) while the maximum hydrogen generation rate occurs for the case (4550 s) with water addition when peak core temperatures are approaching 2000 K (about 250 kg/s). The hydrogen generation is coming mainly from the oxidation of the Zircaloy cladding and channel box but the B_4C also makes a contribution as well. For comparison, approximately 600 kg of hydrogen is produced for the base case calculation with no water addition when the calculations are continued to the point of likely lower head failure around 10 hours. The additional hydrogen after 4.2 hrs in the base case comes from oxidation of metallic materials in the lower power assemblies in the upper core region, high power assemblies in the bottom of the core, and debris in the lower plenum.

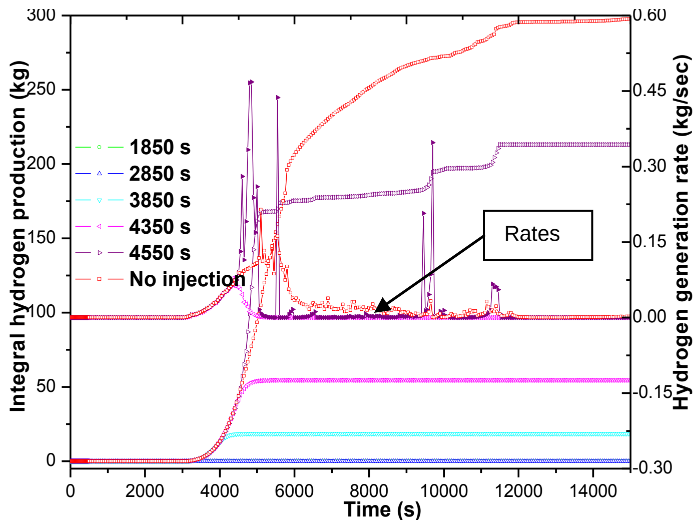

Figure 5 - Integral hydrogen production and hydrogen production rate.

4. Conclusions

The influence of the water addition is very much sensitive to the state of the core and the manner in which the water is added. As shown in the experiments and in these calculations, the temperatures of the core at the time of water addition will largely determine the maximum rates of oxidation and thus the extent of fuel melting that might occur. In the full calculations, a difference of 200 s in the start of the addition of water, which allowed peak core temperatures to exceed 1500 K and approach 2000 K before water entered the core, can determine whether the core is quickly cooled or the core melts (more precisely the ceramic core materials, $\{U\text{-}Zr\}\text{-}O_2$, and liquefied/melts in the uncovered portion of the core). The delay in water addition by 200 s also had a strong impact on the total hydrogen production with the total hydrogen production increasing from 100 kg to 200 kg.

The manner in which the water is added also has a significant impact on the results. This can be seen most clearly in comparing the two water addition cases of 4350 s and 4550 s. In these two cases, nearly the same amount of water had been injected into the vessel by 15,000 s, but the damage was significantly worse in the 4550 s case. Thus both the timing and total amount of water injected into the vessel during this critical period can be important.

5. References

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