Fukushima Type Severe Accident Analysis of Laguna Verde Nuclear Power Plant Using RELAP5/SCDAPSIM

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Abstract

Station Black out (SBO) in Boiling Water Reactor of Laguna Verde Nuclear Power Plant (LVNPP) is analyzed. Each flow channel is modeled as a pipe divided into 14 nodes. Physical and thermodynamic properties of both recirculation loops are identical. There are four steam lines and they are considered separately in the model. SBO lead to loss of cooling in the core, severe damage of the fuel and hydrogen production. The maximum core surface temperature has gone up to 3000 K with total hydrogen accumulation about 430 kg. The maximum debris temperature in the lower plenum is 4233 K.

1. Introduction

LVNPP [1] has two units and is located on the coast of the Gulf of Mexico in the municipality of Alto Lucero in the state of Veracruz. Both units of this plant have a boiling water reactor nuclear steam supply system as designed and supplied by the General Electric Company and designated as BWR-5. The design employs the drywell/pressure-suppression features of the BWR/Mark II containment concept [1]. The thermal power was uprated by 5% (from 1931MWt to 2027MWt). In December, 1999 both units were authorized to operate to power uprate conditions. In July of 2008 the Mexican Electric Power Company submitted the applications to the Mexican Regulatory Authority (CNSNS) for an operating license at power level of 2317 MWt. This corresponds to 120% of the original licensed thermal power (OLTP). This approach is referred as Constant Pressure Power Uprate (CPPU) because there isn´t changes in reactor dome pressure for this Extended Power Uprate (EPU). An important part of mitigating the damage to the core is the cooling of the debris produced during the melting of the core, but so far not known with certainty at what point should the cooling by any of the emergency systems and whether it is appropriate to do so.

The aim of this work is to present numerical experiments [2] in transient conditions to analyze the behaviour of progression of core damage, and consequences such as hydrogen generation and fission products released: 1) SBO with depressurization no CRD flow 2) SBO with depressurization and CRD flow of 2.3 kg/sec 3) SBO with depressurization and CRD flow of 4.6 kg/sec. The depressurization is started about 2 hrs after the transient. These three scenarios have been compared for the following parameters.

- Total reactor power
- Decay power

- Pressure (in channel 370)
- Maximum core surface temperature
- Bundle level (water)
- Hydrogen production
- Debris from COUPLE

1.1 LVNPP Model Description

LVNPP model [3] includes reactor vessel, recirculation loops and main steam supply system. It is a BWR-5 containing 444 assemblies with rated power 2317 MWt. These 444 assemblies are divided into 4 groups using similarity of assembly type as the criteria for division. The grouping of the assemblies is as follows; group 1, 280 assemblies, type P8X8R AVGCEN, group 2, 68 assemblies, type P8X8R AVGPER, group 3, 95 assemblies, type G9B35 AVGCEN, and group 4, 1 assembly, type G9B35 HOTCEN. This simplification is valid since all assemblies in a group have the same pressure drop at steady state. The total mass flow through each assembly grouping and the total flow area for each type of assembly is computed as the sum of mass flows and flow areas for every assembly in the same group. Two flow channels (active and bypass) were used to model each active assembly. Each flow channel is modeled as a pipe divided into 14 nodes. Steam separator and steam dryer assemblies (129 components in each assembly) 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 [3] 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 [4] designed are ECCS and RCIC (Reactor Core Isolation Cooling). ECCS consists of four sub systems: High Pressure Core Spray System (HPCS), Low Pressure Core Spray System (LPCS), Low Pressure Injection System (LPCI) that inject in the upper plenum and Automatic Depressurization System (ADS). ADS is a logic system actuator of Safety Relief Valve (SRV) to depressurize the reactor vessel in the events of HPCI fails [5]. The nodalization diagram [1, 2] representing LVNPP is shown in Figure 1. The operating conditions correspond to power level of 2317 MWt. This model includes the following main elements: 1) feed-water system, 2) reactor vessel and internals, 3) recirculation loops, 4) the reactor core, 5) main steam line 6) bottom of the reactor vessel. The bottom model of the vessel is a fundamental element for the analysis in accident scenarios, for which is included as a separate element that interacts with the other models.

2. RELAP5/SCDAPSIM Code

This code, designed to predict behavior of PWR/BWR systems during normal and accident conditions, is currently under development at Innovative Systems Software (ISS) as part of the international SCDAP Development and Training program (SDTP). It uses RELAP5/SCDAP models developed by US Nuclear Regulatory Commission (USNRC). It is a combination of RELAP5 [6] to calculate overall RCS thermal hydraulic response, control system behaviour, reactor kinetics, and the behavior of special reactor system components such as valves and pumps, SCDAP code for severe accident related phenomena and COUPLE code for a finite element analysis of vessel lower head. SCDAP fuel rod and control components use 2D models to predict temperature (r,z), deformation, chemical interactions and melting. The SCDAP [7] 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 latter models are automatically invoked by the code as the damage in the core and vessel progresses. The code used for this work is RELAP5/SCDAPSIM MOD3.4.

A newer version [2], available with improved SCDAP modelling options is RELAP5/SCDAPSIM MOD3.5. The new SCDAP modelling options include (a) an improved fuel rod gap conductance model, (b) improvements in the electrically heated fuel rod simulator model, (c) improvements in the shroud model, and (d) models to treat the influence of air ingression. The improved electrically heated fuel rod simulator model now includes the option to model tantalum heater elements in addition to the tungsten heater elements historically used in the QUENCH and other European bundle experiments.

2.1. Accident Scenario

The SBO transient is modeled as closure of main steam isolation valves (MSIV) with an assumed unavailability of several critical safety systems [2], which lead to severe core damage. A severe accident is one that exceeds the design basis of the plant sufficiently to cause failure of structures, materials, systems, etc. without which it can ensure proper cooling of the reactor core by normal means. Faced with a severe accident the first response is to try to maintain the core cooling by any means available, but in order to carry out such an effort is understand necessary the progression of core damage, because such action has effects that may be determinants in the progression of the accident.

3. Results and Discussion

Null transient has been run for 500 seconds to stabilize the thermal hydraulic conditions in the system. Transient for the three cases mentioned above has been run for 23000 seconds (6.38 hrs).

3.1. Total Reactor Power

Figure 2 depicts the behaviour of the power. It decreases immediately after the scram because of negative temperature coefficient of reactivity to a value of 80 MWt in about 120 seconds. It further stabilizes to 23.6 MWt at 21600 seconds (6 hrs). Water addition through CRD has no significant impact on power as can be inferred from figure.

3.2. Decay Power

Just after the shutdown of the reactor the decay power is 164 MWt as shown in Figure 3. At about 6 hrs, the decay heat is 23.9 MWt which is consistent with the total power. That the small amount of water addition is not able to remove decay heat is obvious from the figure.

3.3. Core Pressure

Figure 4 shows the pressure in the channel 370. The fluctuating behaviour is maintained in all three cases and can be attributed to uncertainties in reactor because of high temperatures. The hump in pressure plot corresponds to the depressurization of the core by opening SRV's. The depressurization does not significantly drop the pressure in case of 2.3 kg/sec water addition because this small amount of water will be converted into steam at high temperatures.

3.4. Core Maximum Surface Temperature

The behavior of the maximum surface temperature in the core is presented in Figure 5. For the case of no water addition, the temperature increases linearly to a temperature of 2200 K at 1.4 hrs and under this condition fuel clad starts melting, no increase in temperature took place for some time. Temperature increases to its maximum value of 2988 K, 3027 K and 3028.6 K respectively in the three cases. Depressurization has significant impact in the first two cases which results into sharp temperature drop. Temperature is maximum in the case of water addition because of increased oxidation due to small water addition through CRD.

3.5. Bundle Level

Bundle level is shown in Figure 6 for all three cases. The bundle level initially rises to its maximum value of 9 metres and become constant almost. At about 2 hrs the sudden drop in the level is because of depressurization. The level continuously decreases and almost zero after the 3.33 hrs in the case of no water addition while in other two cases, it is 0.4 and 0.45 respectively because of water addition.

3.6. Integral Hydrogen Generation

The total hydrogen generated for three cases is 431 kg, 400 kg and 428 kg respectively as shown in Figure 7.

3.7. Debris from COUPLE

The debris formed in the lower plenum 292 is presented in Figure 8 to Figure 10. They are saved at 6.38 hrs (23000 seconds) and red area shows hot debris. The maximum debris temperature occurring is 4233 K while it is not clear from figure because till 6 hrs the debris is cooled because of water addition. Figure 11 shows maximum temperature which is the case of double CRD flow (4.6 kg/sec). It is because of water addition the damage progression has been delayed which is the purpose of CRD flow. So water addition delayed the occurrence of maximum debris temperature and that is why, we are able to see more red area in this case

4. Conclusion

SBO in the BWR of the LVNPP at power level of 2317 MWt has been analyzed in this work using RELAP/SCDAPSIM code. MSIV closure and ECCS failure took place because of power failure which leads to severe core damage. We observed the maximum core surface temperature of 3000 K with maximum debris temperature of 4233 K. The results of the three cases indicate that CRD flow can delay the damage progression. Intentional depressurization of the vessel after loss of cooling can have a very significant (negative) influence on the destruction and melting of the core. For the calculations for Laguna Verde, the depressurization of the vessel upon loss of emergency cooling accelerated the uncovery of the core and melting of the fuel to a significant degree.

5. References

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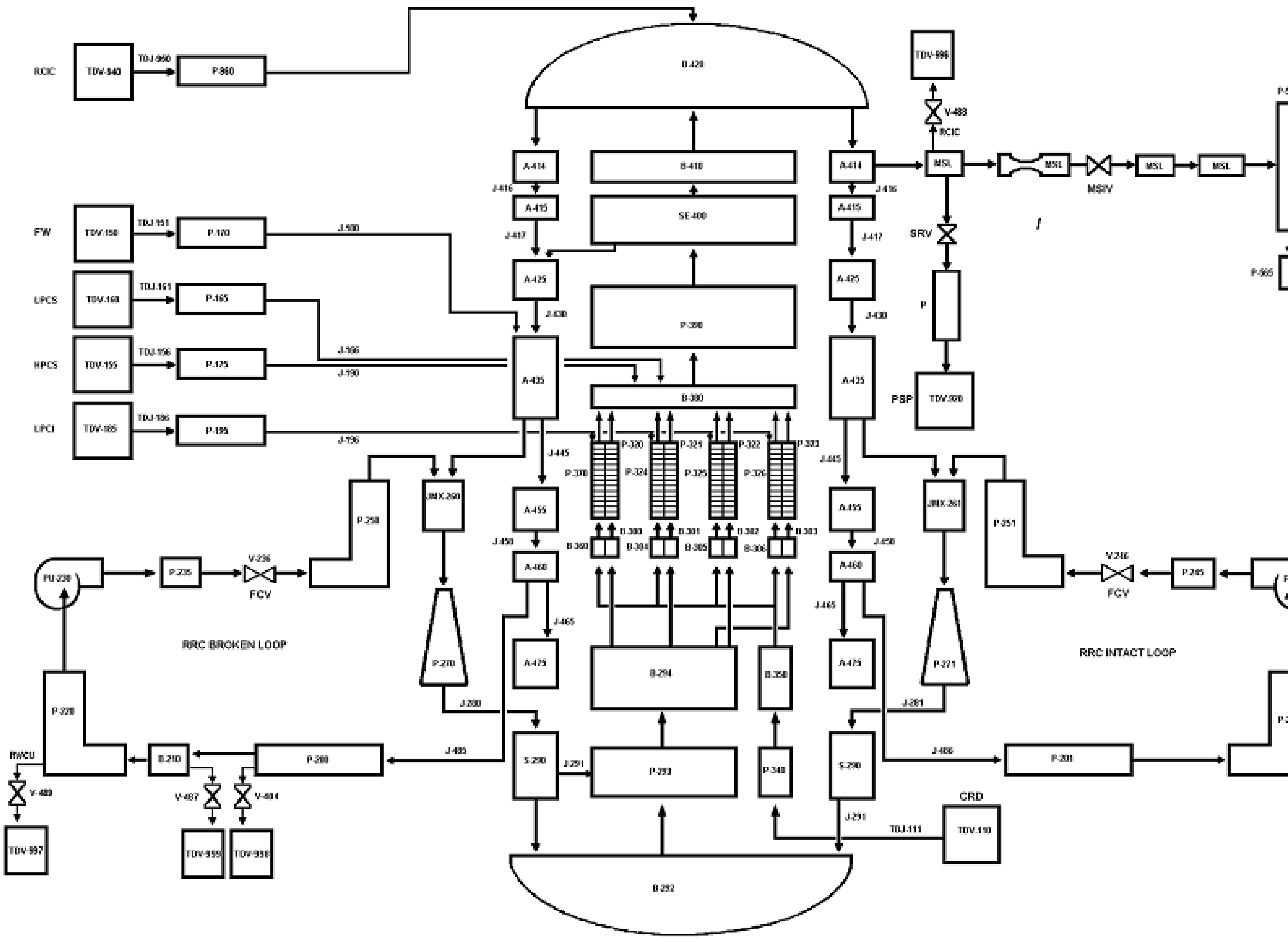

Figure 1 RELAP5/SCDAPSIM Nodalization of LVNPP [14]

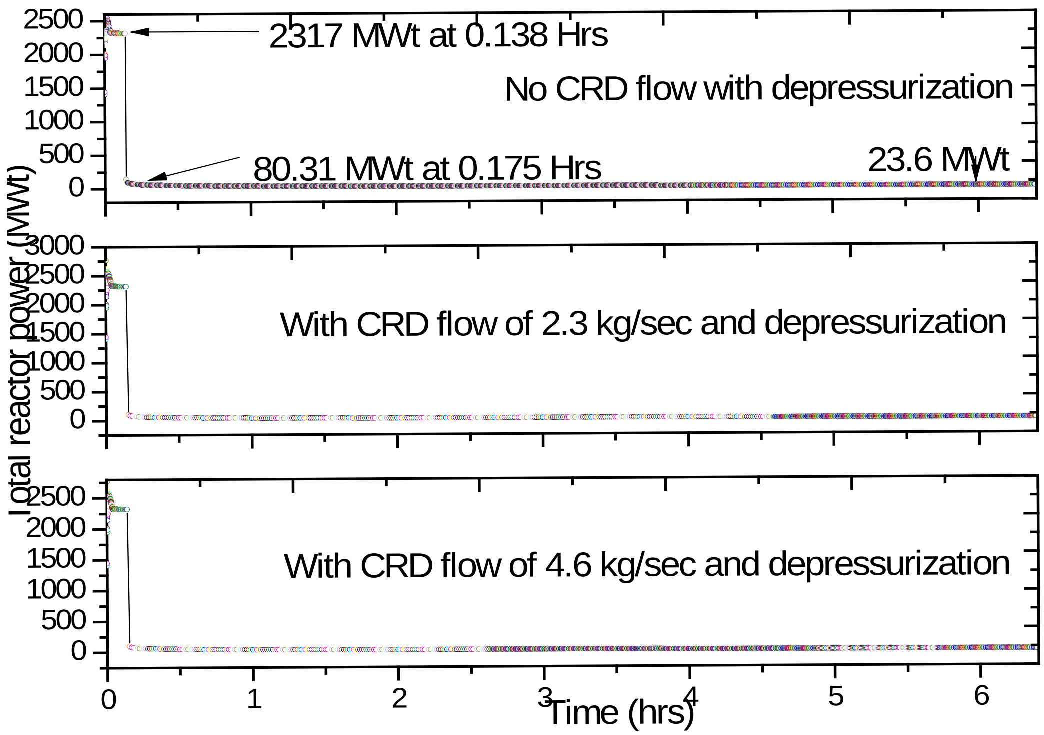

Figure 2 Total Reactor Power

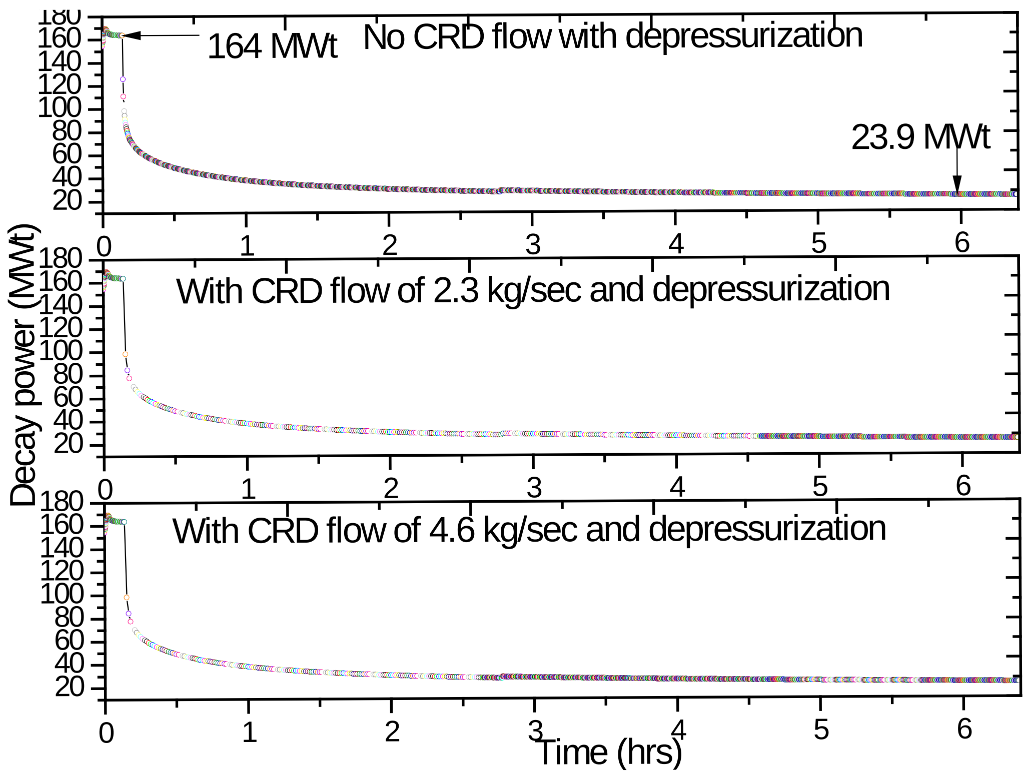

Figure 3 Decay Power

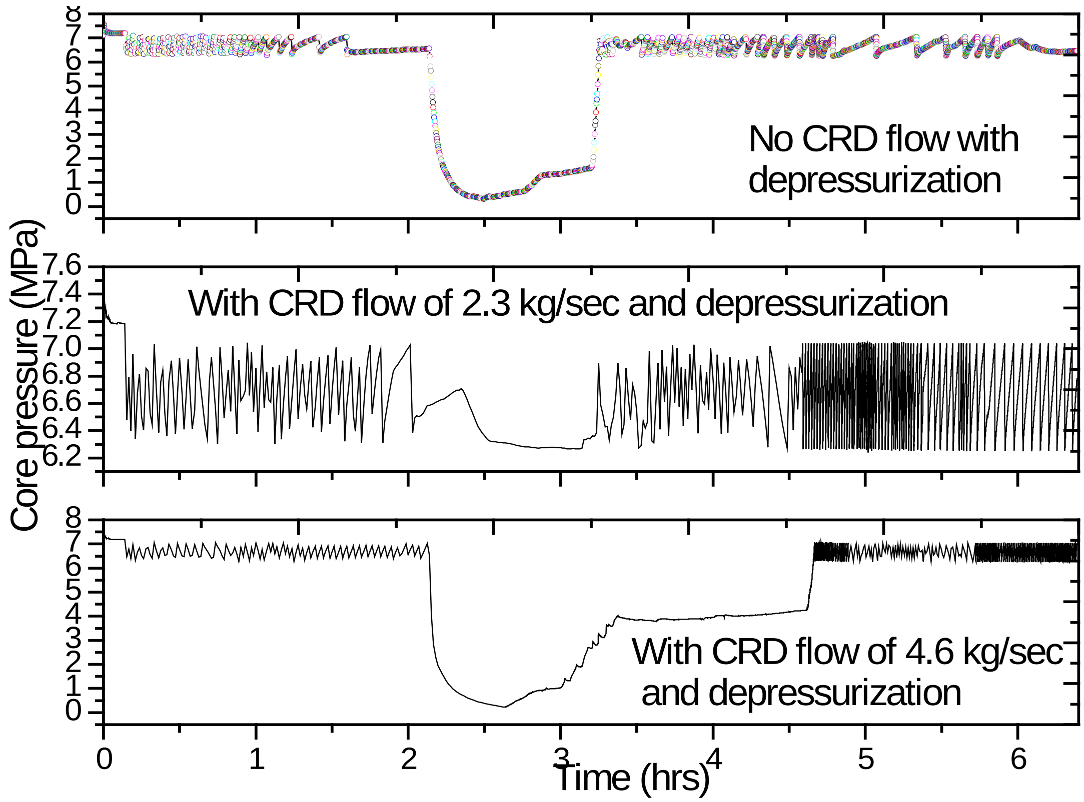

Figure 4 Core Pressure

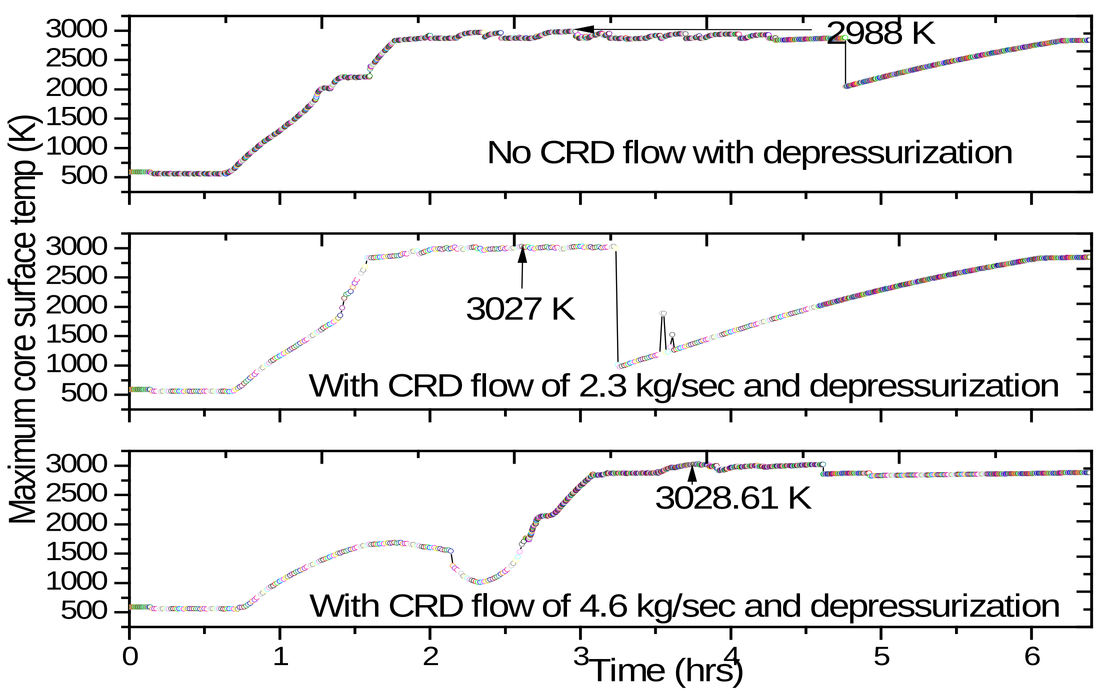

Figure 5 Maximum Core Surface Temperature

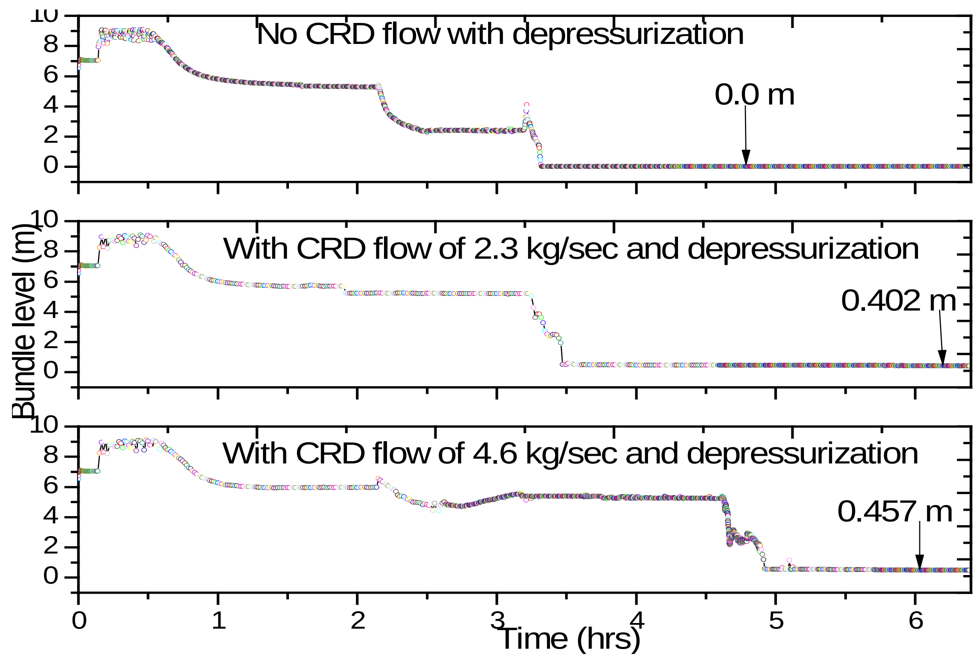

Figure 6 Bundle Level

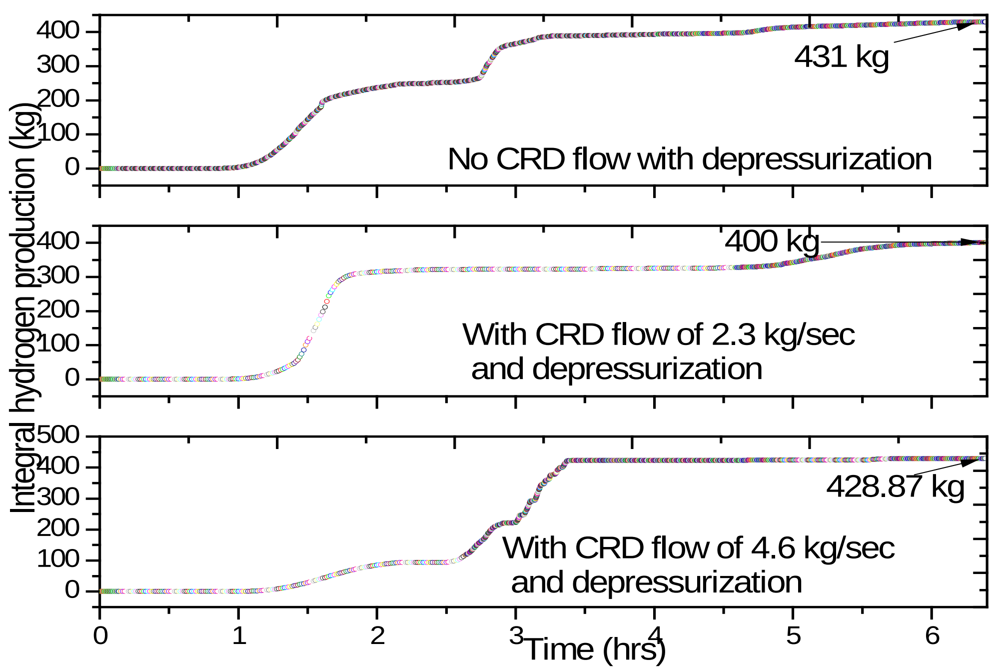

Figure 7 Integral Hydrogen Productions

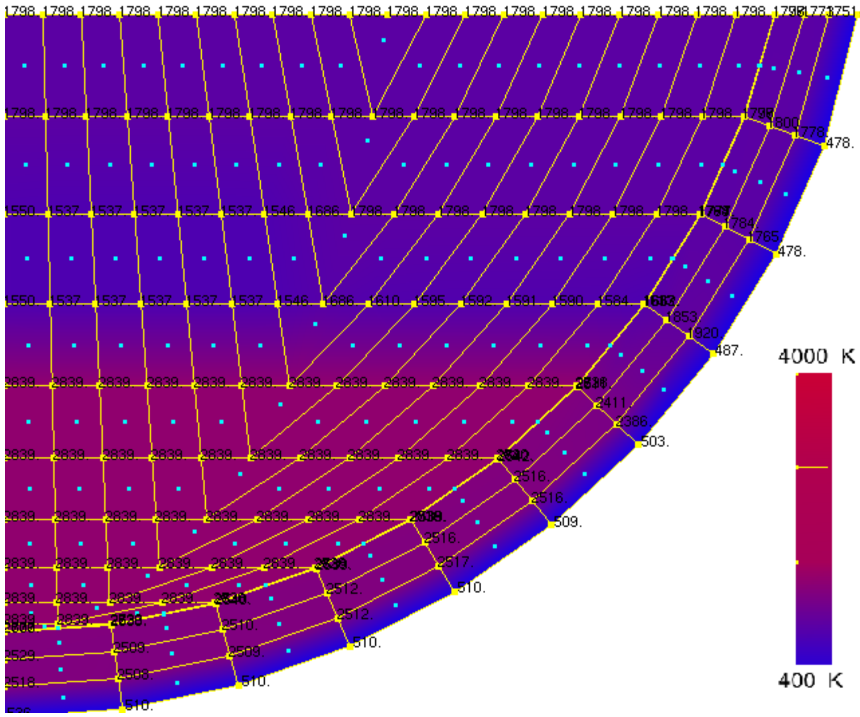

Figure 8 Debris no CRD Flow with Depressurization

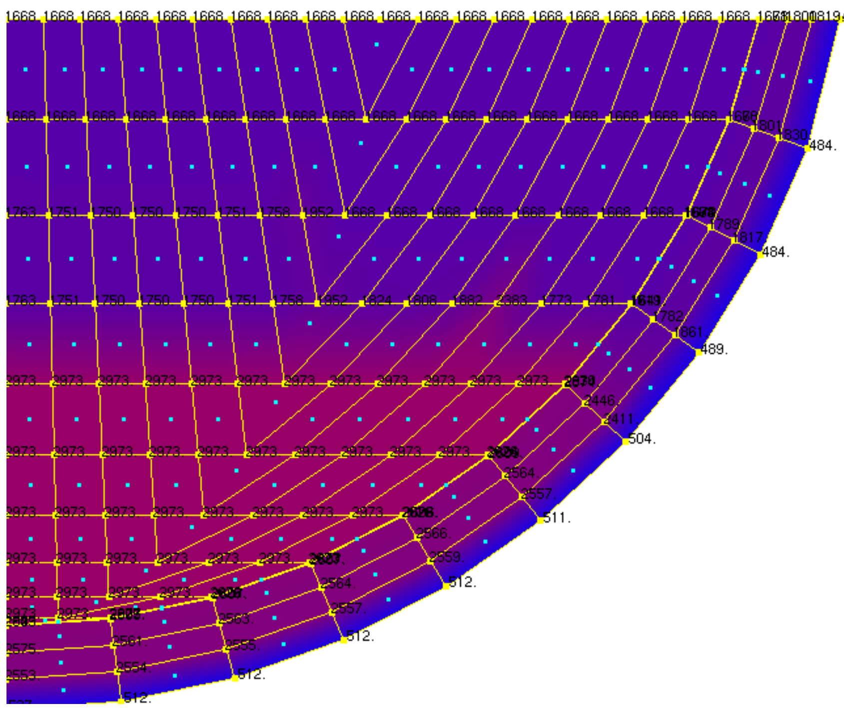

Figure 9 Debris with CRD Flow of 2.3 kg/sec and Depressurization

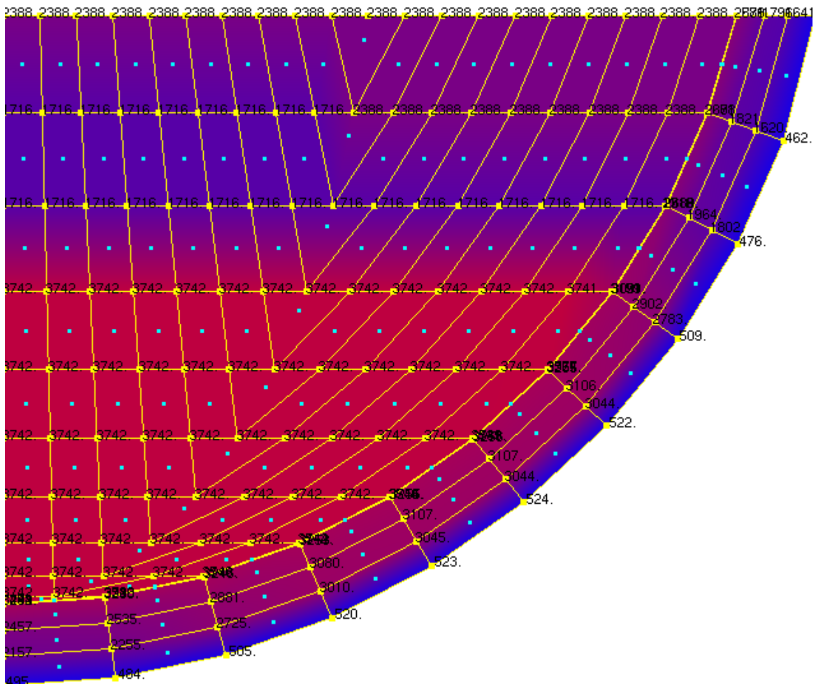

Figure 10 Debris with CRD Flow of 4.6 kg/sec and Depressurization