

# **Steam Line Break and Station Blackout Transients for Proliferation Resistant Hexagonal Tight Lattice BWR**

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## Abstract

Safety analyses of a proliferation resistant, economically competitive, high conversion, boiling water reactor (HCBWR) fueled with fissile plutonium and fertile thorium oxide fuel elements, and with passive safety systems are presented here. The HCBWR developed here is characterized by a very tight lattice with a relatively small water volume fraction in the core which therefore operates with a fast reactor neutron spectrum, and a considerably improved neutron economy compared to the current generation of Light Water Reactors. The tight lattice core has a very narrow flow channels with a hydraulic diameter less than half of the regular BWR core and, thus, presents a special challenge to core cooling, because of reduced water inventory and high friction in the core. The primary safety concern when reducing the moderator to fuel ratio and when using a tightly packed lattice arrangement is to maintain adequate cooling of the core during both normal operation and accident scenarios.

In the preliminary HCBWR design, the core has been placed in a vessel with a large chimney section, and the vessel is connected with Isolation Condenser System (ICS). The vessel is placed in containment with Gravity Driven Cooling System (GDCCS) and Passive Containment Cooling System (PCCS) in a configuration similar to General Electric's Simplified Boiling Water Reactor (SBWR). The safety systems are similar to SBWR; ICS and PCCS are scaled with power. An internal recirculation pump was placed in the downcomer to augment the buoyancy head provided by the chimney, since the buoyancy provided by the chimney alone could not generate sufficient recirculation in the vessel as the tight lattice configuration resulted in much larger friction in the core than the SBWR.

The constitutive relationships for RELAP5 were assessed for narrow channels, and as a result the heat transfer package was modified. The modified RELAP5 was used to simulate and analyze two of the most limiting events for a tight pitch lattice core: the Station Blackout and the Main Steam Line Break events. The results of the analyses indicate that the HCBWR system will be safely brought to the shutdown condition for these transients.

## 1. Introduction

The overarching objective of the research and development presented here is to advance the well-developed water-cooled reactor technology in order to make efficient use of the abundant thorium resources and enhance the proliferation resistance of the nuclear fuel cycle. Considerable effort has been invested in development of the sodium cooled fast reactor to breed fissionable  $^{239}\text{Pu}$  from natural uranium. Much less effort has been expended into development of technologies that take advantage of the considerable experience with light water reactors as an alternative to the fast reactor for providing a hard neutron spectrum to convert thorium into the fissile  $\text{U}^{233}$  isotope.

The analyses presented here supports the feasibility of a plutonium-thorium (Pu-Th) fuel cycle for a new type of high conversion reactor cooled by boiling water (HCBWR) that will burn existing stocks of plutonium, while converting the fertile thorium to fissile  $^{233}\text{U}$ .

The extensive previous research on high conversion LWRs in the U.S., Europe, and Japan over the last several years provided a useful starting point for the design developed here [1, 2]. A design objective has been to achieve a high conversion of thorium to  $^{233}\text{U}$  and to reduce the national accumulated inventory of plutonium, as well as to take advantage of the enhanced proliferation resistance of the thorium fuel cycle. The high conversion will take place in a fast neutron spectrum produced by minimizing the volume of water in very tight fuel assembly lattices. High fuel burnup will be possible as a result of the continuous generation and fission of  $^{233}\text{U}$  as the plutonium is consumed. The high burnups achievable with thorium fuel in a high conversion core will increase the plant capacity factor and lower the overall cost of electricity. Inherent safety will be designed into the reactor because of the favorable feedback neutronics characteristics of thorium and by the use of innovative core heterogeneities. This will insure a negative void coefficient for those accident sequences which result in off-normal coolant boiling. The HCBWR thus developed and presented here is characterized by a very tight lattice with a relatively small water volume fraction in the core which therefore operates with a fast reactor neutron spectrum and a considerably improved neutron economy compared to the current generation of Light Water Reactors.

The primary safety concern when reducing the moderator to fuel ratio and when using a tightly packed lattice arrangement is to maintain adequate cooling of the core during both normal operation and accident scenarios. A tight lattice BWR core has very narrow flow channels with a hydraulic diameter less than half of the regular BWR core. This type of configuration, while reducing moderation, leads to an increase in the frictional pressure drop and a reduced coolant inventory. Maintaining safety margins during the normal operation and during hypothetical accident scenario will be a challenge.

## **2. Conceptual Design of HCBWR System**

A scoping study for a HCBWR was initiated to establish a reactor configuration, which would lead to safe operation with both simplicity and passive safety features. The Simplified Boiling Water Reactor (SBWR) [3, 4] provides these characteristics and was used to initiate the design study. The HCBWR studied in this paper was designed with approximately twice of the power of the SBWR. A conceptual design and safety analysis for this design were performed using RELAP5 computer code [5] that has been modified to account for correlations valid for narrow triangular channels. The constitutive relationships in RELAP5 were assessed with the correlations available for narrow channels: it indicated that only single phase heat transfer correlations were not applicable and were replaced by Subbotin correlation [6] for laminar flow and Pelukhov and Rozzen correlations [6] for turbulent flow.

In the first preliminary configuration, the tight lattice core was placed in a vessel with a chimney in a similar geometry as an SBWR to investigate whether sufficient buoyancy head could be generated to provide a recirculation flow by natural circulation to maintain the core in a coolable condition without pumps. The heights of the vessel and other internal components such as the chimney were similar to those of the GE-SBWR, however the areas of the flow paths of the internal components except core were sized to be approximately twice of GE-SBWR as the power of the proposed reactor (3926 MWt)

was about twice of a SBWR (600MWe/ 2000MWt). This design was analyzed with the modified RELAP5 for steady state operation. Figure 1 shows the nodalization diagram of the vessel for the analysis. The major components of the vessel are the downcomer, core, chimney, separator and dryer. The core consists of four parallel channels, i.e., a bypass channel, a average fuel assembly channel, a water-tube assembly channel, and a hot channel. The core is divided into 10 axial nodes to reflect the axial power distribution. The main observation from the steady state calculations was that the natural circulation with the SBWR-size chimney as the driver was not sufficient to cool the fuel in the core because of the relatively small flow areas and large friction losses in the tight lattice core. The flow through the heated channels was not sufficient to maintain nucleate boiling and the heat transfer mode in the upper sections of the channels turned to film boiling. In addition the flow was not stable. It was apparent that a longer chimney would be needed to generate a sufficient driving force by natural circulation to overcome the large pressure drop in the core. A parametric study, where the height of the chimney was gradually increased, showed that at least a twice-long chimney would be needed to generate sufficient natural circulation to cool the core; however, the recirculation ratio (ratio of core flow to feed water flow) was still much lower than that of the SBWR, i.e., about 2.6 compared to about 7.0 of the SBWR. To achieve the same recirculation ratio as an SBWR, the chimney had to be about 8 times longer than that of SBWR. Since the neutron kinetics will be affected by the recirculation ratio, the actual recirculation ratio will be calculated in conjunction with a neutron kinetics study.

In the second configuration, the SBWR chimney (9m) was retained and a pump was added between the lower plenum and the downcomer similar to the Advanced BWR (ABWR). The pump was sized to achieve a recirculation ratio similar to that of the ABWR. The flow in this case was stable and the fuel was cooled. It is concluded that a pump would be needed to have adequate flow in the core without substantially increasing the vessel height. While the reactor operation will not be passive, the pump will provide easy startup and better control. Furthermore, this chimney/pump design will provide an important protection by providing sufficient flow in the core to maintain the core coolable in case of loss of power (thus, loss of forced flow by pump) such as a station blackout accident. A parametric study was performed by gradually decreasing the reactor power without pumps; it showed that the chimney/pump design would generate a sufficient flow to cool the core without pumps in case of a station blackout accident up to 50 % of designed power. Table 1 shows the comparison of primary design parameters of current HCBWR design and SBWR.

In order to complete the specification of reactor system, the new vessel was placed in the containment system designed for SBWR. In the SBWR design, the containment plays an important role and is very tightly coupled with the vessel during design-base accident scenarios. All the safety systems reside in the containment. The safety systems directly attached to the vessel are Isolation Condenser System (ICS), Automatic Depressurization System (ADS) and Gravity Driven Coolant System (GDACS). The ICS is a bank of heat exchangers that is designed to remove decay heat when the reactor is isolated by closing the main steam isolation valves (MSIV Closure) and the core is scrammed. The heat exchanger tubes are immersed in the water tanks located on

the top of the containment. The ICS removes heat from the reactor system by condensing steam flowing into ICS tubes from the RPV. The condensate is returned to the RPV. The depressurization system, consisting of a set of valves to reduce the vessel pressure, allows passive, gravity driven coolant injection from the GDCS. The GDCS is a part of the emergency core cooling system (ECCS) in the SBWR and HCBWR: it operates on gravity head instead of pumps. It consists of water tanks, which are located at a higher elevation and are connected to the RPV.

The containment has suppression pool (SP) and passive containment cooling system (PCCS). The SP is designed to absorb energy in the early part of the LOCAs. The PCCS is designed to transfer energy from the containment to an external pool by condensing steam.

The ICS and GDCS are designed to protect the fuel by keeping the peak clad temperature (PCT) within the safety limit and the last two safety systems (SP and PCCS) are designed to the increase of the containment pressure.

In this study, the containment is selected to be of the same size as specified for SBWR even though the power of HCBWR is twice of that of SBWR. However, the capacities of the safety systems were scaled in proportion to the power ratio. This implies that, if the PCT and the containment pressure are shown to be maintained within the safety limit, we can have a design with smaller containment relative to the power, which will significantly reduce the cost.

**TABLE 1 Comparison of Primary Design Parameters of BWR and HCBWR**

Item	SBWR	Tight Lattice BWR*
Power, MWt	2000	3926
Core Diameter, m	5.0	5.8
Core Height, m	3.2	2.2
Chimney Height, m	9.1	9.1
Hydraulic Diameter Average Channel, cm	1.56	0.43
Hydraulic Diameter, peripheral channel, cm	1.56	
Hydraulic Diameter, Void Channel, cm		0.62
Flow Area Average Channels, m <sup>2</sup>	5.74	5.04
Flow Area Peripheral Channels, m <sup>2</sup>	1.62	
Flow Area Void Channels, m <sup>2</sup>		1.036
Flow Area Bypass, m <sup>2</sup>	5.3	2.19
Flow Area Downcomer, m <sup>2</sup>	6.5	13
Pressure Drop Across Core, bar	1.1	3.4
Recirculation Flow Ratio	7.0	7.4

\* With recirculation pump

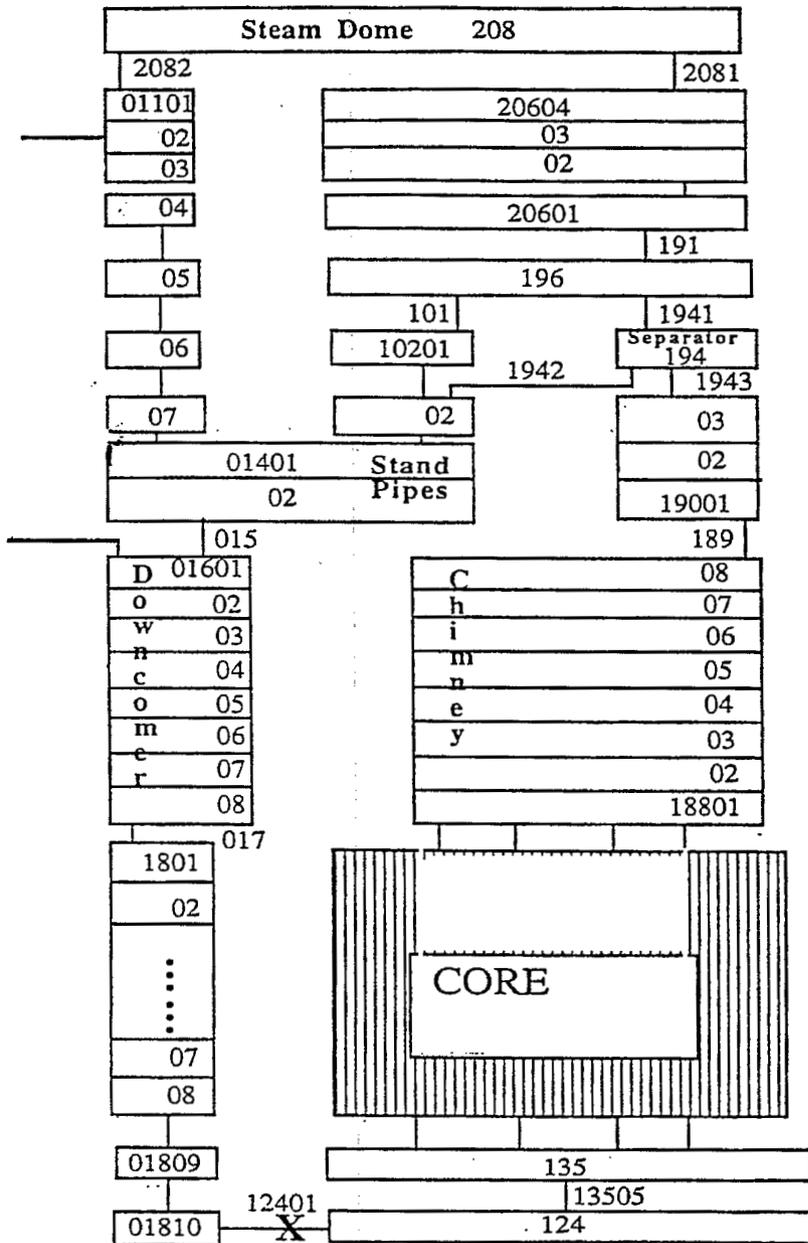


Figure 1 RELAP5 Nodalization Diagram of RPV

### **3. Transient Analyses**

The modified RELAP5, which was discussed in Section 2, and the nodalization of the HCBWR system as specified in the Section 2 were used to simulate two important transients; Station Blackout and Main Steam Line Break. These two transients will provide maximum challenge to safety system under accident conditions when scram is functioning. As this design is new and is based on passive systems, it is important to analyze the reactor system performance under the accident conditions.

#### **3.1 Station Blackout**

HCBWR is designed, like an SBWR, to be brought to the shutdown condition without operator intervention in case of a station blackout (SBO). An SBO transient was analyzed to make sure that the decay heat can be removed from the core and the reactor system can be brought to a safe shutdown condition.

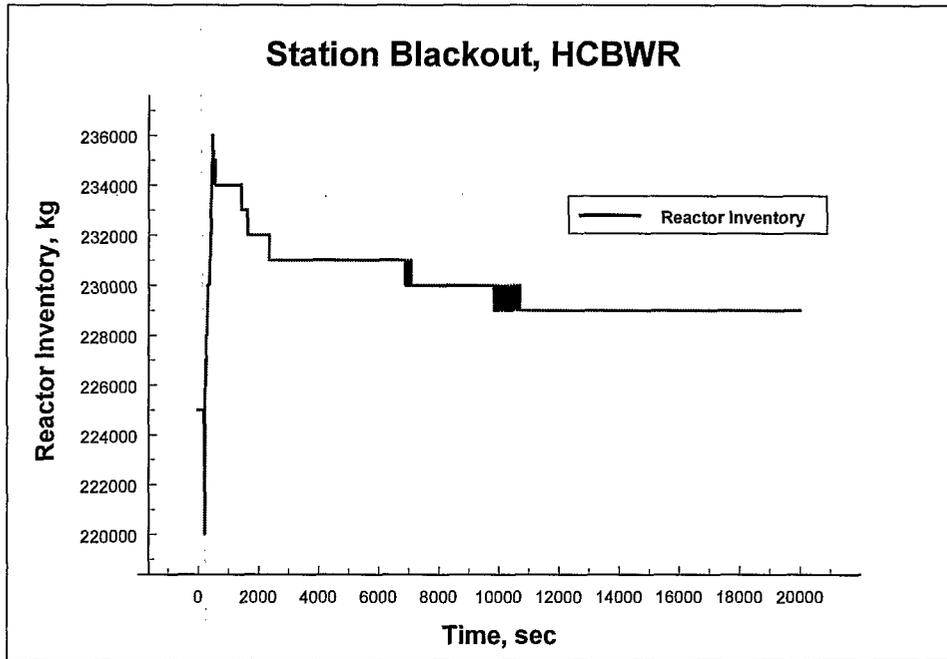
The SBO transient begins with the loss of off-site power. Upon loss of power, the reactor scrams, the feed water pumps starts to coast down, the main steam isolation valves (MSIVs) close and the ICS valves open. At this point, the RPV is isolated. The ICS is the main component that passively removes the decay heat from the vessel during this transient. The ICS is a bank of heat exchangers; which are connected to the RPV. The heat exchanger tubes are immersed in the water tanks located on the top of the containment. The ICS removes heat from the reactor system by condensing steam flowing into ICS tubes from the RPV. The condensate is returned to the RPV.

Figure 2 shows the water inventory in the vessel remains essentially constant at about 230,000 kg; the initial peak is due to the water accumulated in the ICS flowing back to the vessel, and a slight change later is due to the redistribution of coolant between the ICS and the vessel. Figure 3 shows the condensate flow rate from the ICS to the vessel. After the initial peak (which is due to the water accumulated in the heat exchanger tubes and pipes flowing back to the vessel), the flow rate stabilizes at about 100 kg/sec and the gradually decreases as the decay heat decreases. At about 5,000 seconds, oscillations in the ICS flow are observed due to condensation; the condensation reduces pressure that increases the steam flow rate from the vessel and the pressure increases, leading to a reduction in steam flow. The steam condenses and condensate flows back to the vessel. The heat removal rate is estimated to be about 78 MW based on average condensation flow, which is about 2% of full power. Figure 4 shows a comparison of decay heat and ICS heat removal rate. The figure indicates that ICS removes more heat than the decay heat for first 5 hours and then there is an equilibrium. This excess heat removal by ICS contributes to the decrease in the RPV pressure, as shown in Figure 5. The vessel pressure decreases steadily, as expected, as the coolant cools down and contracts. The pressure stabilizes at about 7.5 bar.

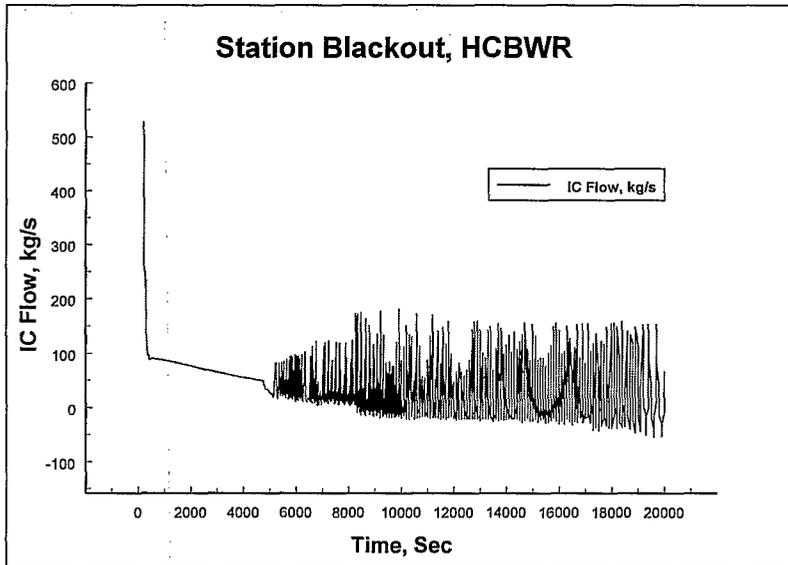
Figure 6 shows the clad temperature for hot fuel rod. The clad temperature decreases during the course of the transient and, as expected, is always below the safety limit. The

clad temperature stabilizes at about 440K (167C) that corresponds to saturation pressure of about 7.5 bars.

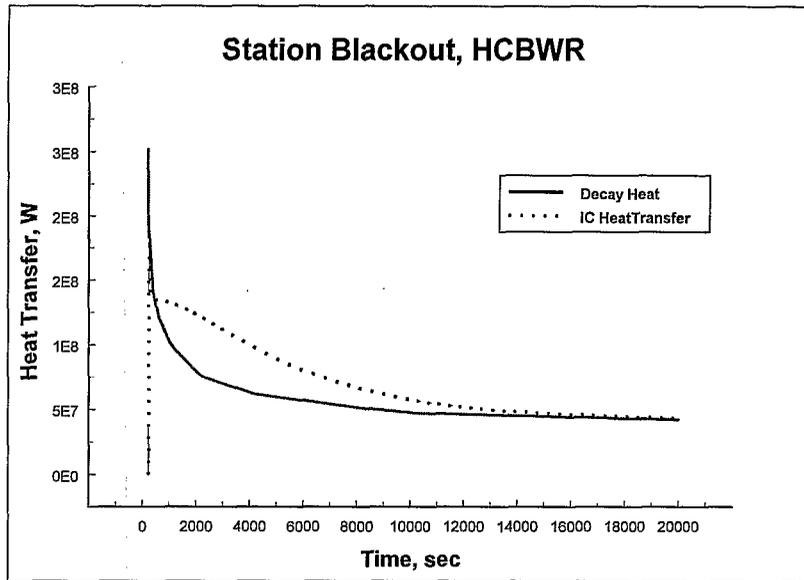
The simulation of this transient confirms that the ICS system is capable of removing the decay heat and the primary system can be maintained in a safe shut down condition under the loss of power or station blackout.



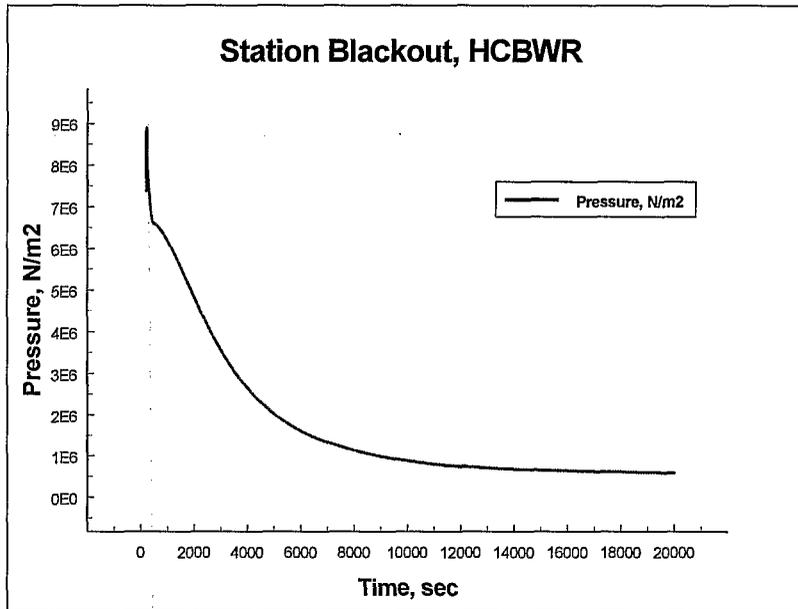
**Figure 2 Reactor Vessel Mass Inventory**



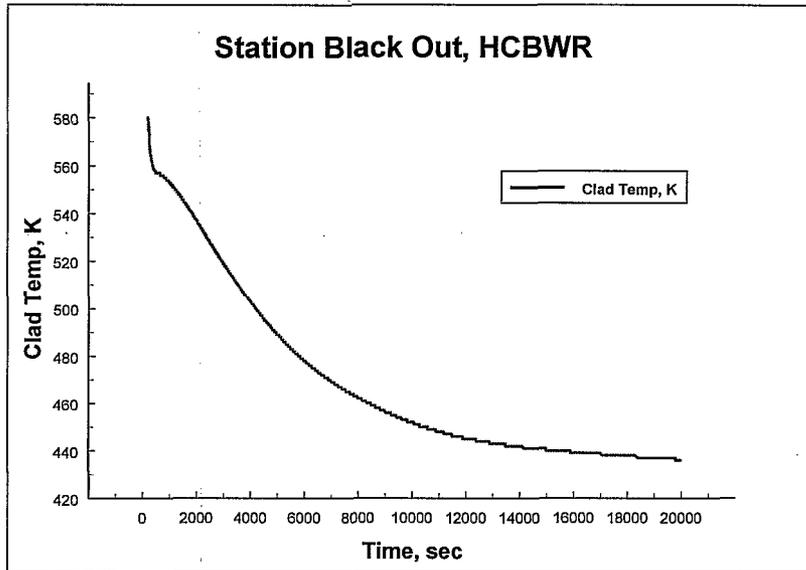
**Figure 3 Condensate Flow from Isolation Condenser**



**Figure 4 Decay power and heat transfer in Isolation Condenser**



**Figure 5 Primary side pressure**



**Figure 6 Hot Channel Clad Temperature**

### 3.2 Main Steam Line Break (MSLB) Transient

General Electric-Nuclear Energy has analyzed a full spectrum of postulated loss of coolant accidents (LOCA) with varying break sizes to evaluate the SBWR design. They include ruptures of a feedwater line, an IC return line, a GDCS line, a vessel bottom drain line, a main steam line and a DPV stub tube. An instantaneous guillotine rupture of a main steam line inside the containment, i.e., Main Steam Line Break (MSLB) transient was found to generate the highest short-term peak pressure and the most limiting short term response in the containment (i.e., immediately after the break.). Since the design of HCBWR is very similar to the SBWR design, MSLB was selected to be the limiting LOCA to be analyzed. This section presents and discusses results of the simulation of the MSLB transient for initial 20,000 seconds.

The double-ended guillotine break of a main steam line is postulated to occur upstream of MSIV concurrent with a complete loss of power (station blackout). Scram occurs at the time of the loss of power. It is modeled in such a way that, at the time of break, the steam blows down to the containment through both lines until the Main Steam Isolation Valves (MSIVs) are closed. Once the MSIVs close, the blowdown from the intact steam line stops, however the blowdown from the broken steam line continues. MSIVs start closing at 0.5 seconds after the break due to low steam line pressure or high steam flow rate and completely close in 5 seconds. ICS opens as the MSIV starts to close. The feedwater pumps start to coast down at the time of break due to loss of power and the feed water flow is assumed to stop in 5 seconds. Automatic Depressurization System (ADS) is

activated when wide range (WR) level measurement stays below LEVEL 1 (3.89 m above the top of active fuel) continuously for more than 10 seconds. GDCS line opens 150 seconds later. These events occur at 566 and 716 seconds respectively in this calculation. Should the water level fall further (1.0 m above the top of active fuel), the GDCS equalization line would open and deliver water from the suppression pool; however, it did not occur in the time period of our calculations (20,000 seconds).

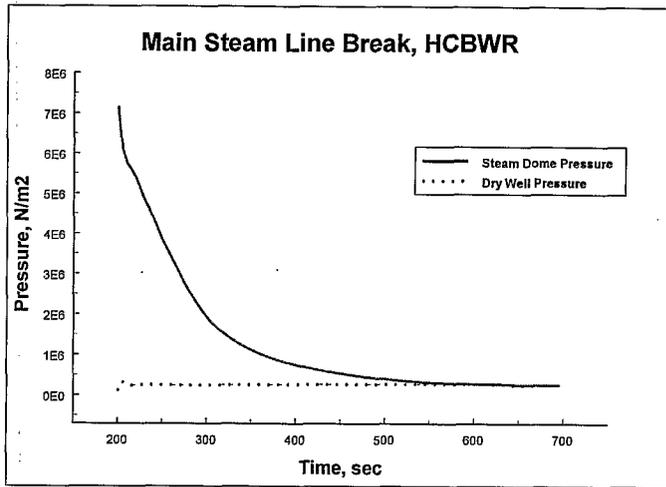
Immediately following the break (i.e., 200 seconds in the figures), the RPV pressure decreases rapidly as shown in Figure 7. The containment drywell (DW) pressure, shown in Figure 8, increases rapidly and reaches its short-term peak pressure of about 3.3 bars in about 10 seconds. These two pressures come to an equilibrium in about 600 seconds (Figure 7). The break flow increases to its peak flow rate of 3000 kg/sec immediately after the break, as shown in Figure 9. This includes the break flow from the turbine side, which is fed from the intact steam line through a header upstream of the turbine. The flow from the turbine side stops in 5 seconds as both MSIVs are closed. From that point on, the break flow continues to decrease as the reactor pressure declines. Figure 10 shows the wide-range (WR) water level in the RPV. The WR water level instrument essentially measures the collapsed water level above Elevation 8.27 m (2.78 m above the active fuel) by measuring the differential pressure between the steam dome and Elevation 8.27 m. The water level rapidly decreases as the RPV loses water through the break. When WR water level reaches LEVEL 1 at 566 second, ADS is activated and subsequently GDCS valves open in 150 seconds. As the GDCS water starts to flow in at 716 second, the RPV water level begins to rise. Figure 11 shows the Fuel Zone Range Level, which is the collapsed water level measured from the bottom of the core. It shows that the collapsed water level stays above the top of active fuel during all of this period and, thus, the core is never uncovered during the time period analyzed in this transient. This is confirmed by Figure 12 which shows the Peak Clad Temperature (PCT), the fuel temperature at the hottest point of the hot channel. This figure also shows that the fuel cools rapidly as the blowdown begins and the reactor is scrammed. It should be also noted that the actual mixture level would be even higher than the collapsed water level. Figure 13 shows the water flow rate to the RPV from all 3 GDCS tanks. The GDCS valves open at 716 second and the water begins to flow immediately. The GDCS flow decreases as the water level in the RPV increases and water level in the GDCS tanks decreases as shown in Figure 14.

At the initiation of break, a large amount of steam starts to blow down to the drywell (DW) from the vessel, causing a steep increase of containment pressure. Figures 8 and 15 show the pressure in the DW of the containment. Immediately after the break, the DW pressure starts to steeply rise; momentarily, it increases to about 3.3 bars. As more steam and non-condensable gas are moved to the suppression chamber (SC), the DW pressure levels off at about 2.5 bars and, then, slowly increases as more energy is shifted from the vessel to the containment, until GDCS injection begins at about 716 second. The containment pressure begins to decrease once the GDCS water starts to flow to the RPV as the subcooled GDCS water absorbs the decay heat and stops steam generation. The GDCS flow is gradually reduced and, consequently, steam production in the RPV begins to increase. From this point on, the containment pressure begins to rise slowly because the remaining non-condensable gas in the lower drywell (LDW) and GDCS tanks

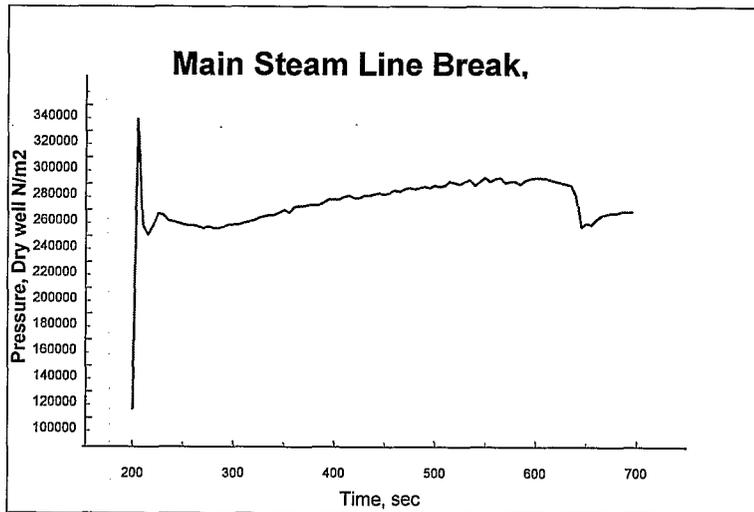
gradually moves to the suppression chamber (SC) and the SP surface water temperature also increases slowly, as some steam continues to condense in the suppression pool.

Figure 16 shows the heat removal rate from PCCS and ICS. The decay heat is also shown in the figure for comparison. As blowdown begins and steam fills the DW, PCCS begins to condense steam and remove heat from the containment at the rate of about 30 MW. The heat removal rate remains at this level until GDCS injection begins. As GDCS water begins to flow into the RPV, the rate of steam production decreases and the PCCS heat removal rate decreases accordingly. PCCS resumes the heat removal when steam generation in the RPV resumes at about 10,000 second. A combination of ICS and PCCS is able to dissipate most of the decay heat. Any decay heat, that is not dissipated, contributes to the gradual rise of the containment pressure (Figure 15).

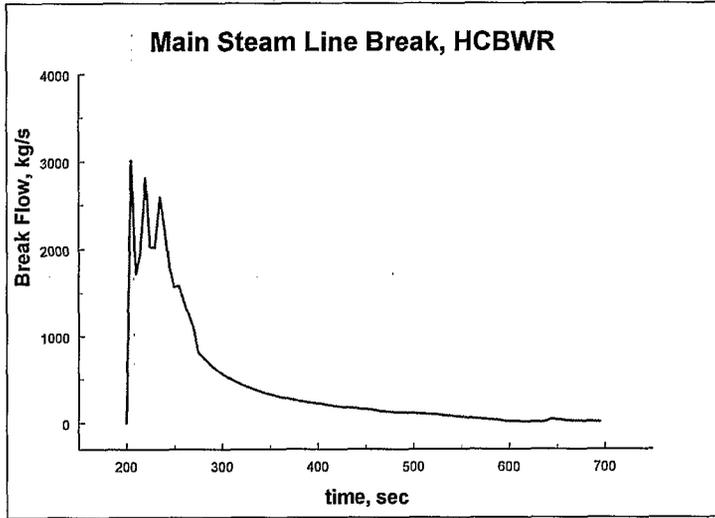
In summary, the core was never uncovered and the clad temperature remained subcooled all the time during the MSLB transient. As the water level in the vessel decreased initially, the GDCS flow was initiated and the vessel was filled with water, as designed. The PCCS removed a substantial fraction of the decay heat. The amount of heat removed by PCCS and ICS was initially less than the decay heat. This caused the gradual increase of the containment pressure. The containment pressure reached its peak pressure of 3.5 bars around 12000 seconds and remained at this pressure during the remainder of the transient as the combined heat removal rate by ICS and PCCS matched the decay heat generation rate.



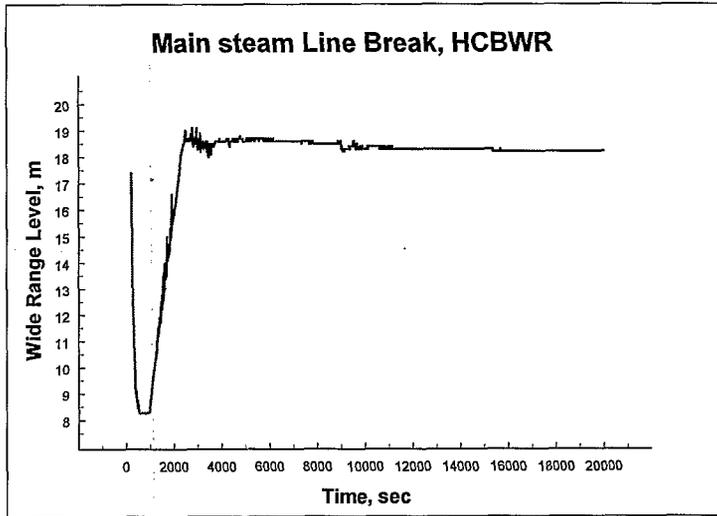
**Figure 7 Vessel and Containment Pressure, short term**



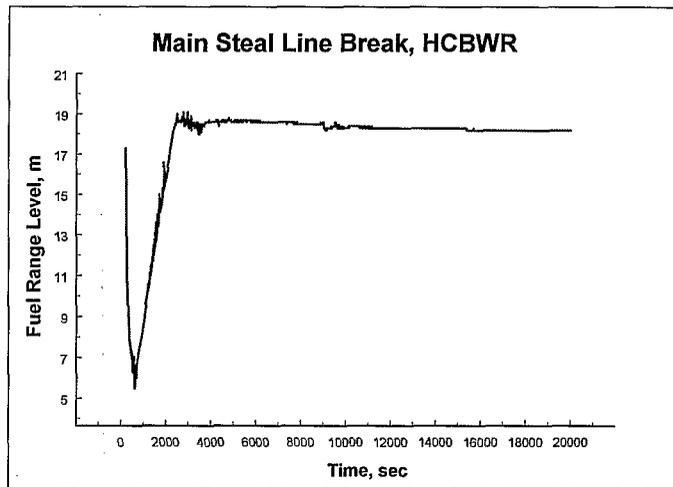
**Figure 8 Containment Pressure, short term**



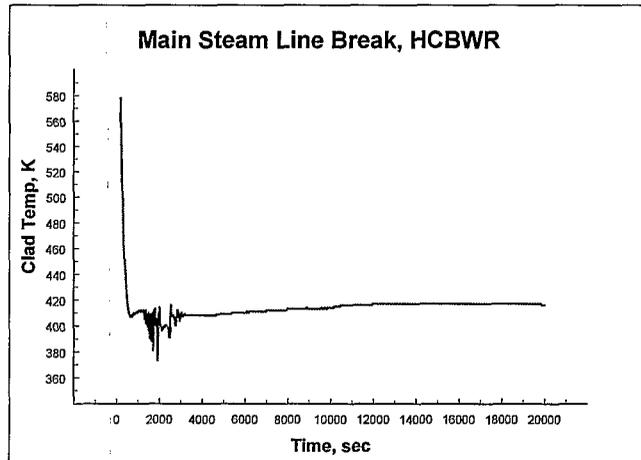
**Figure 9 Break Flow from the steam line break**



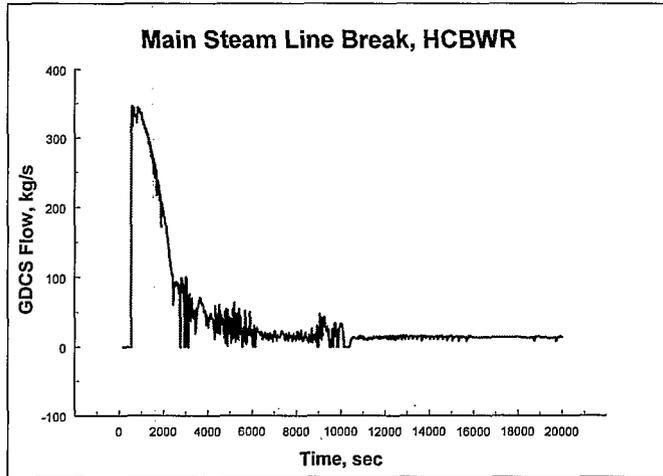
**Figure 10 Wide Range Level Prediction**



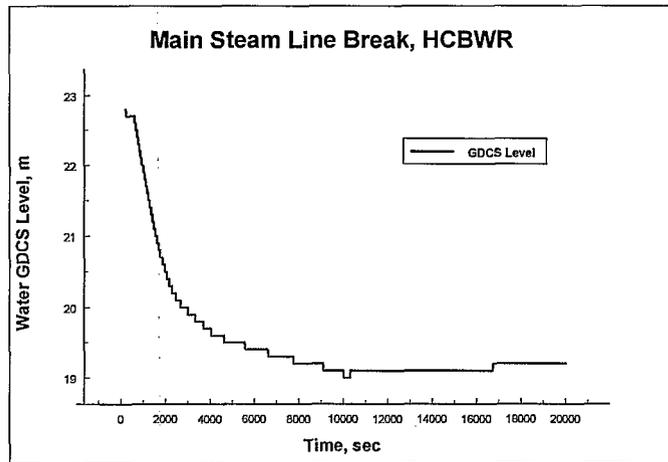
**Figure 11 Fuel Range Level Prediction**



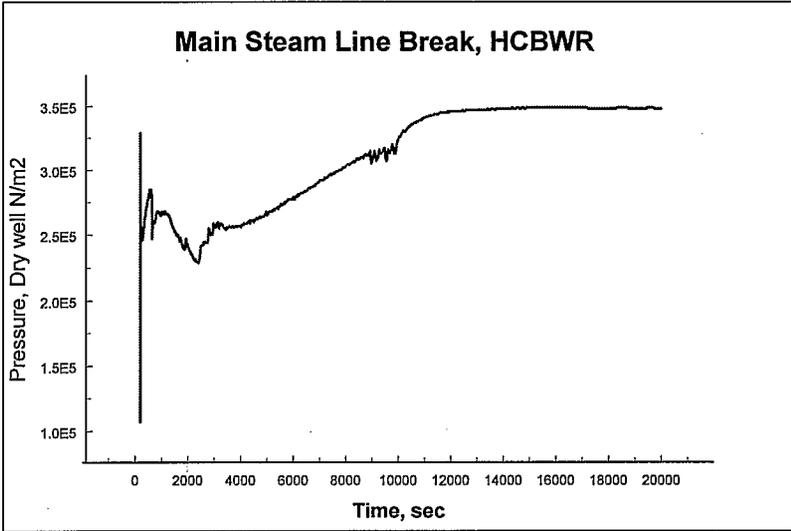
**Figure 12 Peak Clad Temperature**



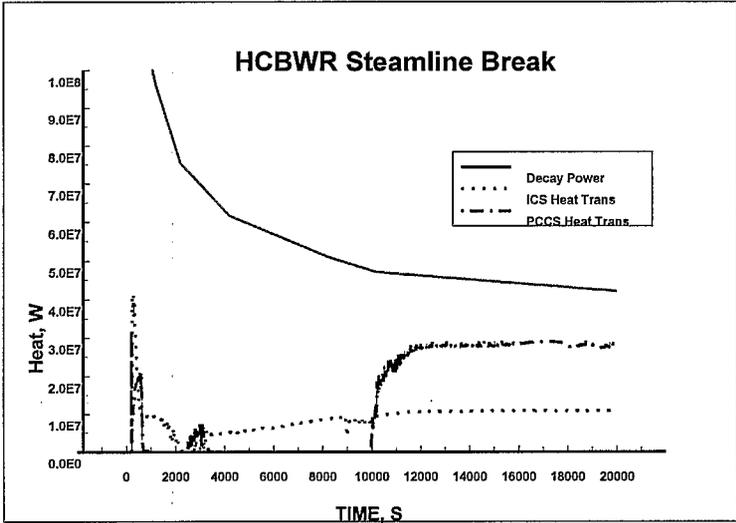
**Figure 13 Flow Rate from Gravity Driven Cooling System**



**Figure 14 GDCS Tank Water Level**



**Figure 15 Drywell Pressure for long term**



**Figure 16 Decay Power and Heat Removal**

## 4. Conclusion

The tight lattice BWR design has been analyzed for two important transients; steam line break and station black out. The core was assumed to be scrammed. The conclusion from the analyses is that the clad temperature decreases from the steady state conditions and is close to the saturation temperature corresponding to the pressure in the primary side. In both the transients the clad temperature for the hot rod was below the 440K.

The containment pressure for the steam line break has an early peak of 3.3 bar. The pressure stabilizes to 3.5 bars for the long term. The containment pressure remains below the safety limit. The containment considered here is same size as for SBWR (2000 MWt) and the containment pressure is only slightly higher for HCBWR with twice the power. The containment pressure is controlled by the pressure of the noncondensibles as they move to the gas space above the suppression pool. The water temperature in the suppression pool is higher and therefore, the partial pressure of the vapor in the gas space is higher. But the sum of these two pressures is still below the safety limit for the containment. A smaller containment is sufficient for HCBWR.

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## **Nomenclature**

**ABWR – Advanced Boiling Water Reactor**

**ADS – Automatic Depressurization System**

**DW – Drywell**

**ECCS – Emergency Core Cooling System**

**GDCS – Gravity Driven Cooling System**

**GE – General Electric**

**HCBWR – High Conversion, Boiling Water Reactor**

**ICS – Isolation Condenser System**

**LDW – Lower Drywell**

**LOCA – Loss of Coolant Accidents**

**MSIV – Main Steam Isolation Valve**

**MSLB – Main Steam Line Break**

**PCCS – Passive Containment Cooling System**

**PCT – Peak Clad Temperature**

**RPV – Reactor Pressure Vessel**

**SBO – Station Blackout**

**SBWR – Simplified Boiling Water Reactor**

**SC – Suppression Chamber**

**SP – Suppression Pool**

**WR – Wide Range**

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