Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

September 7, 2010
U7-C-STP-NRC-100199

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Rockville, MD 20852-2738

South Texas Project
Units 3 and 4
Docket No. PROJ0772
Responses to Request for Additional Information

Reference: Letter from Tekia Govan to Mark McBurnett, "Request for Additional Information Re: South Texas Project Nuclear Operating Company Topical Report (TR) WCAP-17116-P Revision 0, Supplement 5 – Application to the Advanced Boiling Water Reactor" (TAC No. RG0012), June 7, 2010 (ML101580249)

Attached are the 90-day responses to NRC staff questions included in the reference. The responses to the following RAI questions are provided:

RAI-2	
RAI-3	RAI-12
RAI-8	RAI-16
RAI-11	RAI-20

The responses to RAI questions 22 and 32 contain proprietary information and will be transmitted separately no later than September 13, 2010.

There are no commitments in this letter.

If you have any questions, please contact Scott Head at (361) 972-7136, or Bill Mookhoek at (361) 972-7274.

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NRO

STI 32732442

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 9/7/2010

Mark McBurnett
Vice President, Oversight & Regulatory Affairs
South Texas Project Units 3 & 4

jet

Attachments:

1. RAI-2
2. RAI-3
3. RAI-8
4. RAI-11
5. RAI-12
6. RAI-16
7. RAI-20

cc: w/o attachment except*
(paper copy)

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RAI-2

QUESTION:

In Section 4.4.2 of WCAP-17116, fast steam line isolation is stated to be a conservative assumption because it reduces voiding in the core at the start of accident. Would this apply to feedwater as well, noting that figures-of-merit such as PCT in ABWR appear to be a function of very early formation and collapse of voids in the reactor coolant rather than long-term coolant inventory makeup? Demonstrate that the assumption of fast (1 s) feedwater coast down is conservative from the standpoint of PCT.

RESPONSE:

To investigate the effect of feedwater coastdown time on the calculated Peak Clad Temperature (PCT), additional GOBLIN runs were performed with feedwater coastdown times of 0.01, 10 and 20 seconds. These calculations showed no difference in the calculated PCT for hot or average channels when varying the assumed feedwater coastdown time. Because the PCT occurs so early in the accident sequence, continued flow from feedwater pumps is unlikely to have a significant effect on the PCT. Figure 2-1, below, shows clad temperature for the limiting elevation of the hot channel as a function of feedwater coastdown time. Note that the curves for the four coastdown times are superimposed on top of each other.

These calculations conclude that feedwater coastdown times that are longer than the fast (1s) coast down time do not have an effect on the PCT.

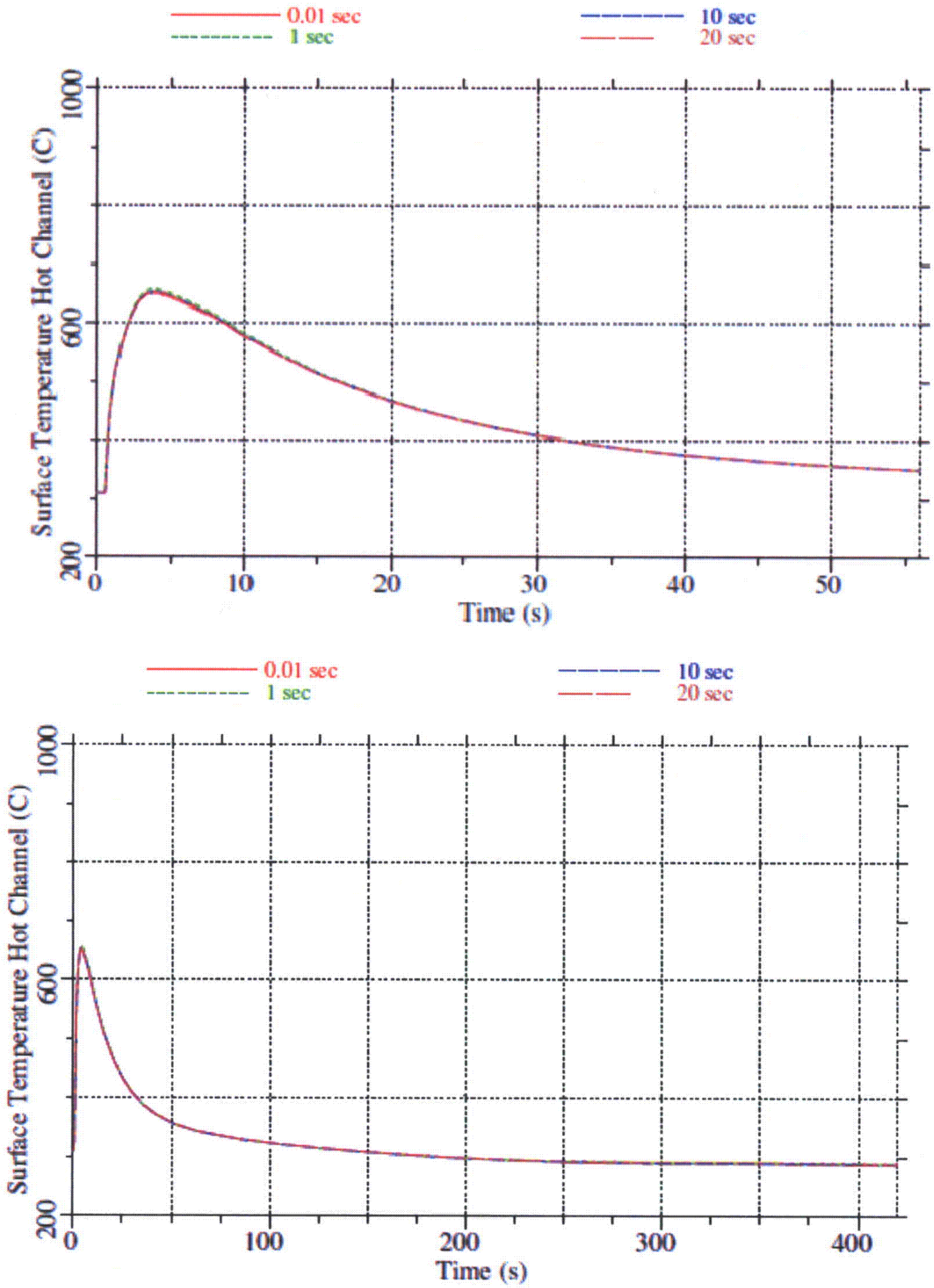

Figure 2-1 – PCT for Different Feedwater Coast Down Times

RAI-3

QUESTION:

Section 4.5.1.2 of WCAP-17116 (HPCF break spectrum sensitivity cases) describes how prediction of core uncovery is more likely in low-power assemblies than in high-power ones due to swelling of the two-phase mixture. In view of this, it is not intuitively obvious a priori which channel would be the most conservative one to consider (highest-power, lowest-power, or somewhere in between).

a) Can you demonstrate that no assembly would experience PCT significantly higher than the ones documented in the sensitivity analysis (i.e., would explicit modeling of an intermediate lower-power assembly yield more conservative results)?

b) What is the procedure for the identification of hottest channel?

RESPONSE:

a) The bundle power peaking factor values for the High Pressure Core Flooder (HPCF) break bundle power sensitivity study of 0.3, 0.6, 0.9, 1.2, 1.5 and 1.7 are expanded to cover peaking factors ranging from 0.1 to 1.7, and a peaking factor resolution of 0.1 is used.

   For each power level, the GOBLIN maximum cladding temperature during the core partial uncovery period after the automatic depressurization system (ADS) is actuated, is shown in Figure 3-1. The GOBLIN peak cladding temperature (PCT) due to the initial dryout is also included for comparison purposes.

   The results of the expanded study, as shown in Figure 3-1, show that the maximum cladding temperature during the partial uncovery is well below the PCT during the initial early dryout time, and the maximum cladding temperature is also a well behaved function of bundle power. Figure 3-1 also shows that the maximum cladding temperatures for these additional assumed bundle peaking factors are not significantly higher than the ones documented in the sensitivity analysis in WCAP-17116-P. Therefore, explicit modeling of lower power assemblies is not warranted.

b) Because the maximum cladding temperature during the partial core uncovery is well below the PCT that occurs during the initial dryout due to the pump coastdown, the hottest channel is the one with the channel power described in Section 4.3.1.2 of WCAP-17116-P.

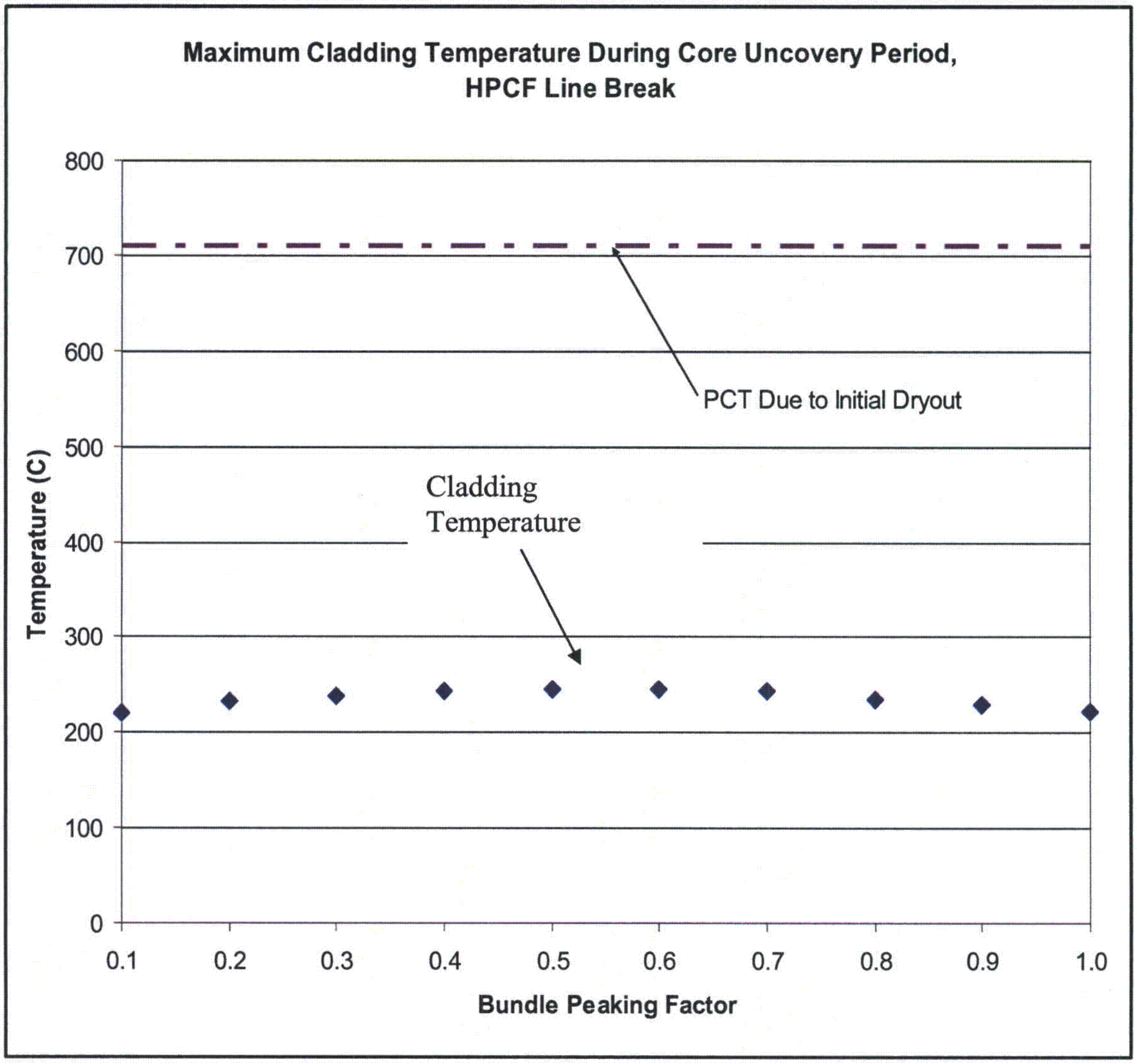

Figure 3-1 Variation of Maximum Cladding Temperature During Partial Core Uncovery with Respect to Bundle Peaking Factor

RAI-8

QUESTION:

I.C.1.b of Appendix K requires that the licensee perform calculations with a range of break discharge coefficients ranging from 0.6 to 1.0. Clarify the evaluation model's compliance with this requirement, given that no values of break discharge coefficient are mentioned for the analyses presented in WCAP-17116-P.

RESPONSE:

The evaluation model, which is the basis for WCAP-17116-P, meets the above Appendix K, I.C.1.b requirement. The discharge coefficient is identified in WCAP-17116-P as "break size." A break size of 100% represents a fully open pipe break (discharge coefficient of 1.0). A break size of 50% would represent a break size of half the pipe cross-sectional area (discharge coefficient of 0.5), etc. Below is a summary of places in WCAP-17116-P that supply this information. In all cases, the range of discharge coefficients of 0.6 to 1.0, as noted in the RAI, is covered in the analyses.

- Table 4-2 in Section 4.5.1.1 shows the range of break sizes that were examined for a HPCF line break. Section 4.5.1.2 has a discussion of the effects of different break sizes.

- Table 4-3 in Section 4.5.2.1 shows the range of break sizes that were examined for a main steam line break. Section 4.5.2.2 has a discussion of the effects of different break sizes. Note that Table 4-3 shows break sizes of 200%, 150% and 100%. Because GOBLIN models the loss of reactor coolant from both sides of this break, a complete pipe break would result in a 200% break size case. As a result, a 200% main steam line break is analogous to a 100% single-sided break, a 150% break is analogous to a 75% single-sided break, and a break of 100% would be analogous to a 50% single-sided break. Therefore, the Appendix K range of 0.6 to 1.0 is covered in this case.

- Table 4-4 in Section 4.5.3.1 shows the range of break sizes that were examined for a feedwater line break. Section 4.5.3.2 has a discussion of the effects of different break sizes.

- Table 4-5 in Section 4.5.4.1 shows the range of break sizes that were examined for a Residual Heat Removal (RHR) suction line break. Section 4.5.4.2 has a discussion of the effects of different break sizes.

- Section 4.7, the summary of limiting cases, includes plots of minimum inventory and Peak Clad Temperature (PCT) as a function of break size.

RAI-11

QUESTION:

10CFR50.46 requires that "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss of coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." A failure of the RIP vertical restraints can result in the loss of reactor coolant from the bottom of the RPV. ABWR DCD states that the break flow due to RIP failure would be bounded by the break flow in BDLB accident scenario. However, neither ABWR DCD nor WCAP-17116-P provides any justification to support the above statement.

a) Discuss the transient response to a RIP failure LOCA.

OR

b) Provide justification to support the ABWR DCD assumption that the break flow due to the RIP failure would be bounded by the break flow in BDLB accident scenario. What is the equivalent break flow area for the RIP failure?

RESPONSE:

A detailed justification, which supports the assumption that the break flow due to Reactor Internal Pump (RIP) failure is bounded by the design basis Bottom Drain Line Break (BDLB), is provided in ABWR DCD Tier 2 Subsection 15B.3.4.

The break area for the RIP failure, which is equivalent to a gap between the stretch tube and the RIP nozzle (See Figure 11-1), is less than the design basis BDLB area (20 cm^2).

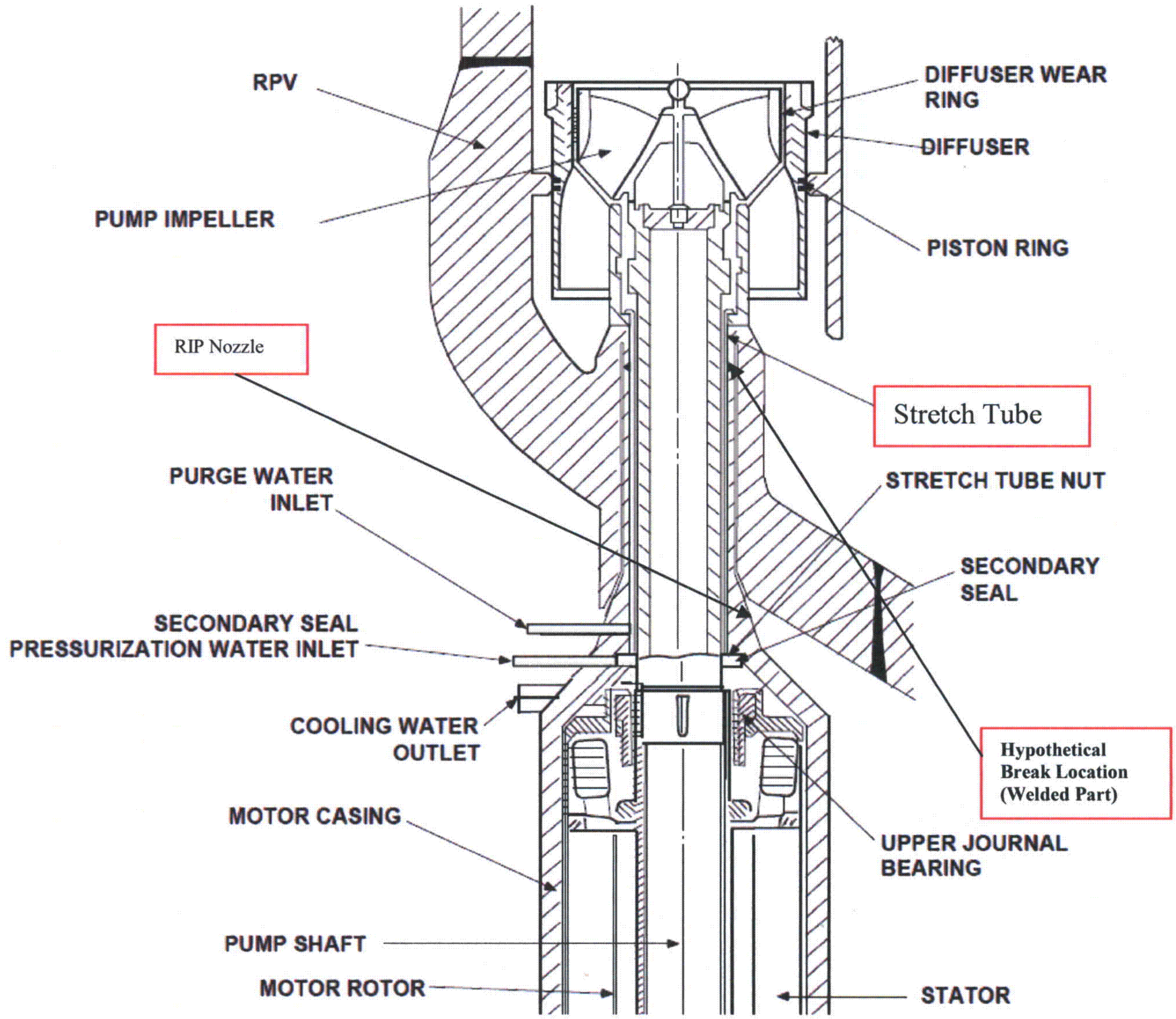

Figure 11-1 Break area for the RIP failure

RAI-12

QUESTION:

As noted in ABWR DCD, "Recirculation Motor Cooling (RMC) is continuously provided by purge water supplied to the motor internals from the Control Rod Drive System (CRDS). A motor internal auxiliary impeller drives the purge water upward through the stator and rotor windings then exits the motor housing or leaks into the reactor vessel along the pump shaft."

Is there potential for a LOCA from the bottom of the Reactor Pressure Vessel (RPV) through the RIPs due to the loss of Recirculation Motor Cooling? If yes, please describe the magnitude of leakage and the resulting LOCA response for ABWR.

RESPONSE:

As noted in ABWR DCD Tier 2 Subsection 5.4.1.3, the reactor recirculation system includes, among other support subsystems, the following two separate support subsystems: (1) the Recirculation Motor Cooling (RMC) subsystem and (2) the Recirculation Motor Purge (RMP) subsystem.

The purpose of the RMC subsystem is to remove heat from the recirculation pump motor (RM), which is generated by the RM internals and is also conducted from the reactor pressure vessel (RPV) to the RM. The Reactor Internal Pump (RIP) motor casing and its heat exchanger, which is part of the RMC subsystem, are filled by the Makeup Water (Purified) System (MUWP) water before operation. Cooling for the heat exchanger is provided by the Reactor Building Cooling Water System. The potential for a LOCA as a result of rupture of the motor cooling piping, including a description of the magnitude of leakage and the resulting response, is discussed in ABWR DCD Tier 2, Subsection 15B.3.4.7. In the event of a loss of RIP motor cooling due to a small size LOCA, the RIPs trip on high temperature of primary cooling water.

The purpose of the RMP subsystem is to prevent the buildup of primary coolant impurities on RM components. The RMP supplies a flow of clean water from the CRD system to the RM shaft stretch tube annular region, which is located just above the RM upper journal bearing. This is intended to minimize radiation dose for RIP maintenance activities. The potential for a small size LOCA as a result of rupture of the purge line, including a description of the magnitude of leakage and the resulting response, is discussed in ABWR DCD Tier 2, Subsection 15B.3.4.5.

RAI-16

QUESTION:

Section I.A.6 of Appendix K to 10 CFR50 requires that "heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account."

a) Since the steam dome volume acts as a dead volume prior to ADS actuation, metal heat release to steam could increase the steam enthalpy above saturation. When ADS activates, the rate of system depressurization could be affected by the amount of energy removal. Describe the metal heat release to the dead volume and estimate the amount of heat released to steam prior to ADS actuation and the potential for superheated steam to be vented through the ADS.

b) Is the volume and metal heat of the piping to the point where it is isolated from the reactor coolant system accounted in the ABWR ECCS Evaluation Model?

RESPONSE:

a) Table 16-1 shows the heat transfer between the metal of the steam dome and the steam immediately before the Automatic Depressurization System (ADS) actuates for each of three breaks. Figures 16-1 through 16-3 include the same data, along with the time when the ADS valves open. The steam flowing to the ADS valves becomes superheated in all cases described below except for the feedwater line break.

   The ADS most affects the high pressure core flooder (HPCF) line break (HPCF3) transient because the system recovery is delayed until the low pressure core flooder (LPFL) pumps start. In this case, note in Table 16-1 that the heat transferred to the coolant in the steam dome is in the negative direction (from the steam to the metal) prior to ADS actuation. The initial temperature of the structures in the steam dome is based on normal operating conditions. After the break occurs, the steam line is isolated by fast closure of the turbine control valves. This results in a pressurization of the system, and the system pressure is controlled by the opening and closing of the safety / relief valves (SRVs). Because the steam dome pressure is higher than the initial operating pressure during this time, the steam temperature will be higher than the temperature of the metal structures. As a result, heat is transferred from the steam in the steam dome to the structures (e.g., the vessel dome and the steam dryer) as shown in Figure 16-3. After ADS actuates, the system pressure (and temperature) decreases and the direction of heat transfer reverses.

Table 16-1 – Integrated Heat Transfer From Metal Components in the Steam Dome Prior to ADS Actuation

	Steam Dome (MJ)	Dryers (MJ)	Total (MJ)
FWLB3	1.8	12.3	14.1
MSLB6	37.6	21.8	59.4
HPCF3	-190.8	-3.9	-194.7

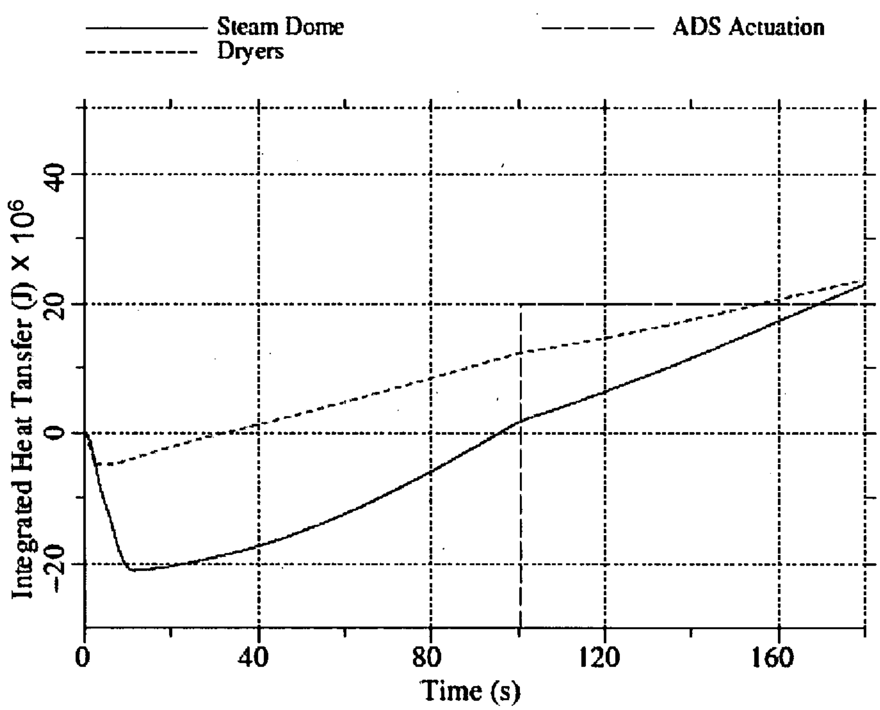

Figure 16-1 Integrated Heat Transfer from Steam Dome Structures to Coolant for FWLB3 case

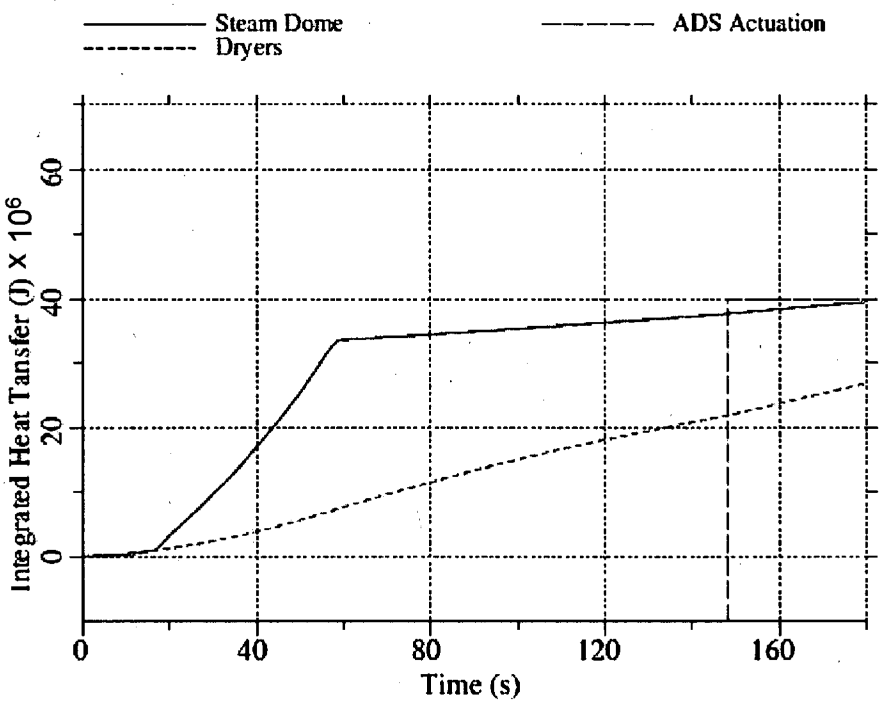

Figure 16-2 Integrated Heat Transfer from Steam Dome Structures to Coolant for MSLB6 case

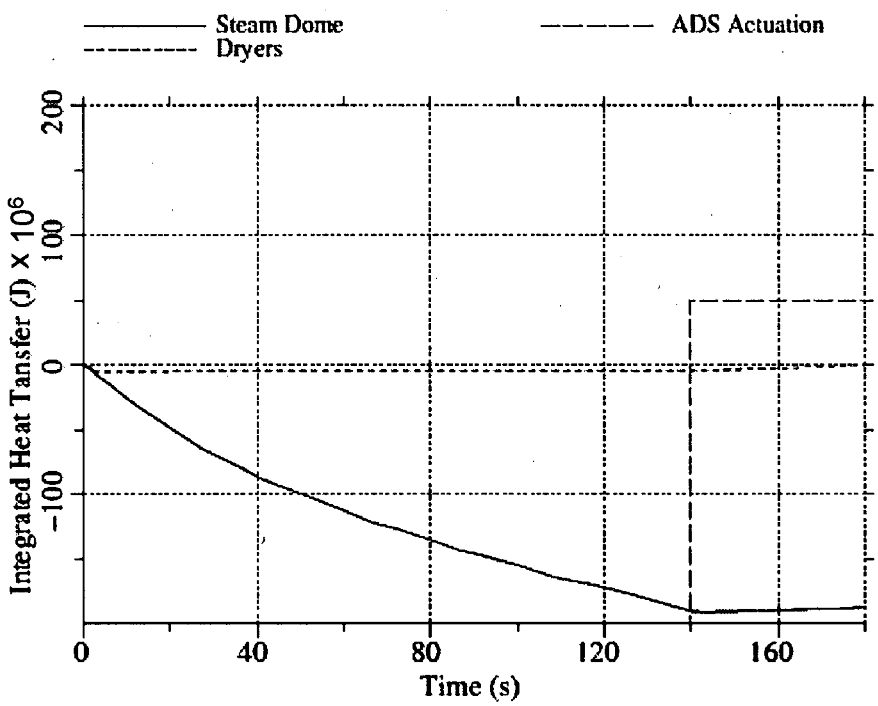

Figure 16-3 Integrated Heat Transfer from Steam Dome Structures to Coolant for HPCF3 case

b) All metal structures in the reactor pressure vessel are included in the GOBLIN heat structure model. The heat contained in the metal of the piping outside the reactor pressure vessel (i.e., the steam lines) is not modeled in GOBLIN. Because the section of the steam lines upstream of the Main Steam Isolation Valves (MSIVs) is quite short, it is assumed that the heat contained in the reactor internals would be far greater than any contained in the steam lines.

RAI-20

QUESTION:

As indicated in Section 5.3 of WCAP-17116, the CPR correlation D4.1.2 (developed for the SVEA-96 Optima2 fuel) is implemented in the GOBLIN code for the ABWR application. It has been stated that the correlation is described in detail in Section 5.0 of WCAP-16081-PA. However, WCAP-16081-P-A provides discussion on CPR correlation D4.1.1.

a) Provide explanation of any changes/modifications made to the D4.1.1 CPR correlation to arrive at the D4.1.2 correlation.

b) The NRC SER documented in WCAP-16081-P-A recommends using different uncertainties for the CPR correlation based on the system pressure (3.15% below 45 bars and 2.32% above 45 bars). Please explain whether and how these uncertainties have been incorporated into the D4.1.2 correlation. Also provide an explanation as to whether the use of system pressure based sensitivities is carried over to the GOBLIN analysis. If not, please provide a justification for not doing so.

RESPONSE:

a) The GOBLIN code currently uses the D4.1.2 CPR correlation. An earlier version of GOBLIN used the D4.1.1 CPR correlation. The D4.1.2 includes correction factors due to the sub-bundle to full-bundle effect, the double-peaked axial power profile correction, and the R-factor correction. These three items are described in Section 5 of WCAP-16081-P-A. Otherwise, the D.4.1.1 correlation described in WCAP-16081-P-A is the same as the D4.1.2 correlation.

b) The GOBLIN code does not specifically account for variations in uncertainty. However, as described in Reference 20.1 below, the GOBLIN code uses two correlations for determining the critical power ratio: one for the flow boiling regime and one for the pool boiling regime. The code computes the critical heat flux using both correlations and uses the maximum of the two to determine the onset of dryout.

   System pressure-based sensitivities are not carried over to the GOBLIN analysis, nor are they required. As described in WCAP-17116, the ABWR LOCA transients are characterized by early dryout due to rapid loss of flow before the system pressure has changed significantly. Figure 5-4 of WCAP-17116-P shows that GOBLIN is consistently conservative in its prediction of dryout for loss of flow transients. Therefore, there is no need to perform pressure-based sensitivity analyses.

Reference

20.1 RPB 90-93-P-A, "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," October 1991.