PWROG
PWR Owners Group

**Program Management Office**
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Project Number 694
WCAP-16793-NP, Revision 1

August 9, 2010

OG-10-253

U.S. Nuclear Regulatory Commission
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Subject: PWR Owners Group
**PWROG Response to Request for Additional Information Regarding PWROG Topical Report WCAP-16793-NP, Revision 1, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," (PA-SEE-0312)**

References:

1. "PWR Owners Group Submittal of WCAP-16793-NP, Revision 1, 'Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating' (PA-SEE-0312)," OG-09-163, dated April 22, 2009.

2. NRC Letter, Jonathan Rowley of NRR to Anthony Nowinowski of the PWR Owners Group Program Management Office, "Request for Additional Information RE: Pressurized Water Reactor Owners Group Tropical Report WCAP-16793-NP, Revision 1, 'Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid' (TAC No. ME1234)," January 8, 2010. (ADAMS Accession Number: ML093490855.)

3. "PWROG Response to Request for Additional Information Regarding PWROG Topical Report WCAP-16793-NP, Revision 1, 'Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid,' (PA-SEE-0312)," OG-10-45, dated February 9, 2010.

In April 2009, the Pressurized Water Reactor Owners Group (PWROG) submitted WCAP-16793-NP, Revision 1, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," for review and acceptance for referencing in regulatory actions (Reference 1). In January 2010, NRC staff provided a formal Request for Additional Information (RAI) (Reference 2) for WCAP-16793-NP, Revision 1. In February 2010, the PWROG provided responses to these NRC RAIs (Reference 3).

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The RAI responses related to WCAP-16793-NP, Revision 1, provided in Reference 3, have since been revised to include additional information regarding the calculation of available driving head and utilities operating with various fuel types. Enclosure 1 to this letter provides the revised RAI responses to the questions received in Reference 2

Enclosure includes:

1. One copy of LTR-SEE-I-10-23, Revision 1, "Transmittal of RAI Responses for WCAP-16793-NP, Revision 1," August 2010, (Non-Proprietary)

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski
Manager, PWROG Program Management Office
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1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at 704-382-8619 or Mr. Anthony Nowinowski of the PWROG Program Management Office at 412-374-6855.

Sincerely,

Melvin L. Arey, Jr.

Melvin L. Arey, Jr., Chairman
PWR Owners Group

Enclosure (1)

MLA:KJN:rfn

cc: PWROG Management Committee
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To: Kenneth J. Nemit (PWROG)

Date: August 5, 2010

From: Systems and Equipment Engineering I

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Your ref:

1. WCAP-16793-NP, Revision 1, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," April 2009.
2. "Request for Additional Information RE: Pressurized Water Reactor Owners Group Topical Report WCAP-16793-NP, Revision 1, 'Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid' (TAC No. ME1234)," January 2010. (ADAMS Accession Number: ML093490855)

Our ref: LTR-SEE-I-10-23, Revision 1

Subject: **Transmittal of RAI Responses for WCAP-16793-NP, Revision 1**

The PWROG undertook a program to provide additional analyses, test data, and information on the effect of debris and chemical products on core cooling for pressurized water reactors when the emergency core cooling system is realigned to recirculate coolant from the containment sump. This program is documented in [1]. After the publication of [1] requests for additional information [2] from the NRC were transmitted.

Attachment 1 transmits the responses to [2] related to WCAP-16793-NP, Revision 1 which have been revised to include additional information regarding the calculation of available driving head and utilities operating with various fuel types.

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# Attachment 1: RAI Responses for WCAP-16793-NP, Revision 1

## Table of Contents

# 1.0 Record of Revisions

| Revision | Date | Summary of Changes |
|---|---|---|
| 0 | 2/10/10 | • Original issue. |
| 1 | See EDMS | • All revisions are marked with revision bars.<br>• Added Section 1.0 to identify revision changes.<br>• Section 2.18, RAI 18, updated to include guidance for available driving head calculation.<br>• Added Section 3.0 to provide guidance for utilities operating with multiple fuel types. |

# 2.0 RAI Responses

This document provides responses to requests for additional information [1] related to WCAP-16793-NP, Revision 1. WCAP-16793-NP, Revision 1 will not be revised. However, changes noted here will be incorporated in the approved version of the WCAP after the SER is received.

## 2.1 RAI #1

### 2.1.1 Question

*On page 7-5, the topical report states that a quantitative estimate of the effect of fiber bypassing the sump strainer can be accounted for in LOCADM by use of a bump-up factor. A bump-up factor is applied to the chemical source term since LOCADM does not directly address small fibers that pass through the strainer and transport into the reactor vessel. The wording suggests that use of a bump-up factor is optional. The staff thinks it is appropriate for all plants to calculate a bump-up factor in their plant-specific LOCADM calculations. Please discuss whether the topical report will be revised to provide more definitive guidance related to the use of a bump-up factor for fiber bypass.*

### 2.1.2 Response

The bump-up factor is not optional. All plants are required to calculate a bump-up factor in their plant-specific LOCADM calculations.

The discussion of the bump-up factor is found in Section 7.2.1.3 of [2] which describes the additional steps required to complete a LOCADM calculation. To increase the clarity of the bump-up factor discussion, Section 7.2.1.3 will be changed to read (change highlighted in **bold**):

> 7.2.1.3 Additional Steps
>
> Aluminum Release Rate
>
> In order to provide more appropriate levels of aluminum release for the LOCADM analysis in the initial days following a LOCA, licensees shall apply a factor of two to the aluminum release. The recommended procedure for modifying the aluminum release rate is described in Reference 7-5.
>
> Bump-Up Factor
>
> LOCADM does not contain an input for debris which bypasses the sump screen and is available for deposition in the core. Only material released from corrosion or dissolution processes is considered. However, some debris fines may bypass the sump screen and enter the core area where it could be deposited. A quantitative estimate of the effect of the fiber on deposit thickness and fuel temperature ~~can~~ **must** be accounted for in LOCADM by use of a "bump-up factor" applied to the initial debris inputs. The bump-up factor is set such that total release of chemical products after 30 days is increased by the best estimate of the mass of the fiber that bypasses the sump screen. This allows the bypassed material to be deposited in the same manner as a chemical reaction product. The recommended procedure for including fiber bypass in the LOCADM deposition calculations is illustrated in Reference 7-4.

The definitive guidance related to the use of a bump-up factor is provided in [3].

## 2.2 RAI #2

### 2.2.1 Question

*On page xx and Section 2.2, page 2-1, the acceptance criterion indicates the total deposit including oxide should not exceed an average of 0.050 inches in any fuel region. Please provide details concerning how an average total deposit thickness is determined and also define a fuel region. Further, in Section 2.2, Subparagraph 2, page 2-1, it is stated that 50-mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur. Therefore, it would appear that the acceptance criterion should be stated as a maximum of 0.050 inches. Please change the criterion given this information or justify the current criterion.*

### 2.2.2 Response

A detailed discussion concerning the average total deposit thickness determination and the definition of a fuel region is found in Appendix E of [2]. In summary, the deposition model divides the core into user defined nodes that differ in location and relative decay power. The node is identified by region number and by axial location number. The number of regions is dependent upon the plant design. Each region has a relative power and the weighted average of all relative powers must be 1.0. The weighting is done by number of rods in each region. The deposition predicted to occur in the core is distributed among the modeled core nodes according to the calculated total decay power for each node.

As stated in Section 2.2, subparagraph 2, page 2-1 and in Section 2.3, the deposition of debris and/or chemical precipitates will not exceed 50 mils on any fuel rod.

The acceptance criteria defined on page xx will be reworded to state (changes are highlighted in **bold**):

> The following acceptance bases were selected for the evaluation of the topical areas identified above:
>
> 1. The maximum clad temperature shall not exceed 800°F.
>
> 2. The thickness of the cladding oxide and the fuel deposits **shall** not exceed ~~***an average of***~~ 0.050 inches in any fuel region.

## 2.3 RAI #3

### 2.3.1 Question

*In Appendix E, Section E.4, page E-4, the LOCADM default deposit density discussed is shown in units of lbm/ft². Please confirm if the LOCADM default deposit density value should be 35 lbm/ft³.*

### 2.3.2 Response

Confirmed. The default calcium deposit density should be 35 $lb_{m\text{-}Ca}/ft^3$.

## 2.4 RAI #4

### 2.4.1 Question

*On page 3-5, Section 3.2.1, the WCAP describes the WCOBRA/TRAC evaluation used to model effect of blockage at the core inlet and makes reference to a dimensionless friction factor ($C_D$). Please define $C_D$, as used in your analysis since many possible definitions exist in the literature. Also, the text states that a $C_D$ of 109 was used in WCOBRA/TRAC to model blockage. Please verify that this is not a typographical error.*

### 2.4.2 Response

The dimensionless loss coefficient ($C_D$) in WCOBRA/TRAC is designed to model local pressure losses in the vertical flow due to local obstructions in the flow field. The pressure loss is modeled in the code as a velocity head loss.

$$\Delta P = C_D \rho \frac{V^2}{2g_c}$$

The WCAP text which states a $C_D$ of 109 was used is a typo; $C_D$ should be $10^9$. Section 3.2.1 will be updated to read (changes highlighted in **bold**):

> The effects of blockage at the core inlet were simulated by ramping the dimensionless friction factor ($C_D$) at the core inlet to a large number, simulating a postulated debris buildup that results in a reduction of flow. A modified version of WC/T was created to allow the friction factor at the core inlet to be ramped. Code simulations were performed using standard input for a problem time of 20 minutes. The 20 minute time was taken to be representative of the earliest time of realignment of the ECCS to operate in the recirculation mode. Starting at 20 minutes, the friction factor at the core inlet was ramped to its terminal value over the next 30 seconds. The core inlet flow blockage occurring in 30 seconds from the start of recirculation is not physical and does not represent any plant condition. The postulated core blockage was modeled in this manner to perform a bounding calculation. After the core inlet resistance was ramped to its terminal value of about $C_D$ = $\mathbf{10^9}$ (which essentially eliminates all flow through the path), the code simulations were run out to 40 minutes to show the flow rate supplied to the core would be sufficient to remove decay heat and maintain a coolable core geometry.

## 2.5 RAI #5

### 2.5.1 Question

*On page 3-2, paragraph 3.1.1, and Appendix B, page B-11, the hand calculation of the pressure drop equation for flow around the system is given as follows:*

$$\Delta P_{flow} = \frac{k}{A^2} \frac{\omega^2}{288 \cdot \rho_g \cdot g_c}$$

*The pressure drop due to flow ($\Delta P_{flow}$) should account for two-phase flow in the core. NRC calculations indicate that during post-Loss of Coolant Accident recirculation two-phase flow exists in the core. The inclusion of a two-phase pressure drop will also affect the value of the available head in:*

$$\Delta P_{avail} = \Delta P_{dz} - \Delta P_{flow}$$

*Please confirm that the hand calculation of the system pressure drop also includes two-phase flow effects in the hydrostatic head ($\Delta P_{dz}$) or justify not doing so.*

### 2.5.2 Response

The Darcy equation is a well established relationship for calculating pressure drop as a function of geometry (unrecoverable losses and flow area) and flow conditions (flow rate and fluid density). This equation is used in Section 3.1.1 of [2] to determine the available driving head without debris following a cold leg break. It is also used in the Appendix B WC/T analyses to determine the appropriate core exit pressure. Since the WC/T analyses were done for dry containments, the Darcy equation was also used to extend the WC/T analyses results to sub-atmospheric containment pressures.

Additionally, the Darcy equation is used to determine plant-specific available driving head (for both hot- and cold-leg breaks). The PWROG provides a methodology in Section 2.18 of this attachment, which utilities can use to define the plant-specific available driving head. This methodology also uses the Darcy equation.

In all cases, the flow losses in the core are neglected.

This approach is justified by the following example calculation.

At the time of sump switchover following a cold leg break LOCA, the core will be covered with a saturated mixture of liquid and steam. The core void fraction is dependent of the initial core power and the core power shape, but will generally be 50% or larger and will decrease with time. The density of this two-phase mixture will generally approach the density of saturated liquid. However, for demonstration purposes, the limiting situation for determining the maximum pressure drop through the core is to assume saturated steam only. Since the steam density decreases with pressure, the saturated steam density of 0.038 lbm/ft$^3$ at 15 psia will be used.

The form-loss coefficient in the core and the core flow area are a function of the fuel and spacer grid design. The form-loss coefficients associated with all of the intermediate spacer grids, upper nozzle of the fuel assembly, and the upper core support plate will sum to a value generally less than 20. Since a larger form-loss coefficient will increase the pressure drop, a value of 20 will be used. The core area generally ranges from 50 to 90 ft$^2$. Since a smaller area will increase the pressure drop, a value of 50 ft$^2$ will be used.

The core flow rate is dependent on the core power level and the time after the LOCA. For a cold leg break, the flow rate will be highest at the time of sump switchover since the core decay heat power will be the highest. The core boiloff rate will generally be less than 70 lbm/s. Since a higher flow rate will increase the pressure drop, a value of 70 lbm/s will be used.

Using the above inputs, the pressure drop due to steam in the core is calculated using the Darcy equation to be

$$\Delta P = \frac{20}{(50)^2} \cdot \frac{(70^2)}{288(0.038)(32.2)} = 0.1\,\text{psi}$$

The total available pressure drop used in the fuel assembly testing was >1.5 psi. The above pressure drop is 6 percent of the maximum value. As more reasonable inputs are used (most significantly a fluid density that better represents the core conditions and the actual core form-loss coefficients), the pressure drop will decrease even further. Therefore, neglecting the pressure drop associated with two-phase flow effects in the core is justified.

## 2.6 RAI #6

### 2.6.1 Question

*Appendix B, Section B.5, page B-27 discusses several WCOBRA/TRAC analyses performed to determine the blockage required to block flow to the core. There are two independently analyzed cases which are described as bounding: one in which the inlet flow area was decreased and one in which the $C_D$ was varied. Please justify why these cases are bounding and explain how these assumptions are representative of debris blockage. Please provide the basis for the assumed $C_D$ variations. Since pressure drop across a porous debris bed is approximately proportional to velocity, explain how these analyses relate to a porous medium pressure drop that would characterize a fiber bed. Provide, or make available for staff review, a WCOBRA/TRAC analysis in which the form loss coefficient (which is related to $C_D$) and area are simultaneously varied. The form loss coefficient could be varied as a function of velocity in order to provide a proportional pressure drop relation with velocity.*

### 2.6.2 Response

As stated in the introduction to Section B.5, these WC/T simulations were performed at the request of the ACRS with the purpose of determining the blockage level that would reduce core flow below that necessary to match coolant boil-off. The cases presented are bounding in the sense that a further increase in core inlet pressure drop (via an increase in $C_D$ or a decrease in flow area) would inhibit core flow such that the flow required to make up for the boil-off could no longer be provided. The modeled increase in core inlet pressure drop in the WC/T simulations is considered representative of debris blockage since debris buildup is likely to occur at the core inlet due to its restrictive flow area, which in turn increases the core inlet pressure drop. Consideration of the effects of debris blockage in other areas of the core is not considered in these simulations, and no quantitative amount of debris is represented in these runs, just an upper bound core inlet pressure drop as predicted by WCOBRA/TRAC.

The assumed $C_D$ variations in Section B.5.3 were selected to determine the blockage level which coolant boil-off could no longer be matched. From the previous runs performed in WCAP-16793-NP Revision 0 (described in Section B-3), it was observed that a $C_D$ of $10^9$ would block all flow through the coolant channel. Therefore, starting from a $C_D$ value which was thought to be low enough to allow coolant boil-off to be matched, increases in the uniform loss coefficients were made until a core inlet pressure drop was obtained such that boil-off could no longer be matched.

Finally, it is noted that the original intent of these runs has been satisfied, and further WCOBRA/TRAC runs are not believed to be necessary at this time since an upper bound core inlet pressure drop based on uniform loss coefficients and area reductions has been determined. Please note that a change in the WCOBRA/TRAC form loss coefficient would cause changes in both the fluid velocity and the core inlet pressure drop since the entire RCS system is modeled as an integrated system. For example, the flow at the core inlet is determined by the driving head available in the downcomer. An increase in core pressure drop, either due to an increase in $C_D$ or a decrease in core inlet flow area, would decrease the flow into the core.

## 2.7 RAI #7

### 2.7.1 Question

*In Section 4.1, page 4-1, the report states that "smaller particulate and fibrous debris of the order of 0.04 inch is smaller than the clearance about the "springs" and will readily pass through the grid structure". It can be argued the smaller debris can be filtered by the larger sized debris which has already accumulated in the clearance space. Please provide additional explanation and justification for the statement.*

### 2.7.2 Response

Smaller debris can be filtered by a developed debris bed. The sentence is not necessary for the discussion presented and can be removed. Section 4.1 will be updated to read (changes are highlighted in **bold**):

4.1 GENERAL DISCUSSION
Each FA has a number of spacer grids. These grids are designed to support the fuel rods. Following a LOCA, they provide the most likely location for debris accumulation within the core region. Spacer grid designs commonly used have hard and soft stops, which are small "springs" in the middle of the grids. These "springs" and the leading edge of the grids are the most likely locations for debris to build up, although flow diversion will limit the buildup at this location.

The size of particulate debris that may pass through the replacement sump screens is dependent upon the hole size of the replacement sump screen. This dimension is 0.11 in. or less. The maximum debris size that may be passed by sump screens is of the magnitude of the maximum clearance between fuel rods and grid. **~~Smaller particulate and fibrous debris of the order of 0.04 in. is smaller than the clearance about the "springs" and will readily pass through the grid structures.~~**

The design of a fuel grid allows for cross flow through the grid between adjacent fuel rods. That is, the stops are punched out of the grid such that a flow path exists from one fuel rod to the next near the middle of the spacer grid. This will limit both the extent of the debris build up and its consequences. Should debris collect and form a resistance to the flow of coolant along the fuel rod, both coolant and debris carried by the coolant will be diverted to adjacent "cleaner" locations. A similar phenomenon will occur for fuel designs without hard or soft stops, albeit at the leading edge of the grid. As debris builds up at the leading edge, the flow will divert around it to open channels, limiting the debris build up.

Debris that does collect will have some packing factor that will allow "weeping" flow through debris buildup to cool the cladding. Complete compaction of the debris will not occur and the packing density of the debris is limited to less than unity or perfect compaction. From Reference 4-3, the packing will most likely be less than ~60 percent. Thus, any debris buildup will not become impenetrable. Boiling in the area of the blockage will occur with less than a 10 to 15°F increase in the clad temperature over the adjacent coolant temperature. Even a small amount of fluid flow through the debris bed will provide sufficient heat removal via convection to maintain the fuel rod a few degrees below the liquid saturation temperature.

This general discussion provides solid arguments for asserting that blockages at the spacer grids will not adversely affect LTCC. Additional arguments and analyses are further developed in the following sections.

## 2.8 RAI #8

### 2.8.1 Question

*Section 4.3.1.1, p 4-4 and Appendix C calculate the cladding heat-up due to debris. The report states that a mesh size of 0.05 inches was used for the cladding thermal analysis model. The description of the noding model is incomplete. Please provide the following information:*

*(1) The basis for the mesh sized used for the analysis;*

*(2) The type of analysis performed--steady-state or transient;*

*(3) Any differences in the node size used to model the rod, cladding and debris; and*

*(4) Any variation in the node size along the radius.*

*Justify the mesh size used in the calculation or perform a sensitivity study to justify the mesh sized used in the model. It is noted that Table C-1 in Appendix C provides more details regarding the analysis model, but this information is incomplete. For example, Table C-1 states that the outer clad diameter was 0.36 inch and that the cladding thickness is 0.225 inch. However, the text states that the model was divided into 20 zones. The relationship between the stated node size and the actual dimensions is unclear.*

### 2.8.2 Response

*Note: Table C-1 lists the cladding thickness as 0.0225 inches.*

The mesh size was chosen because it was the smallest size that would run in a reasonable period of time, in this case less than 8 hours. The acceptability of the results is discussed below.

Looking at this closer, the volume of the quarter rod used in the model, which is the largest single component, is calculated as:

$$V = \frac{L}{4} * \left( \pi * \left( \frac{D}{2} \right)^2 - \pi * \left( \frac{d}{2} \right)^2 \right)$$

$$V = \frac{\pi * L}{4} * \left( \frac{D^2}{4} - \frac{d^2}{4} \right)$$

$$V = \frac{\pi * L}{4 * 4} * \left( D^2 - d^2 \right)$$

$$V = \frac{\pi * 144 in}{16} * \left( (0.36 in)^2 - (0.315 in)^2 \right)$$

$$V = 0.859 in^3$$

Even if this value is conservatively doubled (1.718 $in^3$) and rounded up (2.0 $in^3$) to account for the volumes of the grid straps and the debris, this still gives a maximum average element size of $6.89*10^{-5}$ $in^3$ for the models that include debris and $9.17*10^{-5}$ $in^3$ for the clean model. The number of nodes, elements, and average size are summarized in Table 8-1 below for each debris thickness.

**Table 8-1 - Summary of Mesh Data**

| Debris Thickness (mils) | Nodes | Elements | Average Element Size ($in^3$) |
|---|---|---|---|
| 0 | 168121 | 21810 | 9.17E-05 |
| 5 | 227880 | 29603 | 6.76E-05 |
| 10 | 224372 | 29031 | 6.89E-05 |
| 15 | 230347 | 50154 | 3.99E-05 |
| 20 | 228986 | 49334 | 4.05E-05 |
| 25 | 221643 | 45744 | 4.37E-05 |
| 30 | 224855 | 47353 | 4.22E-05 |
| 35 | 225325 | 47700 | 4.19E-05 |
| 40 | 225041 | 47543 | 4.21E-05 |
| 45 | 225433 | 47730 | 4.19E-05 |
| 50 | 225556 | 47795 | 4.18E-05 |

This size is considered to be small enough to give accurate results. This is supported by the final temperature data that matches up very closely with the regression curves, and does not show a wide scatter of data points as would be expected if the model were not significantly detailed.

8-2 – The analysis is a steady state analysis. The heat flux from the center of the rod assembly and the convection coefficient from outside of the assembly were both constant values that were chosen based on the data from COBRA/TRAC. This data showed that the heat input and heat transfer immediately after the simulated accident was conservative due to the increased decay heat when compared to later in the simulation. Because the COBRA/TRAC model uses different values at 84 different fuel rod elevations these values were averaged to give realistic but high results. The model then simulated a period of 720 hours to be consistent with the methodology described in WCAP-16406-P and each time reached an equilibrium temperature before the end of the 720 hour period.

8-3 – There were no components of the model that had a manually modified node or element size. The node size remained consistent throughout the model with the exception of some of the corners where ANSYS automatically generated smaller nodes in order to accurately model the more complex areas of the geometry.

8-4 – As with the different components, there were no sections along the radius of the model that had a manually modified node or element size. The justification of the mesh and element size is explained in the response to part 1 of this question. Table C-1 summarizes the various dimensions used to create the various components of the model, and Table C-2 summarizes the location of the grids on the model. These two tables contain all the information that was used to create the original SolidWorks model that was in turn imported into ANSYS to perform the FEA analysis. The "zones" were defined to aid in verifying the model by defining various sections of the Fuel Rod model as individual "zones." Table C-3 defines the individual zones. For example Zone 1 is the bottom section of the fuel rod, up to the first grid section at 24.570", where there is no debris, and Zone 19 is the section of the rod with the last grid section, 127.270" to 129.520", where there is debris. The actual node and element sizes were created independently of the information in these sections by using a standard fine mesh that was refined by using a reduction factor of 0.05.

## 2.9 RAI #9

### 2.9.1 Question

*Appendix C states that the input values for fluid temperatures, heat transfer coefficients and heat flux (Table C-4) were taken from the WCOBRA/TRAC model results discussed in Appendix B. Appendix B presents a transient analysis. Please state at what transient time the input values for the Appendix C analysis were obtained. Please justify the input values used. Please explain and justify the type of thermal analysis, steady-state or transient, used in Appendix C.*

### 2.9.2 Response

The transient time was 1230 seconds.

The input values extracted from WCOBRA/TRAC were based on the hot rod shortly after the modeled debris blockage. The heat transfer data used was chosen for two reasons. First, the hot rod at the earliest recirculation time was chosen to maximize heat flux from the fuel rod, which is conservative for the Appendix C analysis. Second, a short delay after the modeled blockage was chosen so representative post debris blockage thermal hydraulic conditions were used while allowing the code some time to become more stable.

The analysis performed was a steady state analysis. This is conservative because it assumes that the heat flux remains constant at the levels immediately following the simulated accident. This also allows for the model to be more detailed as there were fewer variables that ANSYS needed to calculate, and the model was able to reach a steady state condition, which would have not been possible if the heat flux would have been changing.

## 2.10 RAI #10

### 2.10.1 Question

*Page xx, 1st paragraph states that specific areas addressed in WCAP-16793-NP include boric acid precipitation. However, boric acid precipitation is not addressed in WCAP-16793-NP beyond stating that it is being addressed in a program apart from WCAP-16793-NP. Please correct the document to state that the boric acid precipitation issue is being addressed in a separate Westinghouse program.*

### 2.10.2 Response

Page xxi and Section 8 states that the PWROG is funding a program to define, develop and obtain NRC approval of post-LOCA boric acid precipitation analysis scenarios, assumptions and acceptance criteria.

The Executive Summary will remove the statement that boric acid is addressed in WCAP-16793-NP. The first paragraph of page xx will be updated to read (changes are highlighted in **bold**):

> This evaluation considered the design of the PWR, the design of the open-lattice fuel, the design and tested performance of replacement containment sump screens, the tested performance of materials inside containment, and the tested performance of fuel assemblies in the presence of debris. Specific areas addressed in this evaluation include:
>
> - Blockage at the core inlet,
> - Collection of debris on fuel grids,
> - Collection of fibrous material on fuel cladding,
> - Protective coating debris deposited on fuel clad surfaces,
> - Production and deposition of chemical precipitants, **and**
> - ~~**Boric acid precipitation, and**~~
> - Coolant delivered from the top of the core.

## 2.11 RAI #11

### 2.11.1 Question

*Page xx, last paragraph states that the evaluations performed for the areas identified provide reasonable assurance of long-term core cooling for all plants. This statement is only true for those plants that show that they are bounded by the sump strainer bypass debris loads, maximum fuel cladding temperature, and maximum deposit thickness stated in the WCAP acceptance criteria. Please justify the statement or modify it.*

### 2.11.2 Response

The statement, as currently written, could be interpreted that the arguments presented in WCAP-16793-NP – without actions from the plants – provide reasonable assurance of LTCC for all plants. As stated, this interpretation is not correct as this argument is only applicable to plants that show they are bounded by the debris load acceptance criteria, maximum fuel cladding temperature and maximum deposit thickness requirements.

Page xx of the Executive Summary will be revised in the approved version of the WCAP once the SER is received to further clarify this statement is applicable only to plants that meet the defined acceptance criteria. This statement will read as follows (changes are highlighted in **bold)**:

> **~~The evaluations performed for the areas identified above provide reasonable assurance of LTCC for plants, Specifically,~~ In order to demonstrate reasonable assurance of LTCC, all plants must evaluate the areas identified above and demonstrate they are bounded by the debris load acceptance criteria, maximum fuel cladding temperature and maximum deposit thickness requirements. Specifically,**

## 2.12 RAI #12

### 2.12.1 Question

*Page xx, 2nd bullet in the last paragraph states that in the extreme case that a large blockage occurs, numerical analyses [presumably the WCOBRA/TRAC analysis referenced in Appendix B] have demonstrated that core decay heat removal will continue. NRC staff understands that the purpose of the fuel assembly head loss testing was to determine the maximum debris load conditions under which adequate coolant flow to the core can be maintained with the available driving head. Further, as stated in the Appendix B, the objective of the evaluation is to provide additional "defense in depth" to the fuel assembly testing to assure that long-term core cooling will be maintained. Please clarify the intent of the above referenced numerical analysis. If the intent is to justify a higher debris load, please justify the conclusion.*

### 2.12.2 Response

The intent of the WCOBRA/TRAC (WC/T) analyses is not to justify a higher debris load; they were performed to further bolster the assertion that core cooling flow will be maintained. The fuel assembly tests demonstrated a debris blockage can occur. Provided that the plants operate at a debris load that is less than that identified in Section 10, adequate core decay heat removal will be assured. The WC/T analyses provide an additional demonstration that, even with a blockage, sufficient liquid can enter the core to remove core decay heat once the plant has switched to sump recirculation. In this manner, the WC/T analyses are a defense in depth to the entire LTCC evaluation presented in WCAP-16793-NP.

In order to clarify this point, the 2nd bullet on Page xx will be revised as follows (changes are highlighted in **bold**):

> Decay heat will continue to be removed even with debris collection at the FA spacer grids. Plants that operate at the debris loads identified in Section 10 ~~**by the FA tests,**~~ can state that debris that bypasses the screen will not build an impenetrable blockage at the fuel spacer grids. ~~**In the extreme case that a large blockage does occur, numerical and first principle analyses have demonstrated that core decay heat removal will continue.**~~ **This assertion is bolstered by numerical and first principle analyses.** The details of this evaluation are provided in Section 4.

## 2.13 RAI #13

### 2.13.1 Question

*Page 2-1, paragraph 2.2, item 1 the WCAP states that the core "average" clad temperature will not exceed 800 F. As discussed in RAI responses 17 through 20 dated October 23, 2007 (Reference: WCAP-16793-NP, Revision 1, Appendix H, pages 21 and 22) the cladding temperature acceptance limit for long-term cooling shall be 800 F. Please revise the WCAP accordingly or justify the use of "average."*

### 2.13.2 Response

As stated in the Executive Summary and Section 2.3, page 2-2, the cladding temperature during recirculation from the containment sump will not exceed 800°F.

The acceptance criteria defined in Section 2.2, page 2-1 will be reworded to state (changes are highlighted in **bold**):

> **~~Maximum cladding temperature maintained during periods when the core is covered will not exceed a core average clad temperature of 800°F.~~ The cladding temperature during recirculation from the containment sump will not exceed 800°F.**

## 2.14 RAI #14

### 2.14.1 Question

*Page 3-2, paragraphs 3.1.1.1 and 3.1.1.2, state that the driving head criteria used for the Pressurized Water Reactor Owners Group fuel assembly tests can be found in references 3-1 and 3-2 (AREVA and Westinghouse proprietary reports, respectively). However, the proprietary reports do not provide the methods and design inputs used to calculate the driving head criteria. These calculations are required to enable staff to weigh the arguments presented in WCAP-16793-NP to conclude that there is adequate driving head to ensure adequate coolant flow into the core under the postulated debris loading conditions. Please make available, for NRC staff review, the calculations that establish the available driving head to ensure flow to the core. Please include information that shows that the single value chosen is bounding considering the variety of plant designs covered by the report.*

### 2.14.2 Response

The methods and design inputs are provided by reference. The Staff is invited to review and audit the references as desired.

The driving head used in the test protocol was chosen to be a representative value for all plants. As stated in Section 10.2.2, page 10-4, plants have to demonstrate that the available driving head (for both hot- and cold- leg breaks) is equal to or greater than the limits adhered to in the test program. The PWROG is providing a tool which the utilities can use to demonstrate compliance with the debris limits and show how a specific plant is bounded by the test conditions. Section 2.18 provides a methodology for plants to calculate the plant-specific driving head which can be used to determine if a utility is operating within the allowable debris limits.

## 2.15 RAI #15

### 2.15.1 Question

*The cold leg test results did not meet the acceptance criteria set forth for the test protocol. To show acceptable results, Section 4.2.2 acknowledges only the head loss across the bottom portion of the fuel assembly and argues that turbulence within the core would disrupt any debris bed that could form on the spacer grids. Further, the WCAP argues that, for a cold-leg break, analyses have shown that if the required make-up flow reaches the core, adequate long-term core cooling can be accomplished. Since the testing did not simulate actual flow conditions through the reactor core, please provide additional information to demonstrate that adequate turbulence would be present in the core to prevent the collection of debris on the spacer grids.*

### 2.15.2 Response

The original cold-leg break tests had acceptance criteria that differed slightly from the criteria defined by the test protocol. For the cold-leg break tests, the acceptance criterion was confined to the pressure drop at the core inlet. The debris that accumulated at the spacer grids was not part of the acceptance criteria.

Upon receipt of this RAI, the original cold-leg data set was supplemented with additional test data. For these tests, the acceptance criterion was changed to meet that described in the test protocol. That is, the dP over the entire fuel assembly was monitored as opposed to just the dP at the core inlet. These new test results provide the basis for the allowed debris load following a cold-leg break and are discussed in [6 & 7].

The last paragraph of Section 4.2.2 will be updated to state (changes are highlighted in **bold**):

> **~~Based on these conservatisms, it is reasonable to state that the debris buildup seen at the top spacer grids during the tests is not prototypical for a CL break. Instead, the debris buildup at spacer grids will be considerably lower than the debris buildup at spacer grids seen in the test with a low likelihood of extensive blockages at any one spacer grid. While debris may accumulate at these locations, the blockage will be localized and not extend across the core. Therefore, the pressure drop at the intermediate spacer grids for a CL break will be much less than that observed in the tests.~~**
>
> **At the fiber levels equal to or less than the cold-leg break limit (provided in Section 10), the collection of debris is limited to the core inlet. That is, debris does not travel into the fuel and catch at the spacer grids. However, it is possible that for other combinations of debris, debris beds may form at the core inlet and at the spacer grids (i.e. there may be multiple, distributed debris beds). The test results, and resulting debris limit, bound this situation, because the distributed debris beds would contain less fiber than a single debris bed at the core inlet. At less than the cold-leg debris limit, these smaller fiber beds would be considerably less resistant. Further, these distributed debris beds would not preclude fluid flow around the spacer grids such that decay heat removal was impeded.**

## 2.16 RAI #16

### 2.16.1 Question

*Please provide information that justifies the addition of ½ of the microporous insulation prior to the fibrous debris addition and ½ after the addition of fibrous debris and chemical debris. Strainer head loss testing guidance is to simultaneously add particulate insulation (e.g., microporous, cal-sil) and other particulate debris (e.g., coatings, latent dust, etc.). Adding the insulation debris after the chemical debris is potentially not conservative.*

### 2.16.2 Response

In the original fuel assembly tests [4 & 5], the microporous insulation was added in two parts in order to simulate the debris caused by the initial blast and the debris caused by the slow erosion of microporous material. Upon issuance of this RAI the PWROG conducted additional fuel assembly tests and evaluated the debris addition procedure of microporous insulation. These tests are summarized in [6 & 7].

These tests concluded microporous insulation behaves like a particulate and should be characterized as such. Therefore, plants with microporous insulation are bounded by the results of tests conducted with silicon carbide as the particulate source.

## 2.17 RAI #17

### 2.17.1 Question

*Please justify the statements in Sections 4.1 and 4.2.2 that debris accumulation will be localized and will not extend across the entire core. Flow through the core will distribute according to flow resistance. Once blockage occurs in a local area, flow of debris laden water will shift to areas with less resistance and debris will be deposited in those locations. Given sufficient debris, a uniform debris bed could be formed across the core inlet.*

### 2.17.2 Response

Section 4.1 does not state debris will extend across the core. Section 4.2.2 does state that debris accumulation will not extend across the core. However, the wording of Section 4.2.2 will be updated stated in the response to RAI #15. The proposed wording removes this statement. See Section 2.15.2 for the updated wording.

## 2.18 RAI #18

### 2.18.1 Question

*Section 3.1.3.2 states that the cold leg tests demonstrated that the hot leg test results are the bounding condition for in-vessel head loss. The assumption that the hot leg break is more limiting than the cold leg break condition led to the test program concentration on the hot leg break. However, limited cold leg break testing indicates that it may actually be more limiting than the hot leg break. Please provide information that justifies that the cold leg condition has been fully evaluated and that the debris loading acceptance criteria is valid for the cold leg condition.*

### 2.18.2 Response

Upon issuance of this RAI, the PWROG conducted additional fuel assembly tests. These tests included a study on the cold- and hot-leg break acceptance criteria. These tests are discussed in [6] and [7].

Section 10 of [2] will be updated to include the new cold- and hot-leg break debris load criteria. Section 10 will be updated as follows (changes are highlighted in **bold**):

SUMMARY

10.1 DISCUSSION

PWR containment buildings are designed to facilitate core cooling during a postulated LOCA event. In some LOCA scenarios, the cooling process requires water discharged from the break, ECCS, and CSS to be collected in a sump for recirculation by these systems. The discharged coolant water in the sump will contain chemical impurities and debris as the result of interaction with containment materials.

There has been concern that following a LOCA, the chemical precipitate, fibrous and particulate debris within the sump could collect on the sump screen and block the flow of cooling water into the core. There is also concern about the effects of the debris that passes through the sump screen. This debris could be ingested into the ECCS and flow into the RCS.

The PWROG sponsored a program to analyze the effects of debris and precipitates on core cooling for PWRs when the ECCS is realigned to recirculate coolant from the containment sump. The intent was to demonstrate adequate heat-removal capability for all plant scenarios. Additionally, the PWROG initiated prototypical FA testing to establish limits on the debris mass (particulate, fibrous, and chemical) that could bypass the reactor containment building sump screen. These debris limits will not cause unacceptable head loss that would impede core inlet flow and challenge LTCC. These limits will be referred to as the debris load acceptance criteria and are intended to demonstrate that adequate flow for long-term decay heat removal exists at these levels.

This evaluation considered the design of the PWR, the design of the open-lattice fuel, the design and tested performance of replacement containment sump screens, the tested performance of materials inside containment, and the tested performance of fuel assemblies in the presence of debris. Specific areas addressed in this evaluation included:

- Blockage at the core inlet,
- Collection of debris on fuel grids,

- Collection of fibrous material on fuel cladding,
- Protective coating debris deposited on fuel clad surfaces,
- Production and deposition of chemical precipitants, **and**
- ~~**Boric acid precipitation, and**~~
- Coolant delivered from the top of the core.

The following acceptance criteria were selected for the evaluation of the topical areas identified above:

1. The maximum clad temperature shall not exceed 800°F.
2. The thickness of the cladding oxide and the fuel deposits **shall** ~~**should**~~ not exceed ~~**an average of**~~ 0.050 inches in any fuel region.

These acceptance bases were applied after the initial quench of the core and are consistent with the LTCC requirements stated in 10 CFR 50.46 (b)(4) and 10 CFR 50.46 (b)(5). They do not represent, nor are they intended to be, new or additional LTCC requirements. These acceptance bases provide for demonstrating that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal.

~~**The evaluations performed for the areas identified above provide reasonable assurance of LTCC for plants, Specifically,**~~ **In order to demonstrate reasonable assurance of LTCC, all plants must evaluate the areas identified above and demonstrate they are bounded by the debris load acceptance criteria, maximum fuel cladding temperature and maximum deposit thickness requirements. Specifically,**

- Adequate flow to remove decay heat will continue to reach the core even with debris from the sump reaching the RCS and core. Plants that operate at the debris loads identified in ~~**Tables 10-1**~~ **Section 10.2** ~~**(and 10-2, if applicable),**~~ can state that debris that bypasses the screen will not build an impenetrable blockage at the core inlet. While any debris that collects at the core inlet will provide some resistance to flow, in the extreme case that a large blockage does occur, numerical analyses have demonstrated that core decay heat removal will continue. The details supporting this evaluation are provided in Section 3.
- Decay heat will continue to be removed even with debris collection at the FA spacer grids. Plants that operate at the debris loads identified in Section 10 ~~**by the FA tests,**~~ can state that debris that bypasses the screen will not build an impenetrable blockage at the fuel spacer grids. ~~**In the extreme case that a large blockage does occur, numerical and first principle analyses have demonstrated that core decay heat removal will continue.**~~ **This assertion is bolstered by numerical and first principle analyses.** The details of this evaluation are provided in Section 4.
- Fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding. Thus, fibrous debris will not form a "blanket" on clad surfaces to restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of fibrous

debris to the cladding is not plausible and will not adversely affect core cooling. The details supporting this evaluation are provided in Section 5.

- Protective coating debris, should it enter the core region, will not restrict heat transfer and cause an increase in clad temperature. Therefore, adherence of protective coating debris to the cladding is not plausible and will not adversely affect core cooling. The details supporting this evaluation are provided in Section 6.
- The chemical effects method developed in WCAP-16530-NP-A was extended to develop a method to predict chemical deposition of fuel cladding. The calculational tool, LOCADM, will be used by each utility to perform a plant-specific evaluation. It is expected that each plant will be able to use this tool to show that decay heat would be removed and acceptable fuel clad temperatures would be maintained. The details for using LOCADM are provided in Section 7 and Appendix E.
- The commonly used approach for demonstrating adequate boric acid dilution in a post-LOCA scenario includes the use of simplified methods with conservative boundary conditions and assumptions. In light of NRC staff and ACRS challenges to the simplified methods commonly used, it has recently become clear that additional insights and new methodologies are needed to answer fundamental questions about boric acid mixing and transport in the RCS and potential precipitation mechanisms that may occur both during the ECCS injection phase and the sump recirculation phase after a LOCA. This will be addressed in a separate PWROG program. This program is discussed in Section 8.
- The PWROG FA test results demonstrated that sufficient flow will reach the core to remove core decay heat. UPI plants that operate at the debris loads identified in ~~Tables 10-1~~ **Section 10.2**, can state that debris that bypasses the screen will not build an impenetrable blockage within the core region. The details supporting this evaluation are provided in Section 9.

**10.2 Acceptance Criteria Debris Limits**

**10.2.1 Cold-Leg Acceptance Criteria**

**See response to RAI #2 [7] for Westinghouse fuel.**

**See response to RAI #4 [6] for AREVA fuel.**

**10.2.2 Hot-Leg Acceptance Criteria**

**See response to RAI #2 [7] for Westinghouse fuel.**

**See response to RAI #4 [6] for AREVA fuel.**

~~10.2~~ **10.3** GUIDANCE TO LICENSEES CONCERNING EVALUATION OF DEBRIS

Actions are required of utilities to prove acceptable LTCC with debris and chemical products in the recirculating fluid. Plants will have to perform plant-specific LOCADM evaluations and prove the plant conditions are bounded by the debris load acceptance criteria. These actions along with reference to this report provide the basis for demonstrating that LTCC will not be compromised following a LOCA as a consequence of debris ingestion to the RCS and core.

~~10.2.1~~ **10.3.1** LOCADM

Plants will have to perform a LOCADM evaluation (Section 7 and Appendix E) based on plant-specific debris inputs and prove they are within the acceptance criteria.

~~10.2.2~~ **10.3.2** Debris Acceptance Criteria

Debris loads used in the FA test program were based on sump screen bypass information provided by licensees. The FA testing was reported in proprietary submittals that support this document. The results from these FA tests are discussed in the proprietary test reports (References 10-1 and 10-2).

As part of the effort to invoke this WCAP in the plant licensing basis, each plant will compare their plant-specific debris load against the FA debris masses tested. Plants that have bypass debris loadings that are within the limits of the debris masses tested are bounded by the test. Plants will also have to demonstrate that the available driving head (for both hot and cold leg breaks) is equal to or greater than the limits adhered to in this test program. **The following sections demonstrate how to calculate the plant-specific available driving head values:**

**10.3.2.1 Method Discussion**

**The FA testing program undertaken by the PWROG was designed to provide reasonable assurance that sufficient flow will reach the core to remove core decay heat. To that end, it must be demonstrated that the head available to drive flow into the core is greater than the head loss (also referred to as pressure drop) across the core due to possible debris buildup. The following relationship must be true to ensure sufficient flow is available:**

$$dP_{available} > dP_{debris}$$

**The available driving head ($dP_{available}$) is a plant-specific value. The pressure drop due to debris ($dP_{debris}$) is determined by the FA test program (summarized in (References 2, 3, 4 and 5)).**

**The following sections demonstrate the method to be used to calculate the plant-specific $dP_{available}$.**

**10.3.2.2 Discussion of Significant Assumptions**

1. **For hot-leg (HL) $dP_{available}$: Core voiding is neglected and the core liquid level is assumed to be at the bottom of the hot leg. This is conservative because it maximizes the static head of the liquid in the core region.**

2. **For HL and cold-leg (CL) $dP_{available}$: The downcomer (DC) liquid density is based on the sump liquid conditions. Plant specific conditions should be used to define the DC liquid density. Since density is inversely proportional to liquid temperature and a lower density will reduce the driving head from the DC, a conservatively high sump liquid temperature should be selected. For example, at the time of sump switchover the sump liquid temperature is approximately 180°F to 250°F (Reference 6, Attachment V-1). As time progresses, the sump liquid temperature will decrease as core decay heat decreases and the decay heat coolers become more effective. Thirty days after the event, the liquid may approach 120°F to 150°F (Reference 6, Attachment V-1). Therefore, a density**

**corresponding to a liquid temperature of 250°F will bound the sump liquid conditions. The liquid density is also a function of the containment pressure. The containment pressure may be as high as 60 psia early in the event and then continually decrease throughout the event. A density corresponding to temperature and pressure combination in this range is approximately 59 lbm/ft$^3$.**

**Additionally, at the time of the event, the conditions in the core result in a liquid density that is less than the liquid density of the DC. Therefore, it is conservatism to set the core liquid density equal to the density of the DC liquid.**

3. **For HL and CL $dP_{available}$: The reactor vessel DC and lower plenum k/A$^2$ is small (typically << 0.1). Further, the liquid density is large (~60 lbm/ft$^3$) and bulk velocity is low. Therefore, the losses in these regions can be neglected.**

4. **For CL $dP_{available}$: Core Void Fraction ($\alpha$)**
**The core void fraction ($\alpha$) changes with time so a time dependent relationship was developed. The data in Table 10-1 was used to plot Figure 10-1. A trend line was added to Figure 10-1 and that equation is used as the time dependent relationship:**

$\alpha_{core} = 1.1128 * t \wedge -0.1183$

**Table 10-1 Core Void Fraction Following a CL Break**

| Time (sec) | Void Fraction |
|---|---|
| 1200 | 0.5 |
| 36,000 | 0.3 |
| 2,592,000 | 0.2 |

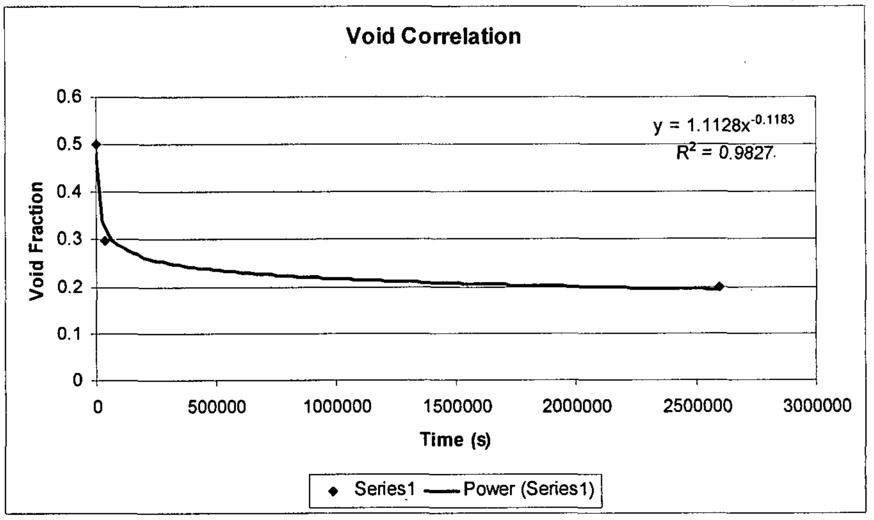


**Figure 10-1 Core Void Fraction Following a CL Break**

5. **For CL $dP_{available}$: Core Boil-off Rate (w)**

**The flow rate to the break is equal to the core boil-off rate: = [(core decay heat)]/[(core enthalpy rise)] = $Q_{DH}/\Delta H$.**

**5.1 Core Decay Heat ($Q_{DH}$)**

**The core heat is calculated as a function of time:**

**$Q_{DH} = (P/P_o)(P_o)(947\ \text{Btu/s})$**

- **$P/P_o$ = decay heat ratio**
- **$P_o$ = Power with uncertainty (MWt)**

**The decay heat ratio is calculated using the following equation:**

**(0.1741)(t ^ -0.2805)**

**This equation is based on the Appendix K decay heat of 1.2*ANSI '71. A comparison of the Appendix K values to the trendline is provided in Figure 10-2.**

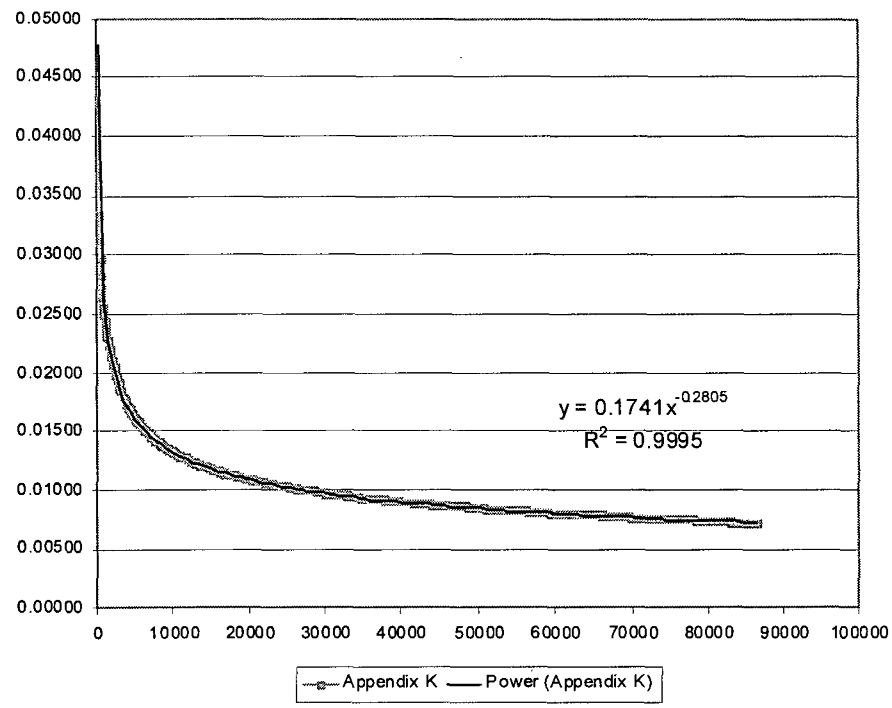


**Figure 10-2 Comparison of Appendix K Decay Heat Ratio to Trendline Equation**

### 5.2 Core Enthalpy Rise (ΔH)

**The enthalpy rise in the core is a function of the core inlet subcooling and the saturation pressure at the core exit. The enthalpy rise in this calculation is the latent heat of vaporization; therefore, $\Delta h = h_{fg}$.**

$$h_{fg} = \left(-0.000131 * P^3_{core\_exit}\right) + \left(0.02838 * P^2_{core\_exit}\right) - \left(2.726 * P_{core\_exit}\right) + 1005$$

**hfg is determined using the core exit pressure, which is based on the containment pressure plus an increase for flow losses through the loops.**

**$h_{fg} = f(P_{core})$ $P_{core} = P_{cont} + dP_{loops}$**

- **$P_{cont}$ = plant-specific input (cases are evaluated for maximum and minimum containment pressures)**
- **$dP_{loops}$ is calculated using the following formula:**

**$= (K/(A \wedge 2) * w \wedge 2) / (288 * \rho_{loop} * g_c)$**

**To develop the relationship for $h_{fg}$, the values were obtained at various pressures (assuming saturated conditions) from the ASME steam tables. The data is shown in Table 10-2 plotted on Figure 10-3. The resulting trendline is used to calculate $h_{fg}$ based on plant-specific containment pressures:**

**$h_{fg} = (-0.000131 * (P_{core}) \wedge 3) + (0.02838 * (P_{core}) \wedge 2) - (2.726 * (P_{core})) + 1005$**

**Table 10-2 $h_{fg}$ as a Function of Saturation Pressure**

| Pressure (psia) | $h_{fg}$ (BTU/lbm) |
|---|---|
| 5 | 992.1 |
| 10 | 982.1 |
| 15 | 969.7 |
| 20 | 960.1 |
| 25 | 952.2 |
| 30 | 945.4 |
| 35 | 939.3 |
| 40 | 933.8 |
| 45 | 928.8 |
| 50 | 924.2 |
| 55 | 919.8 |
| 60 | 915.8 |
| 65 | 911.9 |
| 70 | 908.2 |
| 75 | 904.7 |

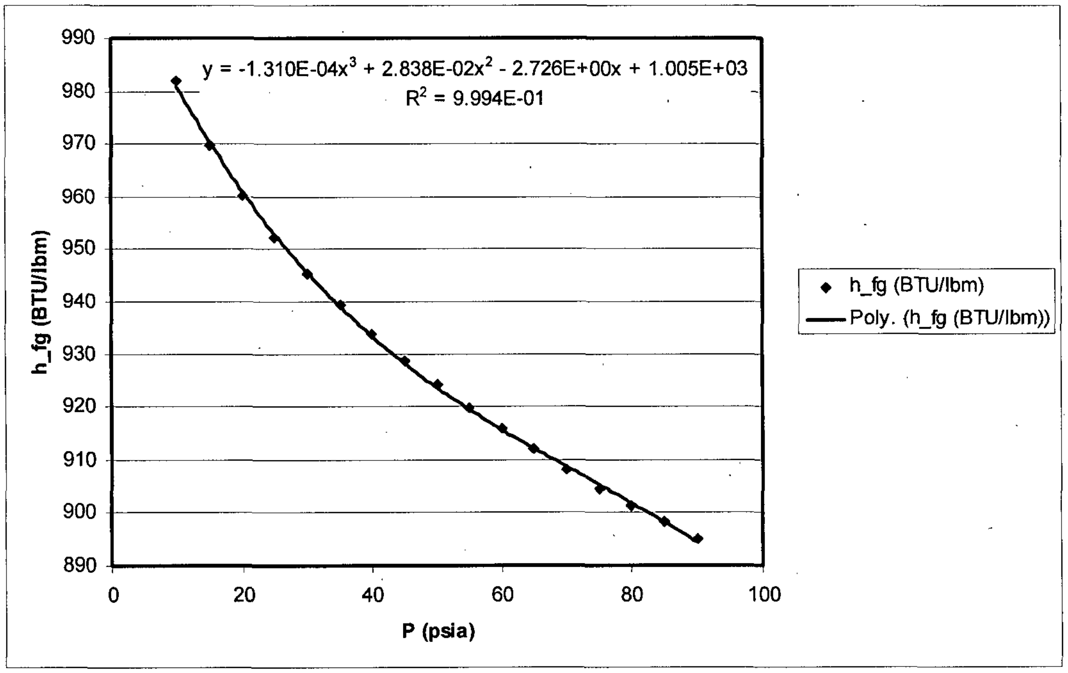


**Figure 10-3 Latent Heat of Vaporization as a Function of Core Pressure**

6. **For CL $dP_{available}$: Fluid Density ($\rho_{loop}$)**

**SG secondary side temperature is considered for steam density in loop losses. The following can be used to calculate this value.**

**Wet (or saturated) steam is produced in the core and begins to makes its way through the loops to the break. Superheated steam has a lower density than saturated steam. Since the density is used in the denominator of the Darcy equation, a lower density produces a higher head loss.**

**When the saturated steam reaches the SG, it will likely pick up heat from the residual heat in the secondary side fluid. As this steam removes the secondary side heat, the secondary side fluid temperature will decrease. At the time of sump switchover, the temperature will be in the 300°F to 400°F range, or below, and continue to decrease as steam flow travels through the SG. To obtain a conservative steam density, the steam is assumed to heat to the secondary side temperature. So, the loop density can be determined at a constant temperature and the containment pressure at the time of interest.**

**To develop the relationship for $\rho_{loop}$, the values for steam density were obtained at various pressures (assuming 400°F) from the ASME steam tables. The data is shown in Table 10-3 and plotted on Figure 10-4. The resulting trendline is used to calculate $\rho_{loop}$ based on plant-specific containment pressures:**

$$\rho_{loop} = 0.002024 * Pc - 0.001583$$

**Table 10-3 $\rho_{loop}$ as a Function of Containment Pressure at 400°F**

| Pressure (psia) | rho (lbm/ft$^3$) |
|---|---|
| 5 | 0.009 |
| 10 | 0.018 |
| 15 | 0.029 |
| 20 | 0.039 |
| 25 | 0.049 |
| 30 | 0.059 |
| 35 | 0.069 |
| 40 | 0.079 |
| 45 | 0.089 |
| 50 | 0.099 |
| 55 | 0.11 |
| 60 | 0.12 |
| 65 | 0.13 |
| 70 | 0.14 |
| 75 | 0.15 |

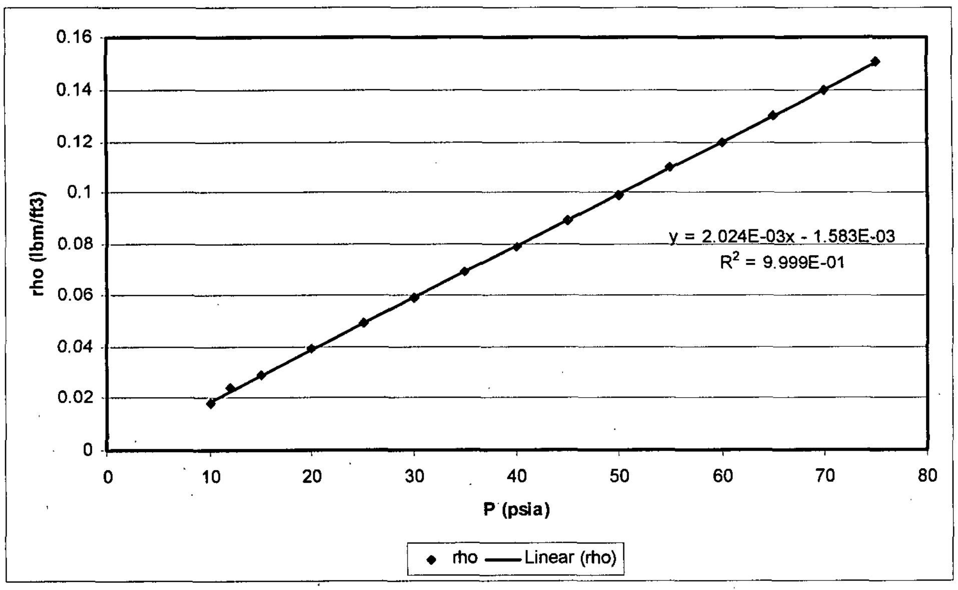


**Figure 10-4 $\rho_{loop}$ as a Function of Containment Pressure at 400°F**

#### 10.3.2.3 Available Driving Head

**Core flow is only possible if the manometric balance between the downcomer (DC) and the core is sufficient to overcome the flow losses in the RV DC, RV lower plenum, core, and loops at the suitable flow rate. Plants must demonstrate the plant-specific $dP_{available}$ is greater than the tested $dP_{debris}$.**

#### 10.3.2.3.1 Hot-Leg Break

**For a hot leg break, the ECCS must pass through the core to reach the break. The driving force is the manometric balance between the liquid in the DC and core. Should a debris bed begin to build up in the core, the liquid level will begin to build in the cold legs and SG. As the level begins to rise in the SG tubes, the elevation head to drive the flow through the core increases as well. The driving head reaches its peak when the shortest SG tube has been filled (for W and CE plants) or the SG and HL have been filled to the U-bend spillover elevation (for the B&W plants). Once the ECCS flow reaches this elevation, no increase in water level to the higher tubes is achieved.**

**Core flow is only possible if the manometric balance between the DC and the core is sufficient to overcome the flow losses in the DC, RV LP, core, and reactor vessel vent valves (RVVVs) at the core boil-off rate. Even in the presence of a debris bed, adequate flow will continue to remove decay heat. Plants can demonstrate this as long as the head available to drive flow into the core is greater than the head loss at the inlet due to the debris buildup:**

$$\mathbf{dP_{available} > dP_{debris}}$$

**<u>$dP_{available}$</u>**

$$dP_{avail} = dP_{dz} - dP_{flow}$$

**The manometric balance between the DC and the bottom of the core is calculated by:**

$$dP_{dz} = dP_{DC} - dP_{core}$$

- **$dP_{DC}$ = elevation head due to liquid in the DC and SG to spillover elevation**

  $$dP_{DC} = (Z_{so} - Z_{core-in})(\rho_{DC}) / (144 in^2)$$

  - **$Z_{so}$ = SG or HL spillover elevation (with respect to datum) (ft)**
  - **$Z_{core-in}$ = Elevation of bottom of the core (with respect to datum) (ft)**
  - **$\rho_{DC}$ = liquid density in DC and SG (lbm/ft$^3$)**

- **$dP_{core}$ = elevation head due to liquid in the core**

  $$dP_{core} = (Z_{brk} - Z_{core-in})(\rho_{core}) / (144 in^2)$$

  - **$Z_{brk}$ = Elevation of the break (bottom of hot-leg elevation) (ft)**
  - **$Z_{core-in}$ = Elevation of bottom of the core (ft)**

- $\rho_{core}$ = liquid density in core (lbm/ft$^3$)

**The above elevations can be determined by consulting plant drawings. The density in the DC and core are discussed in significant assumptions.**

**As stated in the significant assumptions, the flow losses in the DC, lower plenum and core are negligible and the loop losses are zero. Therefore, the HL $dP_{available}$ equation can be simplified to:**

$$dP_{avail} = dP_{dz}$$

**10.3.2.3.2 Cold-Leg Break**

**<u>$dP_{available}$</u>**

**For the CL breaks, the ECCS from each CL runs to the break, ensuring that the DC is full to at least the bottom of the CL nozzles. The $dP_{available}$ is established by the manometric balance between the DC liquid level, the core liquid level, and pressure drop through the RCS loops due to the steam flow:**

$$dP_{avail} = dP_{dz} - dP_{flow}$$

- **$dP_{dz}$ = available driving head due to the liquid level difference in the DC and core**
- **$dP_{flow}$ = pressure drop due to flow losses in the DC, lower plenum (LP), core and loops**

**The available head due to the liquid level difference between the DC and core ($dP_{dz}$) is calculated by:**

$$dP_{dz} = dP_{DC} - dP_{core} = \frac{\left(dz_{DC-core\_inlet} - (1-\alpha_{core}) * dz_{core\_outlet-core\_inlet}\right) * \rho}{144}$$

- **$dz_{DC-core\_inlet} = h_{bottom\_of\_CL} - h_{core\_inlet}$ (ft)**
  - **$h_{bottom\_of\_CL}$ = height of bottom of cold leg (ft)**
  - **$h_{core\_inlet}$ = height of core inlet (ft)**
- **$dz_{core\_outlet-core\_inlet} = h_{core\_outlet} - h_{core\_inlet}$ (ft)**
  - **$h_{core\_outlet}$ = height of core outlet [top of active fuel (ft)]**
  - **$h_{core\_inlet}$ = height of core inlet [bottom of active fuel (ft)]**
- **$\alpha$ = core void fraction**
- **$\rho$ = liquid density (lbm/ft$^3$)**

**The flow losses ($dP_{flow}$) are calculated using the Darcy Equation:**

$$dP_{flow} = \left(\frac{k}{A^2}\right)\left(\frac{w^2}{288 * \rho_{loop} * g_c}\right)$$

- **For Westinghouse and CE plants, $k/A^2$ = (total loss coefficient through the loops)/(area upon which k is based $(ft^2)^2$)**
- **The ratio of $k/A^2$ is geometry dependent and should include losses due to:**
  - **Friction in the SG**
  - **Expansion and contraction through the SG tubes**
  - **Losses in the pipe bends**
- **For B&W plants, $k/A^2$ = (total loss coefficient through the RVVVs)/(area of RVVVs $(ft^2)^2$)**
- **w = flow rate (lbm/s)**
  **The flow rate to the break is the core boil-off rate.**
- **$\rho_{loop}$ = fluid density (lbm/ft$^3$) (Defined in Section 10.3.2.2, Assumption #6)**
- **$g_c$ = gravitational constant = 32.2 $lb_m$-ft/lbf-sec$^2$**

**10.3.2.4 Input**

**All the required inputs can be found from plant-specific drawings and evaluations.**

**10.3.2.4.1 Calculation of Hot-Leg Break $dP_{available}$**

**The values in Table 10-4 are required to calculate the hot-leg break available head loss.**

**Table 10-4 Required Inputs for Calculation of HL $dP_{available}$**

| Variable | Description of Plant-Specific Parameter | Unit | Source |
|---|---|---|---|
| | Datum | ft | Location in the plant to which all subsequent elevations are given with respect to. |
| $Z_{SO}$ | Hot leg (HL) or steam generator (SG) spillover elevation | ft | Elevation of shortest SG tube for W or CE plants with respect (wrt) to the datum. Elevation of HL spillover elevation for B&W plants wrt the datum. |
| $Z_{core\text{-}in}$ | Bottom of active fuel | ft | Plant geometry wrt the datum. |
| $\rho_{DC}$ | RV downcomer (DC) liquid density | lbm/ft$^3$ | Define density at following conditions: ECCS liquid temperature and maximum containment pressure. [The conservative value is to define the density at the saturation pressure at the max containment pressure following a LOCA.] |
| $Z_{brk}$ | $= Z_{RVCL} - (Z_{ID\ HL}/2)$ | ft | |
| $Z_{RVCL}$ | Reactor vessel nozzle centerline | ft | Plant geometry wrt the datum. |
| $Z_{ID\ HL}$ | Inner diameter of HL pipe | in | Plant geometry wrt the datum. |
| $\rho_{core}$ | Core liquid density | lbm/ft$^3$ | Set equal to RV DC liquid density |

#### 10.3.2.4.2 Calculation of Cold-Leg Break $dP_{available}$

**The values in Table 10-5 are required to calculate the cold-leg break available head loss. Additionally, the $dP_{available}$ for a cold-leg break is dependent upon the time at which the value is calculated. Therefore, the inputs described here can be used to calculate the expected $dP_{available}$ as function of time. Since the boiloff rate decreases with time, the $dP_{available}$ will change throughout the event.**

**Table 10-5 Required Inputs for Calculation of CL $dP_{available}$**

| Variable | Description of Plant-Specific Parameter | Unit | Source |
|---|---|---|---|
| | Datum | ft | Location in the plant to which all subsequent elevations are given with respect to. |
| $dz_{DC\text{-}core\ inlet}$ | $=h_{bottom\ of\ CL} - h_{core\ inlet}$ | ft | |
| $h_{bottom\ of\ CL}$ | $= Z_{RVCL} - (Z_{ID\ CL}/2)$ | ft | |
| $Z_{RVCL}$ | Reactor vessel nozzle centerline | ft | Plant geometry wrt the datum. |
| $Z_{ID\ CL}$ | Inner diameter of CL pipe | in | Plant geometry. |
| $h_{core\ inlet}$ | Bottom of active fuel | ft | Plant geometry wrt the datum. |
| $h_{core\ outlet}$ | Top of active fuel | ft | Plant geometry wrt the datum. |
| $\rho$ | RV downcomer (DC) liquid density | $lbm/ft^3$ | Define density at following conditions: ECCS liquid temperature and maximum containment pressure. [The conservative value is to define the density at the saturation pressure at the max containment pressure following a LOCA.] |
| $\alpha$ | Core void fraction. This value changes with time so the time dependent relationship presented in Section 10.3.2.2 (Assumption #4 should be used). | | |
| $k/A^2$ | $k/A^2$ | $ft^{-4}$ | From LOCA analyses. For W or CE plants, this value includes form and friction losses through all loops in a parallel flow configuration. For B&W plants, this value represents the form-losses through the reactor vessel vent valves (RVVVs). |
| w | The flow rate to the break is the core boil-off rate: = [(core decay heat)]/[(core enthalpy rise)] $= Q_{DH}/\Delta H$ This value changes with time so the relationship presented in Section 10.3.2.2 (Assumption #5) should be used. | | |
| $\rho_{loop}$ | This value changes with time so the relationship presented in Section 10.3.2.2 (Assumption #6) should be used. | | |
| $g_c$ | Gravitational constant = $32.2 lb_m\text{-}ft/lb_f\text{-}sec.$ | | |
| $P_{cont}$ | Max containment pressure | psia | The maximum containment pressure. The containment design calculation will have this data. Some values will have to be extrapolated in order to fill in the necessary time steps. |
| $P_{cont}$ | Min containment pressure | psia | The minimum containment pressure. The LOCA linear heat rate (LHR)/ peak clad temperature analyses will have this data. Some values will have to be extrapolated in order to fill in the necessary time steps. |

#### 10.3.2.4 Additional Considerations

**During discussions with the NRC staff, the following information was provided and is recorded here.**

**High Level Summary**

- **LOCA analyses are generally performed at the onset of the event and do not evaluate the onset of recirculation. The core voiding used in this calculation is provided to be conservative. Each utility can demonstrate the applicability of these values by submitting plant-specific core void fraction values. These values can be calculated on a plant specific basis using the same model (RAI F.1, ML051940575) approved by the Staff (ML061720376) for the Beaver Valley Power Station Extended Power Uprate, however other methods may be used provided they produce conservative results.**
- **While the LOCA analyses are used for the early LOCA response, the losses through the loops are appropriate for use in this application.**
- **The ANS-71 + 20% curve with actinides is used and is the same curve used in LOCADM. Therefore, all questions regarding decay heat are answered by the use of the conservative decay heat correlation.**

**Staff Question 1: A simple expression is used to compute the void fraction in the core vs. time and is given in section 3.2 item # 4. The void fraction in the core is a of the core geometry, inlet flow rate, subcooling, decay power level, core pressure, boric acid content, and external loop resistance. It is a very strong function of the core axial power shape (top vs bottom peak can cause a large change in the core void fraction. It is not a simple function of time. The table in item # 4 in this section shows the void fraction in the core of 0.5. The void fraction in the Millstone 3 core at 1200 sec following a large break LOCA is 70%. As such, the void fraction computation may not properly represent the fluid conditions during the long term. Please explain the background and technical basis for the void fraction correlation and show that it is a conservative bound for long term conditions that also include the above effects.**

**Answer 1:**

- **The calculation of these values accounts for the power level and axial shape variation.**
- **This core void fraction does not affect the loop loss calculation. This value is simply used to account for the water column in the core. At a lower core void fraction, the core has a larger water column and the resistance to flow into the core is greater. This results in a conservatively low calculation of available driving head.**
- **The void fraction included in the PWROG submittal was chosen to be conservative. That is, the lower void fraction results in less available driving head. Therefore, if Millstone does have a void fraction of 70% at 1200 seconds, they would actually have a larger driving head than what is calculated with the 50% presented in item #4.**

**Staff Question 2: The loop pressure loss should include the RCP locked rotor k-factor and is not mentioned. Was the locked rotor condition included? Please explain.**

**Answer 2:**
**The locked rotor k-factor is not included. This is consistent with WCAP-8163 and WCAP-12945-P-A, Volume 5, Revision 1, Appendix C, Additional Comment 26.**

**Staff Question 3: Does the loop resistance include the effects of the hot leg nozzle gaps and core barrel alignment key leakage paths? What is their effect? Please explain.**

**Answer 3:**
**The hot leg nozzle gaps and core barrel alignment key leakage paths are not included. It is conservative to ignore these steam vent paths in the loop pressure drop calculation.**

**References used in Sections 10.3.2.1 through 10.3.2.4:**

1. **WCAP-16793-NP, Revision 1, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," April 2009.**
2. **WCAP-17057-P, Revision 0, "GSI-191 Fuel Assembly Test Report for PWROG," March 2009.**
3. **51-9102685-000, "GSI-191 FA Test Report for PWROG," March 2009.**
4. **OG-10-47, "PWR Owners Group: PWROG Response to Request for Additional Information Regarding PWROG Topical Report WCAP-17057-P, Revision 0, 'GSI-191 Fuel Assembly Test Report for PWROG,' (PA-SEE-0480)," February 2010.**
5. **OG-10-46, "PWR Owners Group: PWROG Response to Request for Additional Information Regarding PWROG Topical Report 51-9102685-000, Revision 0, 'GSI-191 FA Test Report for PWROG,' (PS-SEE-0479)," February 2010.**
6. **NEI 04-07, Revision 0, Volume 2, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004.**

Several courses of action have been identified for plants whose debris loads are outside of the limits tested. These actions include, but are not limited to, reduction of problematic debris sources by removing or restraining the affected debris source, plant-specific FA testing, or engineering evaluations. ~~Engineering evaluations could be applicable to plants that have one debris source that is slightly higher than the acceptance criteria but all other debris sources are significantly lower than the recommended limits.~~ These evaluations can also be used for plants that have different fuel filters, greater driving head, among other variables.

The last paragraph of Section 3.1.3.2 will be removed.

## 2.19 RAI #19

### 2.19.1 Question

*Please provide information that justifies the statement in Sections 4.2.1 that "with boiling, additional turbulence is present in the core region which will tend to remove debris from the spacer grids and confine blockages to isolated regions." Provide the bases and assumptions associated with this assertion. Further, it seems that boiling could add solids (due to precipitation) that combine with the debris, increasing the density and decreasing the likelihood that such material would be removed from the fuel surfaces. Please provide evidence to demonstrate that the lack of boiling in the testing is in fact conservative.*

### 2.19.2 Response

Testing without boiling is conservative for both hot- and cold-leg break scenarios, as discussed below.

**Hot-Leg Break**

At hot-leg (HL) flow rates, based on observations from fuel assembly testing, multiple debris beds will form at the spacer grids. While boiling could occur, the lack of boiling in the tests is conservative because the available driving head is calculated assuming a liquid core. If boiling were considered, a void fraction would be added to the available driving head calculation and the available driving head would increase. An increased available driving head would result in an increased maximum fiber limit. Therefore, by not considering boiling, the maximum fiber limit is held conservatively low.

Additionally, WCAP-16793-NP has two calculations that account for the accumulation of debris on spacer grids and fuel rods:

1. LOCADM addresses the concerns related to precipitation. LOCADM deposits chemical products that are dissolved or suspended in solution throughout the core in proportion to the amount of boiling in each core node. In order to demonstrate LTCC, all utilities must demonstrate the accumulation of the fuel deposits and the cladding oxide will not exceed 0.05 inches.
2. Analyses were conducted to predict fuel cladding heat up within a spacer grid. These are detailed in Appendix C & D of [2]. This analysis showed that with localized blockages, the maximum temperature that would be achieved is less than 750°F. This is a very conservative value because this calculation assumed no flow through the debris in the grid.

These calculations were based upon a heat transfer coefficient of 650 Btu/hr-ft$^2$-F. Upon discussion of these calculations, it was determined that some of the assumptions were not clearly recorded. Therefore, key assumptions are summarized here:

- The calculations were made for conditions at time of switchover from RWST/BWST injection to recirculation from the reactor containment building sump:
    a) The time used for this evaluation was 1200 seconds after the postulated LOCA
    b) The decay heat is at its maximum value for the recirculation time period
    c) The blockage was arbitrarily assumed to occur at 1200 seconds
- The heat transfer boundary conditions were taken from a WCOBRA/TRAC calculation used for WCAP-16793-NP:

a) The event was a large break LOCA
b) Flow into the core was by gravity head
c) Flow into the core was driven by matching boil off
d) The core modeled was a high power density core
e) The power shape was skewed to the top
f) These features provide for a maximum clad temperature at the upper elevations of the core

- The heat transfer conditions taken from the WCOBRA/TRAC output are as follows:

a) At ~11.5 ft in the core (top), the core is in nucleate boiling

$h_{LIQUID}$ = 654 Btu/hr-ft$^2$-°F

$h_{VAPOR}$ = 19 Btu/hr-ft$^2$-°F

b) At ~6 ft in the core (mid-plane), the core is either subcooled or in nucleate boiling

$h_{LIQUID}$ = 1,006 Btu/hr-ft$^2$-°F

$h_{VAPOR}$ = 8 Btu/hr-ft$^2$-°F

c) At ~0 ft in the core (bottom active length), the core is single phase liquid heat transfer

$h_{LIQUID}$ = 466 Btu/hr-ft$^2$-°F

$h_{VAPOR}$ = 0 Btu/hr-ft$^2$-°F

- The calculations were performed with the assumption that a blockage at the peak power location would provide for a prediction of the maximum clad temperatures and used:

a) The heat transfer conditions at ~ 11.5 ft in the core
b) The decay heat of 1200 seconds, skewed to the top of the core

Cladding temperatures at or below 800°F maintain the clad within the temperature range where additional corrosion and hydrogen pickup over a 30 day period will not have a significant effect on cladding properties. The data in Table 19-1 is generated from the key assumptions (listed above) and is presented in [2]. This data shows that even with localized blockages, the maximum temperature that would be achieved is less than 750°F.

**Table 19-1 Clad/Oxide Interface Temperature vs. Chemical Precipitate Thickness**

| Chemical Precipitate Thickness (mils) | $k_{precipitate}$ = 0.1 BTU/hr-ft-°F | | |
|---|---|---|---|
| | 0.36" OD rod (°F) | 0.422" OD rod (°F) | 0.416" OD rod (°F) |
| 0 | 273 | 283.6 | 286.6 |
| 10 | 336 | 377.0 | 376.9 |
| 20 | 396 | 466.4 | 466.2 |
| 30 | 453 | 552.1 | 551.9 |
| 40 | 508 | 634.5 | 634.1 |
| 50 | 560 | 713.8 | 713.2 |

## Cold-Leg Break

As for the cold-leg (CL) break, testing was not conducted with boiling. As observed in testing, the low flow rate and small amount of allowed fiber is conducive to the formation of a single debris bed (either at the bottom nozzle or at the first spacer grid). Boiling would not be expected to occur at this elevation so the test results would not be different had boiling been introduced.

Additionally, WCOBRA/TRAC analyses were performed to demonstrate that adequate flow is provided and redistributed within the core to maintain adequate LTCC in the event of core blockage. These analyses, considering up to 99.4 percent core blockage, showed that sufficient liquid could enter the core to remove core decay heat once the play had switched to sump recirculation. The details of this evaluation are provided in WCAP-16793-NP, Revision 1.

However, it is possible that a combination of debris not tested could result in a varying debris bed formation. That is, it is possible that debris beds may form at the core inlet and at the spacer grids. In this instance, boiling would not have to be addressed for two reasons 1) distributed debris bed and 2) LOCADM.

1. In the event of a distributed bed, the beds would contain less fiber than the single bed at the core inlet. At less than 18g of fiber, these fiber beds would be more dispersed and have larger areas with little to no fiber accumulation. The areas with smaller fiber accumulation would promote flow through the debris bed and the resulting overall head loss would be less.
2. LOCADM addresses the concerns related to precipitation. LOCADM already considers the effect of boiling on fuel rods. A quantitative estimate of the effect of the fiber on deposit thickness and fuel temperature can be accounted for in LOCADM by use of a "bump-up factor" applied to the initial debris inputs. The bump-up factor is set such that total release of chemical products after 30 days is increased by the best estimate of the mass of the fiber that bypasses the sump screen. Therefore, boiling considerations regarding fiber have been adequately addressed.

## 2.20 RAI #20

### 2.20.1 Question

*In response to the NRC's earlier RAI 42, contained in Appendix H, it is stated that a guidance document is being developed to assist licensees in implementing WCAP-16793. Please provide the status of this document.*

### 2.20.2 Response

Section 10.2, page 10-3, was written with the intent to provide guidance to licensees to help implement WCAP-16793.

In addition to Section 10.2, utilities have the methodology for the calculation of the plant-specific available driving head presented in Section 2.18 to help implement WCAP-16793.

## 2.21 RAI #21

### 2.21.1 Question

*The AREVA and Westinghouse proprietary test reports indicate that the test for the Combustion Engineering designed plants was conducted at 11 gallons per minute (gpm) and 6 gpm, respectively. Please provide the basis for the difference in flow rates.*

### 2.21.2 Response

The flow rate used in the fuel assembly testing is based on the total ECCS flow rate and the number of fuel assemblies in a given plant. A review of these parameters for the CE plants that are refueled by Westinghouse and AREVA was done by each organization.

The Westinghouse CE tests were designed to be conducted at a flow rate of 6.25 gpm as this flow was high enough to bound the CE plants with Westinghouse fuel. This flow rate corresponds to an ECCS flow rate of 1300 gpm and 208 fuel assemblies.

The AREVA tests were conducted at 11 gpm as this flow rate was high enough to bound the CE plants with AREVA fuel. This flow rate corresponds to an ECCS flow rate of 1350 gpm and 133 fuel assemblies.

## 2.22 RAI #22

### 2.22.1 Question

*The testing for WCAP-16793 was based on specific intermediate spacer and mixing grids. Please explain how licensees should evaluate differences between the tested grids and evolving grid designs.*

### 2.22.2 Response

Justifying future grid designs is not a focus of this program. Utilities and fuel vendors will have to evaluate design changes for compliance with GSI-191 as new designs are implemented.

## 2.23 RAI #23

### 2.23.1 Question

*In Appendix B, page B-2, paragraph B.3.2, the figure reference in the text is Figure B-3. This appears to be an error. The figure reference apparently should be Figure B-1. Please verify.*

### 2.23.2 Response

The last sentence in Appendix B, page B-2 will be changed to read (changes are highlighted in **bold**):

> **~~The radial power distributions for the four core channels shown in Figure B-3 are displayed in Table B-1.~~ Figure B-1 represents the axial power shape and Table B-1 displays the radial power distribution of the modeled plant.**

## 2.24 RAI #24

### 2.24.1 Question

*In the Appendix B figures, please identify vertical and horizontal flows. Do the squared numbers indicate vertical paths and circled number indicate horizontal paths? Please provide better descriptions.*

### 2.24.2 Response

Yes, the vertical flow paths (channels) are designated by squares and horizontal flow paths (gaps) are designated by circles in Figures B-3, B-4, B-6, and B-7. Please note that Figure B-5 represents the one dimensional loop model, where squares represent 1-D components, i.e., pipes, pumps, etc., and the circles represent junctions between adjacent components. The first paragraph of Section B.3.2 will be changed to add clarification (changes are highlighted in **bold**):

> A plant with an existing WC/T model, downflow plant configuration, and high core power density is desired for the core blockage simulations. A three-loop downflow model plant rated at 2900 MWt was chosen. The power shape of the plant's BELOCA reference transient used for these simulations is shown in Figure B-1. **(Figures use squares to designate vertical flow paths and circles to designate horizontal flow paths.)**

## 2.25 RAI #25

### 2.25.1 Question

*In Appendix B, page B-27, paragraph B.5.1, the description of the first bulleted approach is confusing. Please confirm that all the inlet areas except channel 13 were set equal to zero. Please explain more accurately the condition analyzed. Also, the differences in the inlet flow area and the internal core flow area should be described.*

### 2.25.2 Response

As stated in the second paragraph in Section B.5.1, the base case for the calculations presented in Section B.5 is Case 2 from Section B-3. As discussed in Section B-3, Case 2 simulated an inlet flow blockage of 99.4% of the core by ramping the dimensionless loss coefficient to $10^9$ in all core channels except for the hot assembly channel, i.e., channels 10, 11, and 12, to simulate debris buildup. The ramping of the loss coefficients to $10^9$ in channels 10, 11, and 12 was maintained for the additional WC/T runs, which effectively is the same as an area reduction. Further reduction in flow area was then modeled by reducing the physical flow area at the bottom of channel 13. The discussion included in the first bullet refers to the slightly increased flow area through the adjacent channel in the core performed to maintain core flow area prior to the modeled debris buildup and to maintain WC/T modeling requirements.

The vessel model used for the WC/T simulations provided in Appendix B is consistent with the Best Estimate Large Break LOCA modeling practices. The flow area at the Section 2/3 boundary (See Figure B-3) is set the equal to the more restrictive flow area between the lower core plate and the fuel bottom nozzle. The flow area through the remainder of the active fuel height is based on the fuel assembly flow area (note that grids are modeled using loss coefficients).

## 2.26 RAI #26 & 27

### 2.26.1 Question

*26: In Section 3.3.3, on page 3-14, it is stated that "There are no significant PCT excursions" and references Figure 3-9 as evidence. However, Figure 3-9 shows a significant PCT excursion at the end of the plot. The temperature is still rising at the end of the plot. Please explain the apparent contradiction and why this excursion is acceptable.*

*27: In Section 3.3.3, on page 3-14 it is stated that in Figure 3-13 "the PCT increases until the end of the transient calculation". The temperature rise is not shown in this figure. Could the text actually refer to Figure 3-9? If not, please justify the conclusion regarding the PCT.*

### 2.26.2 Response

Section 3.3.3 should read:

*The first uniform loss coefficient run performed applied a uniform $C_D$ of 50,000 at the core inlet.* ***Figure 3-10*** *shows a comparison of the integrated core inlet flow and boil-off rate, again starting at the time of switchover from injection to recirculation from the sump. As shown, even with the increase of the loss coefficient at the inlet, the flow that enters the core is still in excess of the boil-off rate. (Note that the integrated mass flow behavior shown between t = 1200 seconds and time t = 1250 seconds of* ***Figure 3-10*** *is the result of the 30 second ramp-up of the hydraulic loss coefficient, $C_D$, to 50,000 that is initiated in the calculations at time t = 1200 seconds). The PCT is shown in* ***Figure 3-11****. There are no significant PCT excursions after the core inlet loss coefficient is increased.*

*The second uniform loss coefficient run performed applied a uniform $C_D$ of 100,000 at the core inlet.* ***Figure 3-12*** *shows a comparison of the integrated core inlet flow and boil-off rate. As shown, even with the further increase of the loss coefficient at the inlet, the flow that enters the core is still in excess of the boil-off rate. (Note that the behavior of the integrated mass flow rate of* ***Figure 3-12*** *is due to the 30 second ramp-up of the hydraulic loss coefficient, $C_D$, to 100,000 that is initiated in the calculation at time t = 1200 seconds, but extends the behavior over a slightly longer period of time.) The PCT is shown in* ***Figure 3-13****. There are no significant PCT excursions after the core inlet loss coefficient is increased.*

*The next uniform loss coefficient run performed applied a uniform $C_D$ of 1,000,000 at the core inlet.* ***Figure 3-14*** *shows a comparison of the integrated core inlet flow and boil-off rate. With the increased resistance to flow into the core specified for this case, the flow that enters the core can not match the boil-off rate. As a consequence, as shown in* ***Figure 3-15****, the PCT increases until the end of the transient calculation.*

*The results indicate that an increase in the form loss coefficient at the core inlet of up to $C_D$ = 100,000 for the limiting plant and fuel load design will allow for sufficient flow into the core to remove decay heat and provide LTCC.*

# 3.0 Supplemental Information

During operation of the emergency core cooling system (ECCS) to recirculate coolant from the containment sump, debris in the recirculating fluid that passes through the sump screen may collect throughout the fuel assemblies, causing resistance to flow through this path. The Pressurized Water Reactor Owners Group (PWROG) undertook a program to provide additional analyses, test data, and information on the effect of debris and chemical products on core cooling, documented in Reference 2. This program established generic limits on the debris mass that could bypass the sump screen and not impede long-term core cooling (LTCC).

After the publication of Reference 2, requests for additional information (Reference 1) from the NRC were transmitted to the PWROG. Additional testing was conducted to evaluate the published debris limits and it was determined the debris limits were dependent upon fuel vendor type. That is, plants using Westinghouse fuel have to meet debris limits defined for Westinghouse and plants using AREVA fuel have to meet debris limits defined for AREVA.

In the process of addressing Reference 1, Westinghouse, AREVA, and the PWROG conducted weekly phone calls with NRC staff. During these discussions, the NRC staff asked the PWROG to provide guidance for utilities operating with Westinghouse and AREVA fuel. The following sections present the requested guidance.

## 3.1 Introduction

The following sections provide guidance to define the plant-specific debris load acceptance criteria for utilities with cores employing fuel from multiple fuel vendors. This proposal uses generic conditions and fuel vendors are generically referred to as 'high debris load (HDL) fuel' and 'low debris load (LDL) fuel.' The HDL fuel has a higher debris load per fuel assembly than the LDL fuel. The full evaluation is provided in Reference 8.

## 3.2 Guidance to Determine Maximum Debris Load for Mixed Cores

Based on the expected distribution of debris, a mixed core will be assured of LTCC by imposing a fiber limit criterion that is based on the proportion of fuel assemblies from each vendor. The guidance described below ensures the majority of fuel assemblies will only see less than or equal to the maximum amount of allowable debris.

### 3.2.1 Customer Switching from HDL Fuel to LDL Fuel

| Cycle Description | Fiber Limit |
|---|---|
| Cycle 1: Full core of High Debris Load (HDL) Fuel | Meet HDL Fuel limit<br>EXAMPLE: 150 g fiber/FA |
| Cycle 2: 1 batch of Low Debris Load (LDL) Fuel & 2 batches of HDL Fuel<br><br>Batches are various sizes, so determine the limits based on the number of FA from each vendor | Meet hybrid limit calculated by= (#HDL assy)/(# tot assy) * (HDL Fuel criteria) + (#LDL assy)/(#tot assy) * (LDL Fuel criteria)<br>EXAMPLE:<br>HDL Fuel criteria = 150 g fiber/FA<br>LDL Fuel criteria = 15 g fiber/FA<br>New limit = (# HDL Fuel Assemblies/ # Total Assemblies)(150) + (# LDL Fuel Assemblies/ # Total Assemblies)(15)<br>= (129/193)(150)+(64/193)(15)<br>**= 105 g fiber/FA** |
| Cycle 3: 2 batches of LDL fuel & 1 batch of HDL fuel | Meet LDL Fuel limit<br>EXAMPLE: 15 g fiber/FA |
| Cycle 4: Full core of LDL Fuel | Meet LDL Fuel limit |
| *Full Core is defined as a core with at least 90% of fuel from one vendor.* | |

### 3.2.2 Customer Switching from LDL Fuel to HDL Fuel

| Cycle Description | Fiber Limit |
|---|---|
| Cycle 1: Full core of LDL Fuel | Meet LDL Fuel limit<br>EXAMPLE: 15 g fiber/FA |
| Cycle 2: 1 batch of HDL Fuel & 2 batches of LDL Fuel<br><br>Batches are various sizes, so determine the limits based on the number of FA from each vendor | Meet LDL limit |
| Cycle 3: 2 batches of HDL fuel & 1 batch of LDL fuel | Meet hybrid limit calculated by= (#HDL assy)/(# tot assy) * (HDL criteria) + (#LDL assy)/(#tot assy) * (LDL criteria)<br>EXAMPLE:<br>HDL criteria = 150 g fiber/FA<br>LDL criteria = 15 g fiber/FA<br>New limit = (# HDL Assemblies/ # Total Assemblies)(150) + (# LDL Assemblies/ # Total Assemblies)(15)<br>= (129/193)(150)+(64/193)(15)<br>**= 105 g fiber/FA** |
| Cycle 4: Full core of HDL Fuel | Meet HDL Fuel limit |
| *Full Core is defined as a core with at least 90% of fuel from one vendor.* | |

## 4.0 References

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