March 7, 2008

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR (PWR) OWNERS GROUP (PWROG) TOPICAL REPORT (TR) WCAP-16793-NP, REVISION 0, "EVALUATION OF LONG-TERM COOLING CONSIDERING PARTICULATE, FIBROUS AND CHEMICAL DEBRIS IN THE RECIRCULATING FLUID" (TAC NO. MD5891)

Dear Mr. Bischoff:

By letter dated June 4, 2007, as supplemented by letter dated January 17, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080220258), the PWROG submitted TR WCAP-16793-NP, Revision 0, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid" (ADAMS Accession No. ML071580139), to the U.S. Nuclear Regulatory Commission (NRC) staff for review. The supplement provided additional information that clarified TR WCAP-16793-NP, Revision 0, but did not expand the scope of the TR as originally submitted. Enclosed for PWROG review and comment is a copy of the NRC staff's draft safety evaluation (SE) for the TR.

TR WCAP-16793-NP, Revision 0, is intended to provide guidance and a consistent approach for PWR licensees to evaluate the long-term cooling impact of sump debris and chemicals on the performance of their emergency core cooling following a postulated loss-of-coolant accident. The overall issue is being driven by Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance," and the subsequent NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors."

A licensee may reference TR WCAP-16793-NP, Revision 0, to perform an assessment of the impact of particulate, fibrous, and chemical debris on nuclear fuel and the ability to maintain the core in a coolable geometry in a post-accident condition, subject to the conditions and limitations in the enclosure. In addition, a licensee referencing the TR will need to determine its own specific sump particulate, fibrous, and chemical debris mixture and sump screen size in order to initiate the evaluation. The TR provides information on a common evaluation method and acceptance criteria for the components of Westinghouse, Combustion Engineering, and Babcock and Wilcox PWR nuclear steam supply system designs.

Twenty working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The NRC staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Holly D. Cruz at 301-415-1053.

Sincerely,

/RA/

Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Draft SE

cc w/encl: Mr. James A. Gresham, Manager
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ADAMS ACCESSION NO. ML080600876 *No major changes to SE input. NRR-043

OFFICE	PSPB/PM	PSPB/LA	SSIB/BC*	PSPB/BC
NAME	HCruz	DBaxley	MScott	SRosenberg
DATE	2/27/08	3/5/08	2/12/08	3/7/08

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16793-NP, REVISION 0

"EVALUATION OF LONG-TERM COOLING CONSIDERING PARTICULATE,

FIBROUS AND CHEMICAL DEBRIS IN THE RECIRCULATING FLUID"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

TAC NO. MD5891

1.0 INTRODUCTION AND BACKGROUND

Topical Report (TR) WCAP-16793-NP, Revision 0 (Reference 1), describes the results from a program by the Pressurized Water Reactor (PWR) Owners Group (PWROG) to evaluate the effect of debris and chemical products on long-term core cooling following a postulated loss-of-coolant accident (LOCA). Following a large LOCA, long-term core cooling will be accomplished by the recirculation of the water spilled to the containment building sump back into the reactor. It is postulated that debris and chemicals which might collect in this spilled water as it flows over the containment building surfaces might have a detrimental effect on core cooling after it is pumped back into the reactor. The PWROG has addressed this issue by providing evaluations in the following topical areas:

1. Evaluation of fuel cladding temperature response to blockage at the inlet of the core.

2. Evaluation of fuel cladding temperature response to local blockages or scale formation on the fuel cladding surface.

3. Evaluation of chemical effects in the core region, including potential for plate-out on fuel cladding.

Included are evaluations intended to be bounding for a large number of plants, sensitivity evaluations and methodology with which plant-specific calculations can be performed. The TR is intended to provide methodology applicable to all operating PWRs.

The PWROG selected two criterion as the acceptance basis for long-term core cooling to address Generic Safety Issue (GSI)-191. The acceptance basis is designed to demonstrate 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 following criteria were given:

1. Decay Heat Removal/Fuel Clad Oxidation: Maximum cladding temperatures maintained during periods when the core is covered will not exceed a core average clad temperature

of 800 ºF. In Requests for Additional Information (RAIs) during the review of TR WCAP-16793, the U.S. Nuclear Regulatory Commission (NRC) staff requested additional clarification concerning this criterion. As discussed in Section 3.6 of this SE, this issue was successfully resolved.

2. Boric Acid Precipitation and the Chemistry Effects of Debris: A core-flushing flow will be established that is sufficient to prevent the calculated maximum boric acid concentration in the core region from exceeding the precipitation limit. The NRC staff RAIs requested that: (1) the PWROG expand this criterion to include chemical species other than boric acid which might be washed into the core from the containment sump and (2) the effect of debris blockage on the reactor vessel water volume available for boric acid and chemical dilution be included in the evaluation. These questions were subsequently resolved as discussed in Sections 3.2 and 3.5 of this Safety Evaluation (SE).

The details of the review of the TR WCAP-16793 topics, the NRC staff RAIs, the resolution of the RAIs, and conditions and limitations on use of the TR are described in this SE.

2.0 REGULATORY EVALUATION

Generic Letter (GL) 2004-02 calls for holders of operating licenses for PWRs to perform evaluations of the emergency core cooling system (ECCS) and the containment spray recirculation functions. These evaluations are to include the potential for debris blockage at flow restrictions within the ECCS recirculation flow path downstream of the sump screen, including potential blockage at fuel assembly inlet debris screens. Other potential flow restrictions are the fuel assembly inlet debris screens and the spacer grids within the fuel assemblies. Debris blockage at such flow restrictions has the potential to impede or prevent the recirculation of coolant to the reactor core, potentially leading to inadequate long-term core cooling.

The acceptance criteria for the performance of a nuclear reactor core following a LOCA are found in Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR). The acceptance criterion dealing with the long-term cooling phase of the accident recovery is as follows:

> Long-term cooling: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

At the request of the PWROG, the NRC staff provided clarification of: (1) regulatory requirements and acceptance criteria for long-term core cooling once the core has quenched and reflooded and (2) the mission time that should be used in evaluating debris ingestion effects on the reactor fuel. The NRC staff provided these clarifications in a letter dated August 16, 2006 (Reference 2). To summarize the NRC staff's response to the PWROG, long-term cooling capability must be provided despite potential challenges from chemical effects (e.g., boron precipitation) or physical effects (debris), as demonstrated by no significant increase in calculated peak cladding temperature (PCT). After quench and re-flood, moderate increases in cladding temperature could be considered, if justified. In addition, adequate core cooling performance during the ECCS mission time is demonstrated when bulk and local temperatures are shown to be stable or continuously decreasing with the additional assurance that any debris entrained in the cooling water supply would not be capable of affecting the stable heat removal mechanism due to sump screen clogging or downstream effects.

3.0 TECHNICAL EVALUATION

Following a large LOCA, water from the ECCS will initially be injected into the reactor system from stored water supplies. Before these supplies are exhausted, suction to the ECCS pumps will be switched to the containment sump so that the coolant spilled from the reactor system and collected from the containment spray will be recirculated back into the reactor system to provide for continued long-term cooling. During the recirculation of coolant after a LOCA, coolant will be passed through the sump screens to the ECCS pumps and heat exchangers, and then returned to the reactor core. During the recirculation process, depending on the break location, some of the coolant will be spilled immediately on to the containment floor through the break. The rest will flow into the reactor vessel and eventually spill out onto the containment floor and be directed into the containment sump to be recirculated back through the ECCS. There are two separate cases for LOCA break location, and two locations of core injection depending on whether the plant is a two (upper plenum injection (UPI)) or three/four loop (cold leg injection) design.

Case 1 – Hot Leg Break

For the majority of PWRs, the ECCS pumps would be aligned to inject stored borated water into the reactor cold legs or directly into the reactor vessel downcomer in the event of a LOCA. Once the stored water source becomes exhausted, the ECCS pumps would be aligned to take suction from the containment sump for continued recirculation of coolant so as to provide for long-term core cooling. In the event of a hot-leg break, the ECCS water would be forced through the reactor core toward the break. Core flow, including a small amount of core bypass flow, during the long-term cooling period would be equal to the total ECCS flow. If all ECCS pumps were assumed to operate, ECCS flow into the reactor system through the reactor vessel and into the core would be maximized. The maximum flow condition should be evaluated since it provides the greatest potential for debris transport to the reactor core and subsequent lodging within flow restrictions.

Initial ECCS flow for UPI plants is a combination of injection into the cold legs and the upper plenum. At the time of recirculation switch-over, the cold leg flow is secured. In the case of a large hot-leg break, the ECCS flow directed into the reactor upper plenum will flow into the core from above at a rate needed to replenish that boiled away. Any excess will be spilled out of the break. The long-term cooling period following a large hot leg break represents a minimum core flow condition for a UPI plant. With water only being added above the core, the NRC staff expects that the water in the reactor system cold leg piping will be stagnant since for flow to be established through the cold legs, water would have to be pushed over the tops of the U-bends of the steam generator (SG) tubing. Both excess ECCS flow and steam from the core would be expected to flow out of the broken hot leg because of its lower elevation relative to the top of the SG tubes. Without a net flow through the core, boiling in the core would continue for an indefinite period, causing debris and chemicals to be concentrated.

Case 2 – Cold Leg Break

For the majority of PWRs, ECCS flows into the reactor cold legs or directly to the vessel downcomer. Following a cold-leg break, water would flow to the core from the intact loop cold legs. By procedure, flow into the reactor core will eventually be limited to the replacement of that boiled away. The excess will be spilled out of the break, including any water injected into the intact cold legs which will flow around the upper elevations of the downcomer to reach the break without passing through the core. The long-term cooling period following a large cold leg break represents a minimum core flow condition. Core blockage by debris under these conditions would add to the resistance which must be overcome for the ECCS water to reach the core and lead to additional spillage from the break.

For two-loop Westinghouse UPI plants, the low pressure injection pumps inject directly into the upper plenum and the cold leg. For a cold-leg break, the ECCS flow would be forced through the reactor core toward the break. During the long-term cooling period, core flow, including a small amount of core bypass flow, would be equal to the total ECCS flow. If both low pressure ECCS pumps were assumed to operate, ECCS flow into the reactor system through the reactor vessel and into the core would be maximized. The maximum flow condition should be evaluated since it provides the greatest potential for debris transport to the reactor core and subsequent lodging within flow restrictions.

Following a large cold-leg break at most operating PWRs or a large hot-leg break at a UPI plant, continued boiling in the core will act to concentrate the debris and chemicals in the water within the core coolant channels. Chemical reaction of the debris with the containment spray buffering agents and boric acid from the ECCS water in the presence of the core radiation field could change the chemical and physical nature of the mixture. Heat transfer could be affected by direct plate-out of debris on the fuel rods or by accumulation of material within the fuel element spacer grids.

At a meeting with the PWROG on February 7, 2007, and at an Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting May 15, 2007, the NRC staff provided the PWROG a list of considerations which should be addressed to resolve GSI-191 for the reactor core:

1. Methodology should account for differences in PWR reactor coolant system (RCS) and ECCS designs. Examples include:
    - Combustion Engineering (CE) plants with smaller recirculation flows may produce extended core boiling long after hot leg recirculation begins. The extended boiling period may impact concentration of debris in core, plate-out, etc.

    - Use of pressurizer spray nozzles for hot-leg recirculation should be evaluated for the potential of clogging with debris.

    - UPI plants with no cold-leg recirculation flow may have no means of flushing the core following a large hot-leg LOCA and may need special consideration.

2. Hot spots may be produced from debris trapped by swelled and ruptured cladding.
   - Debris may collect in the restricted channels caused by clad swelling and at the rough edges at rupture locations.
   - Full-length emergency cooling transfer (FLECHT) tests have shown that swelled and ruptured cladding may not detrimentally affect the cladding temperature profile. The FLECHT tests did not include post-LOCA debris.
3. Long-term core boiling effects on debris and chemical concentrations in the core should be accounted for.
   - The evaluations should be similar to post-LOCA boric acid precipitation evaluations.
   - They should account for the change in water volume available to mix with constituents concentrated by the core from debris accumulation.
   - Partial blockage of the core creates alternate circulation patterns within the reactor vessel and will affect the concentration analysis.
   - Will the solubility limits be exceeded for any of the material dissolved in the coolant that is being concentrated by boiling in the core?
   - The lower plenum is often credited as part of the mixing volume to determine the timing for precipitation. Partial filling of the lower plenum by debris may reduce its effectiveness.
4. Debris and chemicals which might be trapped behind spacer grids could potentially affect heat transfer from the fuel rods and should be evaluated.
   - Analyses show that a partially filled spacer grid produces only a moderate cladding temperature increase even if only axial conduction down the cladding is considered.
   - Similar analyses show that a completely filled spacer grid with only axial conduction will result in unacceptable temperatures.
   - A physical basis for determining to what extent the spacer grids can trap debris, and the ability for the debris to block heat transfer needs to be provided.
   - The evaluation needs to include all the chemical and physical processes which may occur in the core during the long term cooling period.
5. Consideration should be included for plating out of debris and/or chemicals on the fuel rods during long-term boiling.
   - Long-term boiling in the core following a large LOCA may last for several weeks for some designs depending on the ECCS flow and core inlet temperature.
   - The concentration of materials in the core, and the potential for plate out on the fuel rods (boiler scale) from this material should be determined.

- When the composition and thickness of the boiler scale has been determined, the effect on fuel rod heat transfer should be evaluated.

6. The PWROG needs to address whether high concentrations of debris and chemicals in the core from long-term boiling can affect the natural circulation elevation head which causes coolant to enter the core.
   - For a large cold-leg break, the density difference between the core and the downcomer determines the hydrostatic driving head, and consequently the flow rate into the core.

   - As boiling continues, a high concentration of debris and chemicals in the core may increase the core density and reduce the flow into the core.

7. If hot spots are found to occur, the PWROG should address cladding embrittlement. Applicable experimental data for the calculated condition and type of the cladding should be presented to demonstrate that a coolable geometry is maintained.

Based on NRC staff review, as discussed below, the above considerations were satisfactorily addressed in the TR, and/or the responses to the NRC staff's RAIs.

The following evaluations and conclusions of TR WCAP-16793, Revision 0, with the appropriate technical report sections, are discussed in this SE.

1. Adequate flow to remove decay heat will continue to reach the core even with debris from the sump reaching the reactor coolant system and core (WCAP-16793-NP, Revision 0, Section 2.1, SE Section 3.1).

2. Decay heat will continue to be removed even with debris collection at the fuel assembly spacer grids (WCAP-16793-NP, Revision 0, Section 2.2, SE Section 3.6).

3. Fibrous debris, should it enter the core region, will not tightly adhere to the surface of fuel cladding (WCAP-16793-NP, Revision 0, Section 2.3, SE Section 3.6).

4. Using an extension of the chemical effects method developed in WCAP-16530-NP, to predict chemical deposition on fuel cladding, a sample calculation using large debris loadings of fiberglass and calcium silicate was performed (WCAP-16793-NP, Revision 0, Section 2.4, SE Section 3.5). The result is a maximum deposition thickness of 257 microns (10 mils) and a maximum fuel cladding temperature of 324 ºF.

5. The three categories of protective coatings used inside reactor containment buildings have been evaluated to have negligible effect on the generation of precipitate (WCAP-16793-NP, Revision 0, Section 2.5, SE Section 3.5).

6. As blockage of the core will not occur, the mixing volumes assumed for the current licensing basis boric acid dilution evaluations are not affected by debris and chemical products transported into the reactor coolant system from the containment sump (WCAP-16793-NP, Revision 0, Section 2.6, SE Section 3.2).

7. Westinghouse two-loop PWRs which incorporate UPI as part of their ECCS will be provided adequate flow to remove decay heat (WCAP-16793-NP, Revision 0, Section 2.7, SE Section 3.3).

Discussion of individual licensee application of WCAP-16793, Revision 0, is found in Section 3.7 of this SE.

3.1 Blockage at the Core Inlet (TR WCAP-16793, Revision 0, Section 2.1)

For most PWRs (other than UPI plants that are addressed in Section 3.3 below), the primary location for safety injection is into the reactor cold legs or into the reactor vessel downcomer. For these plants, a large cold-leg break provides the minimum driving head. Following a large cold-leg break, the elevation head between the downcomer and the core provides the driving potential for coolant to enter the core. A mat of fiber and debris at the core inlet will act to reduce the flow rate into the core. The PWROG performed analyses using WCOBRA/TRAC (Reference 1, Appendix B) which demonstrated that with a considerable amount of core inlet blockage, core cooling could be maintained following a large cold-leg break. The NRC staff performed similar informal, confirmatory calculations using the RELAP5 and TRACE computer codes which also showed a considerable potential for core cooling in the presence of a substantial amount of core inlet blockage.

As debris and chemicals are concentrated within the core by the boiling process, the density of the fluid in the core will increase. The increase in density will act to retard core flow. This effect was not accounted for in the RELAP5 and TRACE analyses performed by the NRC staff. The NRC staff requested in its RAIs that the PWROG provide an evaluation of the effect of increased fluid density in the core on the results from the WCOBRA/TRAC analysis of core blockage. The PWROG responded (Reference 3) that an increase in core fluid density will result in an unstable configuration leading to natural circulation flow patterns between the core and the lower density in the lower plenum, limiting the density build-up. Results from the BACCHUS tests were submitted in Reference 4, in support of the Waterford power uprate. The PWROG stated in Reference 3, that the effects of an increase in density in the core due to concentration of debris and chemicals would not be expected to affect the main conclusions reached in the WCOBRA/TRAC study. However, WCOBRA/TRAC does not have the modeling capability to demonstrate this quantitatively. Actual densities would be plant dependent.

In addition, in recent extended power uprate submittals it has been shown that boric acid solutions approaching the solubility limit would have a density increase on the order of 10 percent, based on information provided in EPRI NP-5558. An increase of this order of magnitude would not be expected to offset the available driving head predicted by the WCOBRA/TRAC calculations presented in WCAP-16793, Appendix B. Even though those calculations did not include the effects of debris buildup, such buildup would be expected to be gradual and not appreciably alter the execution of adequate core flow over the ECCS mission time.

Reference 1, Section 2.1, states that actual core inlet debris bed will be porous and allow adequate cooling flow to enter the core. It refers to demonstrations of a mock fuel assembly performed at Continuum Dynamics, Inc. (CDI) in Ewing, New Jersey, and previously published work on debris transport during recirculation (Reference 5). The demonstrations involved addition of fiber, followed by particulate, to a lower section of an unheated mock fuel assembly under simulated flow conditions to observe the potential for debris transport, capture, and bed formation through the mock fuel assembly. Reference 6 (observations of the NRC staff observing the demonstration) states that the flow velocity in the demonstrations had to be maintained higher than would be expected following a LOCA to achieve transport of the debris into the fuel assembly. Fibrous debris used in the demonstrations consisted of output from previous active strainer tests. Particulates were simulated by the same materials used for the active strainer testing. The following observations were noted by NRC staff representatives who were present:

1. As fibers were loaded into the test rig, fibers would be captured, as designed, by the lower nozzle block and simulated core structures. A small amount of fiber was transported in the water stream to the upper parts of the mock up assembly and pass completely through the core. These were captured on subsequent passes. All fibers, no matter how short, appeared to be susceptible to capture at the test assembly entrance. This observation is contrary to statements made in Section 2.2 of Reference 1, which indicates that short fibers will pass through reactor fuel without being captured.

2. Most fibers were captured at the lower fuel nozzle location. As more fiber was added, a fiber bed began to form at the lower nozzle. The fiber bed grew from the outside edges towards the center and appeared to become uniform.

3. As particulates were added, they immediately distributed with the flow of the fluid throughout the test fixture. Differences in color between the fiber and particulate showed that, over time, the inlet nozzle fiber bed efficiently captured the added particulates.

4. The bed formed at the lower core plate/fuel nozzle did not develop a large enough pressure drop to block flow.

5. Addition of extra fiber and extra particulate (beyond the formation of a complete fiber bed at the bottom nozzle) did not change the observed behaviors significantly.

The type, amounts, and characteristics of the fiber and particulate matter were not described in Reference 6. The specific flow velocities that existed in the mock up fuel bundle inlet were not given nor were the resulting pressure drop. In order for a licensee to relate a postulated post-LOCA core inlet debris bed to the tests described in Reference 6, this information was needed. In response to an RAI from the NRC staff, the PWROG provided, in Reference 3, additional test details and description as follows:

1. Screen Design: Active sump screen, plow and blade design.
2. Core Simulation: Single part length assembly with CE bottom nozzle, guardian grid.
3. Core Flow: Six gallons per minute, maximum plant fuel grid flow rate.

A schematic of the test loop is shown in Figure 1, below. Test instrumentation consisted of differential pressure measurement across the test bundle. The test bundle consisted of the simulated core support plate, bottom nozzle, debris-capturing guardian grid, an intermediate support grid, simulated fuel rods (3/8-inch plastic rod) with simulated control rods, and an upper support grid.

The fibrous debris used in the test came from the fiber that bypassed the active sump screen in the bypass testing. The fibrous debris was shredded by what has been described as a leaf shredder (see Reference 6) for the bypass tests. The particulate debris consisted of a mixture of insulation, dust, and dirt that was determined to be prototypical for a plant.

The PWROG stated that all quality assurance related activities related to test planning and execution were performed consistent with the CDI Quality Assurance Program in compliance with the reporting requirements of 10 CFR Part 21.

The key observations reported by the PWROG in Reference 3 were consistent with those of the NRC staff noted above.

Measured head loss as a function of the fiber mass of debris for one fuel assembly for the tests is shown in Figure 2. The PWROG stated in Reference 3, that a bounding head loss, based on tests performed assuming collection of 21.7 ft^3 of fibrous debris and 1388.8 lbm of particulate debris at the entrance to the core, would be expected to be about 10.2 inches of water, or an increase in pressure drop of 0.37 psi at the core inlet. The NRC staff did not review the test results in detail but believes them to be reasonable because of the observation of little debris capture on the fuel inlet grid. It was further noted that the WCAP-16793, Appendix B, WCOBRA/TRAC bounding analysis, with an assumed flow blockage of 99.4 percent, demonstrated adequate flow rate to remove decay heat. A flow blockage of 99.4 percent would result in a head loss substantially greater than 10.2 inches of water. Thus, a plant with a calculated head loss of 10.2 inches of water would be bounded by the TR results and would be able to conclude it would have adequate core cooling.

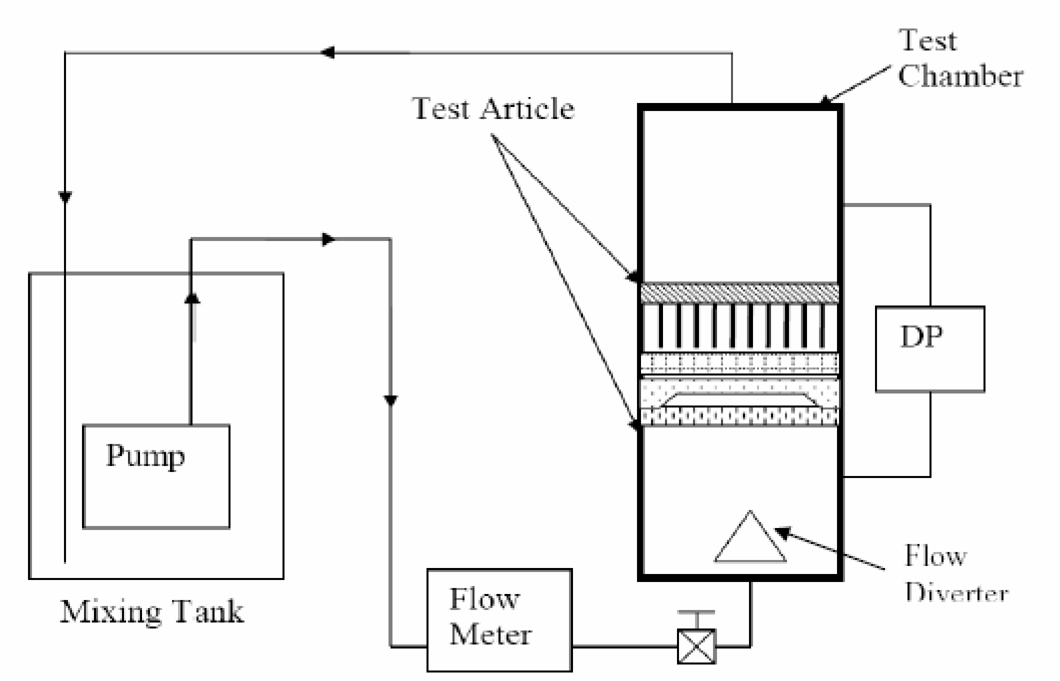

Figure 1
Schematic of Fuel Assembly Fibrous Debris Capture Test Loop

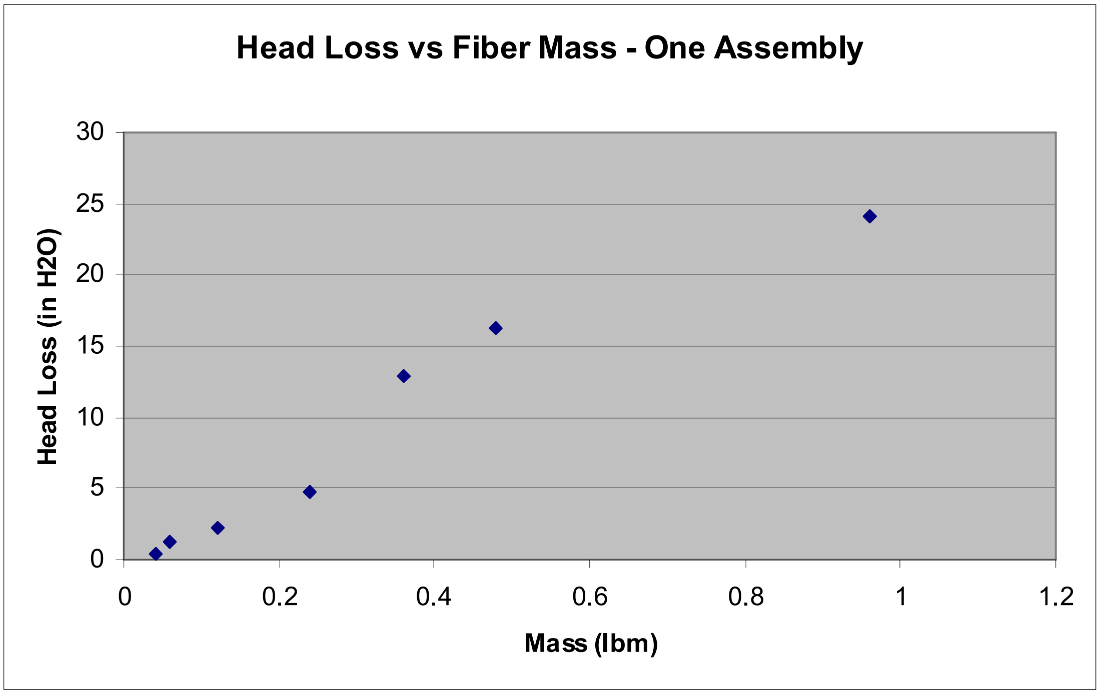

Figure 2
Head Loss versus Debris Mass

The NRC staff performed confirmatory analyses using the RELAP5 and TRACE codes and obtained results consistent with the PWROG WCOBRA/TRAC results. All of these analyses indicate that significantly greater blockage than observed in the CDI demonstration would not preclude adequate coolant reaching the fuel to remove the decay heat. Figure 5, in Section 3.7 of this SE, is a PWROG submitted curve of the flow rate needed to match the boil-off rate in a Westinghouse 4-loop PWR. From this figure, it can be concluded that following the postulated LOCA, there is an exponential decrease in required coolant flow with time, thus requiring greater inlet blockage to inhibit adequate core cooling.

The NRC staff accepts the position of the PWROG, based on review of the PWROG submittal and confirmatory analyses performed by the NRC staff, all of which demonstrate that there are very large margins between the amount of blockage that could occur (based on the observations discussed above) and the blockage that would be required to degrade the coolant flow to the point that the decay heat could not be adequately removed. Thus, although only limited test results were provided, and the testing was not conducted using hot or borated water, the NRC staff concludes, from the large margins, that acceptable performance is likely for all fuel types and at expected post-LOCA case conditions. However, plant-specific evaluations must verify the applicability of the blockage conclusions to their fuel designs.

Following a large hot-leg break for plants which inject water into the cold legs or directly into the downcomer, the discharge pressure of the ECCS pumps will be available to force water into the core. Should the core inlet become blocked by debris during the long-term core cooling period, the pressure of the ECCS pumps might be sufficient to pump water through the steam generator tubes, back through the hot legs, to cool the fuel from the top of the reactor core. Licensees seeking to credit this flow path would need to provide a supporting analysis demonstrating that the pumps have sufficient head to overcome the flow path resistance and can provide adequate coolant through this path. In addition, alternate flow paths for ECCS water to reach the core might be available at some plants including holes drilled through the core baffle plates. Should a licensee choose to take credit for this flow path, it will have to demonstrate that the flow holes will not be become blocked with debris during a LOCA and that the flow path would be effective.

The resistance to flow resulting from a mat of material at the core entrance will be a function of the composition of the mat, its thickness, and the incoming flow velocity. NUREG/CR-6224 contains methodology for calculating the flow resistance from mats of mixed fiber and debris. Provided that the characteristics of a postulated core inlet mat can be shown to be similar to those of the tests from which data was taken, licensees determining that a mat of debris and fiber might form at the core inlet following a LOCA might utilize the methodology of NUREG/CR-6224 to calculate the flow resistance. However, the PWROG has stated (Reference 3) that the NUREG/CR-6224 head loss coefficient is not applicable to core inlet conditions because it was developed for a different situation than that expected to occur for collection of fibrous debris on grids. The PWROG further supports its position pointing out:

1. Replacement sump screen testing has shown that, due to the small hole size used for the replacement sump screens, the fibrous debris bypassed through replacement sump screens is much shorter than the fibrous debris that was used in the testing performed to support the development of the NUREG/CR-6224 head loss correlation. The shorter fibers provide for a different fiber bed morphology than what was used for the development of the NUREG/CR-6224 correlation.

2. The flow in the demonstration test was upward with gravity acting to pull the fiber bed down and away from the bottom of the fuel. The same condition would apply to an actual core post-LOCA. For the testing performed to support the NUREG/CR-6224 correlation, the fibrous bed was formed on the top of a mesh or perforated screen with the flow downward through the fibrous bed and the screen, which tended to compress the fiber bed.

3. Similarly, gravity worked on particulates that were caught in the fibrous bed of the demonstration testing, tending to pull the fibrous bed apart, whereas, for the testing performed to support the development of the NUREG/CR-6224 head loss correlation, particulates were trapped on top of the fibrous debris bed and gravity, like the flow, worked to compress the fibrous debris bed.

4. The NUREG/CR-6224 head loss correlation was developed from data collected using a vertical loop test facility with a small diameter flow channel in which all of the flow was directed downward through a fixed, predetermined debris bed. In a reactor, the flow area is large and the flow patterns sufficiently varied that the uniform directional flow conditions of the NUREG/CR-6224 test do not apply.

5. Finally, the NUREG/CR-6224 correlation was developed from test data for velocities ranging from 0.15 ft/sec to 1.5 ft/sec (NUREG/CR-6224, Section 6.4). As shown in

Table 1 below, the liquid velocities in the core are at or below the bottom range of the velocities used in developing the NUREG/CR correlation. At lower velocities associated with PWRs, there would be less compaction of any fiber bed.

The PWROG provided in Reference 3, the information given in Table 1, to demonstrate representative velocities expected to occur during long-term cooling.

Table 1

Representative Core Velocities at Time of Initiation of Recirculation from the Reactor Containment Building Sump			
NSSS Design	**Break Location**	**ECCS Operation**	**Core Velocity**
B&W, CE and W 3- and 4-Loop Plants	Cold-leg Break	1 or 2 trains	0.10 ft/sec
	Hot-leg Break	1 train	0.20 ft/sec
		2 trains	0.40 ft/sec
W 2-Loop Plants	Cold-leg Break	Max. UPI flow	0.10 ft/sec
	Hot-leg Break	Max UPI flow	N/A

The NRC staff concludes that NUREG/CR-6224 may be used to calculate head loss across a debris mat at the core inlet because the approach is conservative based on the factors cited in Reference 3. However, the approach presented in the TR is also conservative and is acceptable. The NRC staff also agrees with the TR WCAP-16793, Revision 0, Section 2.1 concluding paragraph that there is adequate flow to remove decay heat with the amount of debris demonstrated to be transported from the sump to the reactor core. Analyses have shown that there is large margin between the amount of debris expected to be transported and that necessary to prevent adequate core cooling. The NRC staff has performed independent calculations that confirm core cooling capability.

The PWROG made three observations in Section 2.1 of the TR regarding strainer bypass:

1. The amount of fiber that passes through the replacement screens is small, typically on the order of about 1 ft^3 per 1000 ft^2 of sump screen area or less.

2. Some bypass testing has shown that the length of the fibrous debris that passes through these replacement screens is short, the longest being reported in the range of 2000 to 2500 microns (0.08 to 0.10 inches).

3. The largest particle that can pass through the screens is dependent upon the hole size in the sump screen, characteristically on the order of 0.10 inches.

The PWROG has stated that licensees should either demonstrate that previously performed bypass testing is applicable to their plant-specific conditions, or perform their own plant-specific testing. The NRC staff agrees with this stated position.

3.2 Boric Acid Precipitation (TR WCAP-16793, Revision 0, Section 2.6)

Licensing basis boric acid precipitation analyses have been performed for various licensees, and reviewed and approved by the NRC. Typically, the cold leg break is the limiting event for boric acid precipitation. Combining the limiting event for debris accumulation with the limiting event for boric acid precipitation results is a more restrictive event. The PWROG noted three recent licensee submittals (listed in Table 2) that address boric acid precipitations that span the Westinghouse designs.

WCAP-16793-NP, Revision 0, Section 2.6, contains qualitative evaluations of the effects of suspended and settled debris on mixing volume in the vessel, the effect of core inlet blockage on the mixing volume, and the effect of blockage on alternate core cooling paths. In all cases, the effects were found to be insignificant. The NRC staff reviewed this information and concluded that the generic approach presented is conservative and acceptable.

Table 2

Plant Type	ADAMS Accession Number
Westinghouse 2-Loop PWR	ML060180262
Westinghouse 3-Loop PWR	ML053290133
Westinghouse 4-Loop PWR	ML072000400

The NRC staff notes that existing plant analyses showing adequate dilution of boric acid during the long-term cooling period have not considered core inlet blockage, nor have they considered reductions in mixing volume due to the presence of debris. Licensees referencing TR WCAP-16793-NP, Revision 0, will need to evaluate and verify that possible core blockage from debris will not invalidate the existing post-LOCA boric acid control analysis for the plant. The concern is not dilution itself but rather the potential for plate-out which could occur from inadequate dilution.

In Reference 7, Westinghouse has recommended that licensees with Westinghouse designs calculate the amount of debris in the lower plenum based on the limiting large break LOCA and maximum reactor vessel throughput scenario, and compare that with the volume of the lower plenum. Thus, the licensees will obtain a minimum available lower plenum volume to be considered as part of the boric acid-coolant mixing volume. The NRC staff accepts this approach to predicting the minimum volume available for boric acid mixing.

3.3 UPI Plants (TR WCAP-16793, Revision 0, Section 2.7)

Following a large cold-leg break at an UPI plant, ECCS water will flow from the upper plenum into the core and out of the cold leg break. Blockage at the core inlet might occur as the ECCS water flows through the core and deposits debris at the fuel inlet strainers. Even if the core inlet were to become blocked, cooling might still be available from countercurrent flow from the top. However, without a path for ECCS to flow through the core, it was unclear to the NRC staff that adequate boric acid dilution would be maintained. In Reference 3, the PWROG attempted to address the NRC staff concern by referring to the test described in Section 3.1 of this SE. The PWROG stated that the test demonstrated that flow of the recirculating coolant/debris mixture would not result in blockage of the core inlet, and when there is a reduction in flow rate the accumulated debris would drop to the lower plenum.

However, the NRC staff noted that the test described did not use flow introduced from above the core, nor did it use a full-length fuel assembly. If flow is introduced from above, debris would potentially be able to collect on the upper fuel hold down, the spacer grids, and ultimately the lower core support. If flow is decreased, the debris would still settle out on the same locations. Thus, the test configuration did not address the issue being raised.

All operating UPI plants have submitted boric acid control analyses. The NRC staff has approved the analyses. However, the approved analyses do not take into consideration effects of debris introduction and accumulation. While the NRC staff agrees that the test performed by the PWROG at CDI demonstrates that limited debris would be carried into the core through the lower inlet, the NRC staff does not agree that the test is an applicable demonstration of debris accumulation for injection into the upper plenum region. Individual UPI plants will need to analyze boric acid dilution/concentration in the presence of injected debris for a cold-leg break LOCA.

For UPI plants once recirculation from the containment sump to the reactor system has been established, flow into the cold legs may be terminated so that ECCS flow will only enter the core from the top. Following a large hot-leg break at such a plant, ECCS flow from the upper plenum would still be available to cool the core from the top by countercurrent flow. As with a postulated cold leg break at a UPI plant with core inlet blockage, it was unclear to the NRC staff that adequate boric acid dilution would be maintained. In addition, debris might continually be added to the core and not adequately removed, thus adversely affecting core cooling. The PWROG responded (Reference 3) to the NRC staff concern regarding termination of cold-leg injection for hot-leg breaks in UPI plants noting that TR WCAP-16793-NP, Revision 0, will be corrected in the "A" version of the TR to indicate that the licensing basis for Westinghouse two-loop PWRs is for the recirculation flow to be provided through the UPI ports with the cold-leg flow secured. Therefore the NRC staff concern regarding blockage of the core inlet from debris induced through the cold legs is resolved.

The UPI injection nozzles are located 180 degrees apart and inject at a flow rate of 600 gpm, or a jet velocity of 15.3 ft/sec. The NRC staff agrees that this velocity will prevent stagnation in the upper plenum. However, the NRC staff is not convinced this will result in debris being carried out the hot-leg in the event of a hot-leg break and, thus, prevent debris buildup in the lower core and lower plenum.

As discussed above, the NRC staff does not fully agree with the statements made by the PWROG regarding debris transport and accumulation in upper plenum injection design plants. However, due to the large margins between the amount of blockage that could occur (based on the lack of test data that demonstrate that debris will be carried into the upper plenum and exit without deposition on the fuel bundles) and the amount of blockage necessary to inhibit adequate decay heat removal, the NRC staff concludes that the methods presented in the TR are acceptable for plant-specific analyses of UPI plants.

3.4 Fuel Swelling and Blockage

Following a large break LOCA, many of the fuel rods in the core may swell and rupture leaving sharp edges at the rupture locations and a diminished channel flow area. Debris may collect in the restricted channels and at the rough edges of the rupture locations. Swelling and rupture of the fuel rod cladding during design basis LOCAs is one of the phenomena which licensees are required to evaluate under Appendix K to 10 CFR Part 50 of the Commission's regulations. This subject was not addressed in the TR. The NRC staff requested that the PWROG evaluate the possibility of excessive blockage being produced by the combination of swelling and rupture and debris collection. Such blockage might produce the occurrence of the hot spots above the blockage location.

The PWROG stated (Reference 3) that based on work performed and reported in Reference 8, only about 10 percent of the fuel rods in the core would experience cladding rupture. Therefore, they conclude, “wide-spread blockage due to swelling and rupture would not be expected in a large break LOCA scenario.” Accumulation of significant debris in the balloon and burst region would have a low probability of occurrence. While that is true for the three-loop and four-loop plants, the two-loop UPI plants have significantly different flow patterns during re-flood. The general flow pattern is for the upper plenum to drain into the lower power regions of the core, while the hotter regions are cooled by a bottom-up flow. Thus, the UPI plants are expected to encounter a circulation pattern with both up flow and down flow. This would make the accumulation of debris in the hotter regions of the core less likely. The NRC staff reviewed the Reference 3 evaluation conclusion that the blockage due to swelled or burst fuel cladding will not cause unacceptable core heat-up (PCT > 800 ºF). The NRC staff concluded, based on a review of LOCA analyses for all operating nuclear power plants, along with NRC staff experience on performing confirmatory analyses, that the cited PWROG statement is correct.

The NRC staff expects the PWROG to include a discussion of this subject in the “A” version of the TR.

3.5 Chemical Effects (TR WCAP-16793, Revision 0, Sections 2.4 and 2.5)

<u>Background</u>

Information regarding how post-LOCA chemicals and containment debris combine to form potential impediments to cooling water flow and heat transfer is found, principally, in TR WCAP-16530-NP (Reference 9), “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191,” and TR WCAP-16793-NP, Revision 0. References to substantiating evidence are provided in each of these documents as well as the responses to RAIs for each of these TRs.

In order to best explain how potential chemical effects could occur, the LOCA sequence of events is presented from a chemical perspective. Following the reactor coolant system blow-down during the initial phase of the LOCA, borated water from storage tanks is pumped into the reactor core through the cold legs (for most plants) and through the UPI points (for a few plants). Simultaneously, some of this water is sprayed into the containment building, thereby cooling the released hot liquid/steam to maintain overall building pressure to less than the plant’s Technical Specification limit to ensure low leakage to the environment. The initial water source used for injection into the reactor vessel and for containment spray is of very high purity (stored in tanks after purification) and with appropriate boron concentration (approximately 2,500 ppm B) for emergency core cooling.

Depending on plant design, there are several ways that chemicals are added to adjust the pH following a LOCA. Plants using sodium hydroxide (NaOH) to adjust pH inject it into the containment spray during the initial spray period. Thus, the containment spray is initially at the high end of the design pH range, and this pH decreases once all NaOH has been added to the spray system and the RCS volume has diluted the NaOH as well. The pool that forms on the containment floor will initially be acidic, due to the borated water from the RCS and storage tanks that has spilled out the break. The pH of that pool and the water collected in the sump will increase to greater than 7.0 over time as the higher pH spray collects at the containment floor and adds to the pool volume.

Plants using trisodium phosphate (TSP) to adjust pH store the granular TSP powder in wire baskets on the floor of the containment building. The TSP dissolves (over approximately a two-hour period) as the water level on the floor of the containment building rises with time. In this case, initial RCS blowdown and containment spray are acidic until the switch from injection phase to the recirculation phase and the dissolution of TSP has adjusted the pH above 7.0. A few plants store sodium tetraborate (STB) in baskets on the containment floor for post-LOCA pH control, and their overall system pH trend with time would be similar to that for a plant with TSP. Plants designed with ice condensers have STB frozen into the ice to adjust the post-LOCA containment building pool pH. Following a LOCA, STB would be added to the pool as the ice melts, thereby increasing the pool pH to greater than 7.0.

Figure 3 shows a schematic of the initial phases of the LOCA with chemicals and flow indicated. This schematic is representative of some plant designs. Over time, the liquid on the containment floor builds up dissolved and suspended chemical concentrations as a result of the reaction between liquid spilling out the break (initially acidic and gradually trending higher to a pH greater than 7.0), plant debris, and wash-down from the spray. Reactions of the containment spray with containment materials take place at a range of pH values, depending on the plant design, the chemical used to adjust pH, and whether this chemical is added directly into the spray system or dissolves in the pool.

During the early injection phase, the spilled water at elevated temperature collects on the containment building floor and interacts with latent debris plus debris generated by the LOCA, such as insulation materials, ablated concrete, coatings, etc. Water that is sprayed reacts with all surfaces of containment materials but at the ambient temperature of containment. Eventually this water joins that spilled onto the containment floor and collects in the sump.

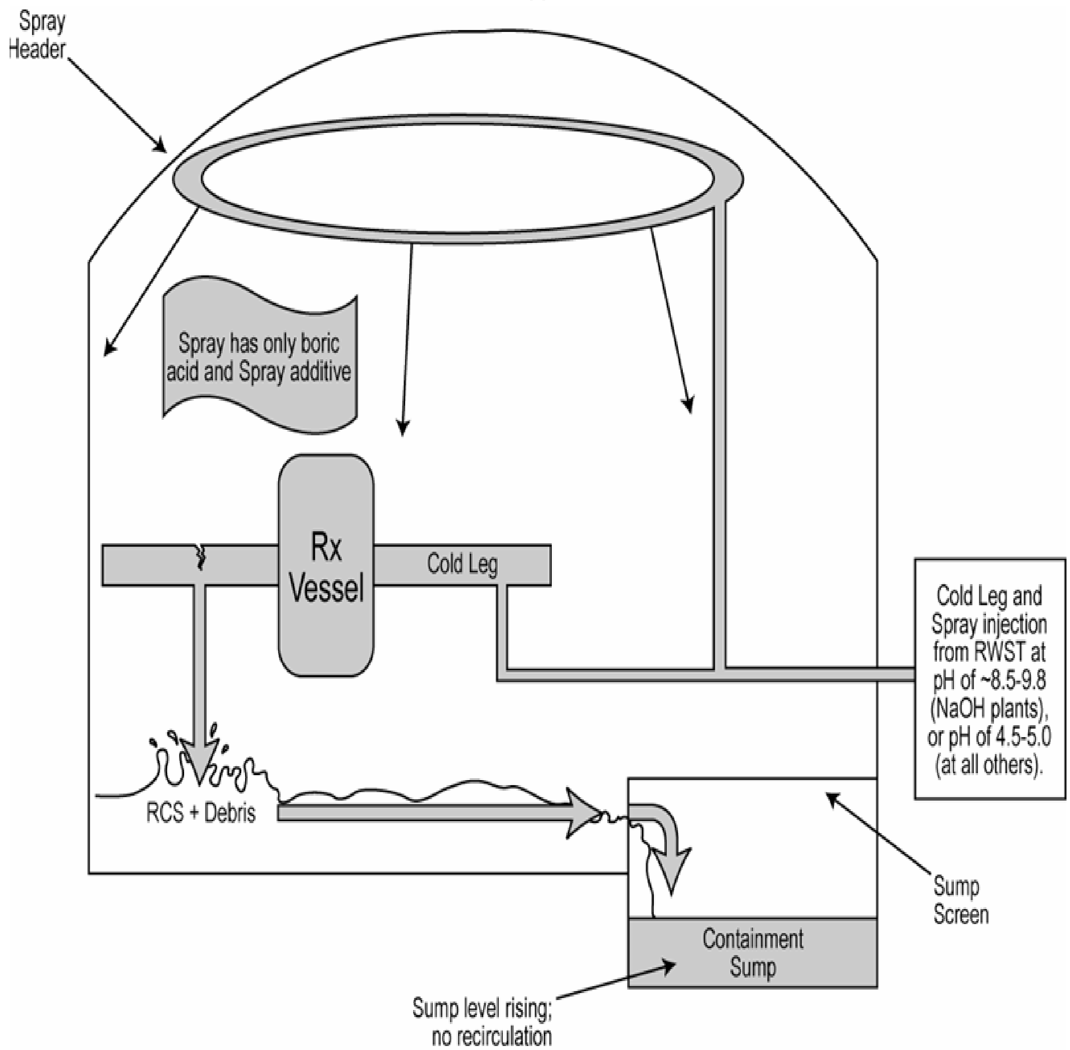

Figure 3
Initial Injection into RCS Immediately Following Break
(Schematic is representative of NaOH injection plants).

Up until this point, nothing is changed from original post-LOCA assumptions. However, potential chemical reactions between the post-LOCA containment building environment and plant materials to produce chemical products are now considered as chemical effects in GSI-191. The Integrated Chemical Effects Tests (ICET) (Reference 10) as well as testing reported in TR WCAP-16530-NP, showed that certain combinations of materials and representative post-LOCA containment environments could produce hydrated, amorphous precipitates that would need to be considered in GSI-191 chemical effect evaluations. Figure 4 shows the change in flow path for the core cooling water as well as the source of the contaminants to the core following the switch to recirculation flow.

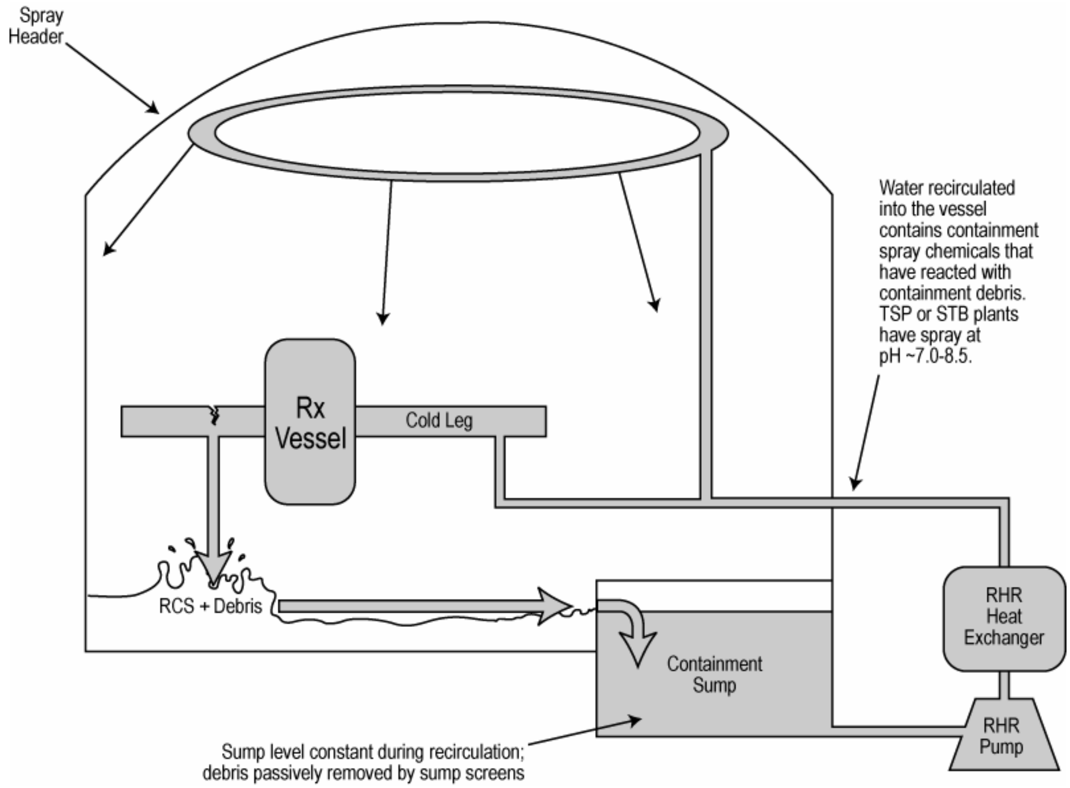

Figure 4
Injection into Cold Leg Using Previously Injected Water Captured in the Containment Sump.
(Note: features such as the presence of a sump pit and screen design are plant-specific)

Although this SE primarily focuses on TR WCAP-16793-NP, there is a link between this TR and TR WCAP-16530-NP since the chemical model contained in TR WCAP-16530-NP is used to develop the potential source term of species that may enter the reactor vessel.

TR WCAP-16530-NP provides a method for evaluating plant-specific chemical effects in a post-LOCA environment, including guidance for how to prepare surrogate chemical precipitates that may be used in strainer head loss tests. The NRC staff has reviewed TR WCAP-16530-NP, and the NRC staff SE is available in ADAMS at Accession No. ML073520891.

TR WCAP-16530-NP, however, does not explicitly address potential chemical effects that may occur in the reactor vessel. TR WCAP-16793-NP, Revision 0, evaluates potential chemical effects that may occur downstream of the sump strainer in the reactor vessel.

The materials tested in TR WCAP-16530, Revision 0, included:

1. commercially pure aluminum and galvanized steel,
2. calcium silicate (Cal-Sil) insulation,
3. NUKON™ fiberglass,
4. other fiberglass - Temp Mat™,
5. Interam™ E-Class insulation,
6. powdered concrete,
7. mineral wool insulation,
8. microporous insulation (e.g., Min-K™), and

9. fire retardant material (e.g., FiberFrax™).

To understand possible chemical effects, TR WCAP-16530-NP describes a number of dissolution tests conducted to examine the chemical behavior of various materials found in the sump environment. Sampling times for the dissolution test were set at 30 minutes, 60 minutes, and 90 minutes. The results of the TR WCAP-16530 test program are consistent with previous work such as the ICET program and show that the predominant materials that are leached from containment materials are:

1. aluminum ions
2. silicates
3. calcium ions

and the predominant chemical precipitates formed are

1. aluminum (oxy) hydroxide,
2. sodium aluminum silicate
3. calcium phosphate (for plants using trisodium phosphate for pH control).

It is possible that other silicate materials may be generated (e.g., calcium aluminum silicate or zinc silicate), but their contribution based on the referenced studies will be small (contributing less than five percent of the total mass) relative to the predominant precipitates.

Technical Evaluation

The TR WCAP-16530-NP model considers the release rates of aluminum, calcium and silicate, as these provide the greatest masses of materials that can become insoluble. Given a source term of material from the TR WCAP-16530-NP model, the NRC staff reviewed the methodology used to determine that these materials:

1. would not deposit on fuel surfaces to the extent that heat transfer is unacceptably low, and
2. would not block flow through the fuel channels should the scale materials deposited become dislodged by spallation during fuel cooldown.

In evaluating the potential for plate-out of dissolved or suspended chemical compounds on the fuel surface, the TR WCAP-16793-NP methodology assumes that all of the dissolved species and compounds resulting from the TR WCAP-16530-NP assessment are transported through the containment sump screen to the reactor vessel. This material represents the source term in TR WCAP-16793-NP for evaluating plate-out of scale-forming materials on the fuel cladding. The NRC staff finds this source term assumption to be acceptable since the chemical source term is based on TR WCAP-16530-NP testing, and it is conservative for the reactor vessel analysis to assume that no precipitate settles on the containment floor, becomes trapped in a debris bed covering the sump screen, or deposits in downstream components such as heat exchangers.

Although the NRC staff finds the use of the chemical model spreadsheet contained in TR WCAP-16530-NP to be acceptable for determining the chemical source term for LOCADM, a

Limitation and Condition was provided in the SE for TR WCAP-16530-NP (ADAMS Accession No. ML073520891) related to the aluminum release rate. The TR WCAP-16530-NP chemical model aluminum release rate is based, in part, on a fit to ICET data using an averaged 30-day release. Actual corrosion of aluminum coupons during the ICET test appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the TR WCAP-16530-NP model under-predicts aluminum release by about a factor of 2 during the active corrosion part of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. To account for potentially greater amounts of aluminum during the initial days following a LOCA, a licensee's LOCADM input shall apply a 2x increase to the TR WCAP-16530 spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the TR WCAP-16530-NP spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the TR WCAP-16530-NP spreadsheet, but the timing of the release is accelerated. Alternately, licensees may choose to use a different method for determining aluminum release, but licensees shall not use an aluminum release rate equation that under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1.

Use of plant-specific refinements to the TR WCAP-16530-NP base chemical model is addressed in Section 4.0, Number 11 in this SE. If a licensee uses plant-specific refinements to reduce the chemical source term calculated by the TR WCAP-16530-NP base model, the licensee shall provide technical justification demonstrating that the refined chemical source term adequately bounds the postulated plant chemical product generation.

TR WCAP-16793-NP used different heat transfer computer programs [ANSYS Mechanical Software and WCOBRA/TRAC WCAP-12945-P-A] and a commercially available calculational software package (MATHCAD) for estimating the effects of the plate out of dissolved materials on the increase in fuel clad temperature. TR WCAP-16793-NP relied on the LOCADM code for their final assessments since the LOCADM calculations address non-uniform chemical deposition due to variation of core power and boiling.

The starting assumption for the LOCADM model with respect to chemical effects was that all the dissolved and suspended chemicals were passed through the containment sump screen and into the reactor core. This is a conservative assumption because it maximizes the amount of chemicals available to cause deleterious effects.

The LOCADM model also assumes that some of the fibrous material from destroyed insulation is not removed by the sump screen and that this material also passes on to the reactor core area. The mass of containment fibrous matter that passes through the screen is assumed to be 1 ft^3 of fiber for every 1,000 ft^2 of sump screen area. For example, a 5,000 ft^2 screen would allow up to 5 ft^3 of fiber bypass (Reference 3). Based on screen bypass test results as discussed in Reference 11, the NRC staff finds the assumption for the amount of fiber bypass to be a reasonable estimate. LOCADM assumes instantaneous chemical participation of this fiber. Therefore, the fiber bypass quantity is converted to a mass of fiberglass and then to an equivalent mass of elements [calcium + aluminum + silicon] that is immediately available to be deposited in the LOCADM analysis. This increase in the mass of dissolved chemicals is compared to the original mass of dissolved chemicals determined by the TR WCAP-16530-NP calculations [calcium + aluminum + silicon] and a percent increase is calculated. This increase is on the order of one to two percent, and is referred to as a "Bump-Up Factor."

These two chemical sources are then used in the plant-specific application of LOCADM. Given the potential plant-specific chemical source term in the reactor vessel, LOCADM determines the amount of scale that deposits on the fuel over time and then calculates maximum fuel clad temperature. An assumption that is very important to the LOCADM calculations is the coefficient of thermal conductivity for the chemical deposits. In order to determine an appropriate thermal conductivity coefficient for the LOCADM calculations, two different thermodynamic equilibrium based codes were used to assess the chemical species that may form in the post-LOCA reactor vessel environment. Westinghouse performed an analysis using the HSC program by Outokomptu, and AREVA performed similar analysis with the OLI StreamAnalyzer Version 2.0.43 program by OLI Systems, Inc. Two thermodynamic equilibrium codes were used to evaluate potential differences in the predicted species and to support the choice of a limiting thermal conductivity value for a chemical deposit that may form on the fuel. Using the chemical species predicted by these thermodynamic equilibrium analyses, a lower-bound thermal conductivity value was selected for LOCADM analysis in TR WCAP-16793-NP to minimize heat transfer and maximize the temperature rise on the fuel surfaces. A chemical deposit thermal conductivity value of 0.11 BTU/(h-ft-ºF) was selected based on the possible formation of a postulated sodium aluminum silicate scale. A thermal conductivity value of 0.11 BTU/(h-ft-ºF) is the minimum thermal conductivity value reported for sodium aluminum silicate scale. For comparison, the thermal conductivity of dry fiberglass insulation is approximately 0.05 BTU/(h-ft-ºF), and, with eight percent of its mass wetted, it increases to approximately 0.1 BTU/(h-ft-ºF). The NRC staff questioned if there were any materials from the thermodynamic predictions for fuel clad surface deposits which could have lower thermal conductivity values. The PWROG responded that 0.11 BTU/(h-ft-ºF) was a bounding thermal conductivity value reported for any of the postulated species that could form a scale deposit on the fuel clad surface. Information provided by the PWROG [RAI #34 from Reference 3] showed thermal conductivity coefficients of representative calcium-based boiler scale deposits that were in the 0.3 to 0.5 BTU/(h-ft-ºF) range, and the thermal conductivity of glass was reported as 0.59 BTU/(h-ft-ºF).

Since the LOCADM calculations do not consider the presence of large debris, the NRC staff questioned whether small pieces of insulation ("fines") incorporated into a deposit could result in a lower thermal conductivity value than the 0.11 BTU/(h-ft-ºF) assumed for a sodium aluminum silicate scale. The PWROG responded that since core temperatures have decreased by the time the ECCS switches from injection to recirculation mode, which is the time when the first fibrous debris could bypass the sump screens and enter the core, the temperature of the core is insufficient to cause melting of the fiberglass or other fibrous material. Therefore, the presence of fiber fines would not create a different type of scale other than that predicted by the thermodynamic models. The PWROG also responded that although dry fiberglass has a lower thermal conductivity than the 0.11 BTU/(h-ft-ºF) assumed for the chemical deposit, a fiber deposit would be porous and would allow water to fill in the porosity. Since water has a much higher thermal conductivity than air, the overall thermal conductivity for a deposit containing fiberglass would be bounded by the assumed 0.11 BTU/(h-ft-ºF) value. This reasoning is supported by literature (Reference 12) that indicated the fiberglass thermal conductivity constant increases by a factor of two with an eight percent volume of water incorporated into its structure. This is also consistent with insulation manufacturer recommendations to change insulation if it is wetted since the heat conduction through the insulation increases; in other words, it is no longer an effective insulator.

Based on the above discussion, the NRC staff finds that the 0.11 BTU/(h-ft-ºF) thermal conductivity value assumed for deposition of scale and particulate represents an acceptably low value to help achieve a conservative prediction of fuel clad temperature increases due to chemical deposits. Since the assumed deposit thermal conductivity has a significant effect on the heat transfer analysis, the use of a 0.11 BTU/(h-ft-ºF) value for sodium aluminum silicate scale is discussed further in Section 4.0, Number 12, in this SE. If plant-specific calculations use a less conservative thermal conductivity value for scale, i.e., greater than 0.11 BTU/(h-ft-ºF), the NRC staff requires the licensee to provide a technical justification for the plant-specific thermal conductivity to the NRC staff. This justification should demonstrate why it is not possible to form a sodium aluminum silicate scale or other scales with conductivities below the selected plant value.

Given the potential chemical source term and using a conservative value for thermal conductivity, LOCADM calculates deposit growth over time. The default initial oxide and crud thicknesses assumed by LOCADM are based on the fuel age and the limiting values that have been measured at modern PWRs. Since the boiling deposition mechanism results in the most rapid deposit growth and forms the most tenacious deposits, LOCADM assumes that all deposition occurs through the boiling process if conditions at a core node predict any boiling. The amount of scale calculated to be deposited under boiling assumes that 50 percent of the water present at the clad surface boils and all solutes transported into the deposit by boiling are deposited locally, as liquid evaporates, at a rate proportional to the steaming rate. Subsequent plate-out of solids, once boiling subsides, is estimated from other literature sources to be $1/80^{th}$ of the solids deposition rate during boiling based on the temperatures encountered at the fuel surfaces. Once formed, deposits are not assumed to thin by flow attrition or by dissolution. The sample LOCADM calculation in TR WCAP-16793-NP, Revision 0, included a 3188 megawatt-thermal PWR with high fiber (7000 ft^3) and a large quantity of calcium silicate insulation (80 ft^3). The NRC staff questioned what additional effect the existing clad crud film and oxide scale (from three cycles) would have on the LOCADM calculations. The PWROG responded (Reference 3) that the sample LOCADM calculation, for the conditions stated above, including initial fuel clad oxide and crud, showed the maximum chemical scale thickness calculated over 30 days was 0.010 inches (10 mils). The maximum clad surface temperature after the start of recirculation was 324 ºF, which meets the acceptance criteria of 800 ºF discussed in Section 3.6 of this SE.

Since LOCADM does not directly account for fiber fines bypassing the sump screen, the NRC staff also questioned how possible effects from fibers depositing in the core are assessed. The PWROG stated that based on test observations, they expected most of the fibrous debris to settle in the lower plenum or to be captured prior to entering the core. Analysis of core inlet blockage is discussed elsewhere in this SE, but modeling demonstrated that with 99 percent of the core flow blocked, sufficient cooling water would be provided as a result of boiling and back flow from above to prevent clad temperatures exceeding 800ºF. To model potential local hot spots, heat transfer analysis was provided in Appendix D of TR WCAP-16793-NP, Revision 0, assuming heat transfer in the radial direction only (i.e., ignoring any axial heat transfer) and using a chemical scale thermal conductivity of 0.1 BTU/(h-ft-ºF). These calculations showed that for a chemical scale thickness of 0.050 inches (50 mils) that formed “instantaneously” at the start of recirculation, the maximum fuel clad surface temperature for a fuel rod diameter of 0.36 inches is 560 ºF. This prediction compared favorably with the acceptance basis value in the TR of a fuel clad surface temperature of 800 ºF. The NRC staff finds this analysis to be acceptable since the assumptions of instantaneous chemical precipitate formation, heat transfer only in the horizontal plane (radial direction), and the assumed thermal conductivity for chemical scale are judged to be conservative.

The NRC staff also questioned whether blockage of core flow channels might occur from scale initially deposited on the fuel surface that would flake off during the cooldown process. The PWROG responded (Reference 3) that the thickness of the scale formed is limited by the amount of solids dissolved in the water. Using scale deposition models they demonstrated that the thickest scale fragment would be insufficient to bridge a fuel rod to fuel rod span to block flow. The NRC staff finds this justification acceptable because the spallation process from the fuel is slow, and experience from spent fuel pool debris generated at PWRs shows these scale materials to be granular and of small size rather than large flakes.

The NRC staff also considered whether sufficient cooling water flow to the core may be compromised by other chemical precipitation effects outside the fuel assemblies. Specifically the NRC staff asked about potential blockage of the residual heat removal (RHR) heat exchanger tubes due to precipitate formation from lowering temperature. The PWROG presented industry work that had been performed to identify how deposits build up and block flow orifices (Reference 3). In the case of the RHR heat exchangers, the linear velocity of the water (approximately 2.5 -5.0 ft/sec) and the RHR pump discharge pressure (approximately 300 psi) are relatively high. The chemical flocculent that might form as fluid temperature is reduced would have very little shear strength, and flocculent formation is typically time dependent. Thus, the water velocity and pump output pressure would be sufficient to prevent blockage from deposition of these materials. The flow rate changes considerably once the water is in the lower core area due to a much greater surface area as well as flow lost to the break. The flow prior to encountering the baffle is downward. Therefore, precipitates, if formed due to temperature decreases in the RHR heat exchanger, will be more likely to be transported to the reactor vessel than to cause blockages in the RHR heat exchanger or piping. Therefore, the NRC staff agrees with the PWROG evaluation that chemical precipitates will not block the RHR heat exchangers.

The NRC staff finds that:

1. The mass of material used to determine the debris and scale loading is conservative based on the source term calculated from the TR WCAP-16530-NP tests, along with the assumption that no precipitates settle on the containment floor, are filtered at the sump screen, or deposit in heat exchangers, piping or in the reactor vessel outside of the core. The mass of materials includes a “bump up factor” to account for leaching of aluminum, calcium and silicon from pieces of insulation materials that bypass the sump screens. The NRC staff finds this bump up factor to be acceptable for reasons stated in this SE section.

2. The thermal conductivity assumed for chemical scale and debris deposits represents an acceptably low value (0.11 BTU/(h-ft-ºF) to help achieve a conservative prediction of fuel clad temperature increase. Wetted insulation allows for better conduction of heat and

the thermal conductivity of wetted insulation would be higher. Thus the use of 0.11 BTU/(h-ft-ºF) is a conservative assumption.

3. Blockage of the RHR heat exchanger based on chemical deposition is unlikely due to time-dependent formation of precipitates and system flows and pressures being able to overcome the low shear strength of precipitate deposits.

4. Industry-recognized calculation models were used to predict temperature increases at the fuel surface as a result of chemical plate-out, and these models confirm that the limit of 800 ºF is not exceeded when these models are used in conjunction with the assumptions in TR WCAP-16530.

5. Blockage of fuel rod spans by spalled fuel scales is unlikely due to the time dependency for spallation and the low thickness of the scale compared to the space between the fuel rods.

Overall, this is a valid approach to determining potential flow restriction due to chemical effects of RCS liquid and containment debris and materials, and is both conservative and representative of the post-LOCA conditions based on chemical reactions described in TR WCAP-16530-NP. Therefore, the NRC staff concludes the chemical effects on core cooling resulting from debris and scale deposition following a LOCA are insufficient to create a condition whereby fuel clad temperatures exceed the temperature limit of 800ºF.

3.6 Local Heating of the Fuel Rods (TR WCAP TR WCAP-16793, Revision 0, Sections 2.2, 2.3, 4.1, and 4.2, and Appendices C and D)

During the long-term cooling period following a LOCA, local hot spots might form within a reactor core as a result of boiling. For some reactor safety system designs, boiling might occur for an extended period as a result of relatively low ECCS flows or relatively high ECCS water temperatures. Local hot spots might result from the plate out of material directly on the fuel rods or as a result of material trapped between the spacer grids and the fuel rods.

The mock fuel assembly test performed at CDI (Reference 6) contained two simulated spacer grids. As fibers were added to the lower section of the mock fuel assembly, most fibers were observed to be captured at the lower fuel nozzle location. Some fibers were also captured at the upper grid strap at the sharp edges and interfaces with the mock fuel rod. Some clumping was observed for the fibers that were captured at the upper grid strap, but it appeared that after a time the clumping ceased. These fiber clumps were observed never to extend the complete length of the grid strap even when more fiber was added. When particulates were added, the clumps of fibers at the grid straps were observed to be as effective in capturing particulates as was the fiber bed at the inlet nozzle of the fuel element mock up.

The test description in Reference 6 did not provide details as to the extent of the blockage of the grid straps by the clumps of fiber and particulate which were observed to form. The NRC staff notes that the fuel element mock-up was not heated so that boiling did not occur. Following a large break LOCA at a PWR, the core is postulated to boil for an extended period of time, concentrating the fiber, particulates and dissolved chemicals in the adjacent coolant. The mixture of material within a post-LOCA reactor core might be much more concentrated than that of the test. In addition, the boiling action might act to deposit a mixture of material behind the fuel grids which would be of a less permeable nature than the clumps of fiber and particulate which were observed in the test. The NRC staff asked the PWROG to provide additional information concerning the effect of fuel rod temperature from debris trapped behind core grid straps during the post-LOCA long term cooling period.

The PWROG described the scenario as follows (Reference 3):

1. Ingested debris will not begin to collect until the ECCS enters recirculation phase. Since the core boil-off rate decreases with time, the period of the greatest rate of boric acid accumulation will occur prior to recirculation.

2. For a hot leg break, the scenario for greatest debris accumulation is not a concern. Conversely, for a cold leg break, which is the boric acid precipitation scenario of concern, core flow is stagnant and therefore debris accumulation is minimal.

3. All plants have boric acid precipitation control measures that promote core dilution after a LOCA. These measures typically rely on an operator initiated action, at some specified time, to start core dilution prior to reaching the boric acid solubility limit. These measures will serve to dilute concentrated sump chemical and suspended debris that might accumulate in the core region. Core dilution flow will flush concentrated chemicals and suspended debris out of the vessel and out the break.

And, further, to assess a maximum clad temperature under worst case debris deposition in a single spacer grid/fuel rod configuration, the following assumptions are made:

1. A uniform debris layer thickness of 110 mils is assumed on the cladding, and,

2. The debris layer is assigned the conservative effective thermal conductivity for a fibrous debris bed recommended in the response to RAI #15 (Reference 3) for debris thermal conductivity = 0.1 BTU/(h-ft-ºF).

Under these limiting assumptions, extrapolating the calculated clad temperatures listed in Table 4-3 (Reference 1) listed for the effective thermal conductivity k_{EFF} = 0.1 BTU/(h-ft-ºF), it is estimated that the maximum clad temperature behind a grid would be less than 738 ºF. This is an extremely conservative estimate of the clad temperature for the following reasons:

1. A conservatively small value of conduction through the debris bed identified in the response to RAI #15 (Reference 3) is used,

2. The calculation does not account for circumferential heat transfer about the debris bed which would form in the spacer grid between the dimples and springs and the corners of the spacer grid, and,

3. Convection of heat by the flow of coolant through the debris bed is neglected. (The ability of coolant to pass through a fibrous and particulate debris bed under PWR flow conditions was demonstrated in the response to RAI #2 (Reference 3).)

Therefore, Reference 3 concluded that the estimation of a maximum clad temperature of 738 ºF is a very conservative estimate of the maximum clad temperature if the location between a spacer grid and a fuel rod were to become completely filled with debris.

In addition to conduction through any plated out or accumulated material, heat must also traverse the oxide layer built up on the surface of the cladding during normal operation and during the earlier phases of the LOCA event. The oxide will have a thermal conductivity lower than the original cladding material. The PWROG described the approach to calculating the oxide layer thickness and heat transfer in Reference 3. The PWROG proposed that the assumed cladding oxide thickness for input to LOCADM be the peak local oxidation allowed by 10 CFR 50.46, 17 percent of the cladding wall thickness. A lower oxidation thickness can be used on a plant-specific basis if that value is justified. The NRC staff does not agree with the degree of flexibility in this approach and will impose a condition of 17 percent oxidation on the LOCADM analysis.

Reference 3 then used as an example an analysis based on a cladding thickness of 0.0225 inches with a reduced metal thickness of 0.0187 inches and an oxide layer of 0.006 inches. The calculation presented in the TR was based on an oxide thickness of 0.004 inches with an oxide thermal conductivity of 0.1 BTU/(h-ft- ºF). Increasing the clad oxide thickness to 0.006 inches results in a temperature increase of 2 ºF over that of the 0.004 inch oxide layer. Based on review of the evaluation provided in Reference 3, the NRC staff agrees that this is a conservative approach to the effect of the oxide layer on cladding heat-up. Therefore, the NRC staff finds its use acceptable for this application.

The PWROG provides three methods for calculating heat-up at local hot spots. The first method for evaluating hot spots is described in Section 4.1 and Appendix C of TR WCAP-16793-NP, Revision 0. This method uses the ANSYS Mechanical software to calculate cladding temperatures. The ANSYS Mechanical software provides solutions for linear and nonlinear and dynamic analysis for a variety of engineering problems. For the cladding heat up calculations, only the thermal solution capabilities were used. The software was used in a manner consistent with standard industry practices for ANSYS mechanical software. Therefore, the NRC staff finds its use acceptable for this application.

Using the ANSYS Mechanical software, TR WCAP-16793-NP, Revision 0, provided sample calculations for debris thicknesses of up to 50 mils which might form between a spacer grid and an enclosed fuel rod. The NRC staff understands that the distance between a fuel rod and the enclosing spacer grid may be considerably larger than 50 mils. TR WCAP-16793-NP, Revision 0, stated that the minimum clearance between two adjacent fuel rods, including an allowance for the spacer grid thickness, is greater than 100 mils. Therefore, a 50-mil debris thickness on a single fuel rod is the maximum deposition to preclude touching of the deposition of two adjacent fuel rods with the same thickness of deposition. The 50 mil thickness is the maximum acceptable deposition thickness before bridging of adjacent fuel rods by debris is predicted to occur. In the worst case scenario (minimum gap and maximum deposition), local bridging might occur but would not be expected to lend to significant localized fuel rod heating.

For current fuel designs, the minimum clearance between the cladding and the spacer grid is about 40 mils; this occurs where the springs and dimples of the grid contact the fuel rod. The maximum clearance between the cladding and the spacer grid occurs along the diagonal of a grid cell and is about 110 mils. Thus, if a spacer grid were to become completely filled, the radial thickness of the debris on the outside clad would vary from about 40 mils to about 110 mils about the circumference of a fuel rod.

In Section 2.3 of Reference 1, the PWROG used the ANSYS code to evaluate the resulting fuel cladding temperatures for fuel rods covered with debris having thermal conductivities as low as 0.1 BTU/(h-ft-ºF). The NRC staff questioned whether insulating fiber beds can have thermal conductivities considerably smaller than this value. The PWROG contacted Performance Contracting, Inc. (PCI). PCI is the owner of the NUKON® product line, a low density insulation material commonly used in PWR containment buildings. PCI was requested to provide information regarding the effect of wetting on the thermal conductivity of the NUKON® insulation. PCI noted that the thermal conductivity of fibrous insulation is a function of the moisture content. Under normal industrial and nuclear applications, NUKON® is used on hot piping and components; thus the moisture content is low. PCI stated they had no data for the thermal performance of wetted insulation because, for industrial applications, when insulation becomes wetted it ceases to perform its function because the thermal conductivity of the wetted insulation will increase towards the value for water. Thermal insulation would be replaced if wetted since it is no longer insulating. The NRC staff reviewed the performance of the NUKON product line and prepared an SER as contained in Reference 13. The NRC staff accepts this rationale as justification for the assumption that a thermal conductivity for the insulating material equivalent to its dry conductivity is conservative.

The PWROG also provided an analysis of the thermal conductivity of wet versus dry fiber and showed that the insulating property of the dry material is five times, or more, effective as compared to wet material. The NRC staff has reviewed the submittal and agrees with this assessment. The NRC staff believes this conclusion to be valid for all fiber types.

The second method for evaluating hot spots is described in Section 4.2 and Appendix D of TR WCAP-16793-NP. This method uses steady state cylindrical heat conduction methodology to calculate fuel rod cladding temperature as a result of crud deposits on the surface of a fuel rod at locations which are not within a spacer grid. Sample calculations are provided for crud thicknesses up to 50 mils and for crud thermal conductivities as low as 0.1 BTU/(h-ft-ºF). Appendix E to the TR presents the basis for calculation of thickness, composition, and heat transfer characteristics of crud. The deposition process described is based on work such as that detailed in Reference 14.

Calculation of the heat transfer through the crud deposit is done using LOCADM with either a plant-specific estimate based on fuel examinations that have been performed for the plant fuel, or using limiting values from industry fuel examinations. The default version of the model uses a crud thickness of 100 microns for second and third cycle fuel, and 50 microns for first cycle fuel. The thickest crud measured in PWR fuel is 127 microns.

The limiting crud thermal conductivity quoted for PWR fuel, 0.3 BTU/(h-ft-ºF), falls at the lower end of the measured calcium-rich crud on boiler tubes, 0.29-0.55 BTU/(h-ft-ºF). Sodium aluminum silicate based crud deposits have thermal conductivities as low as 0.11 BTU/(h-ft-ºF). Thus, the PWROG concludes that using a limiting thermal conductivity of 0.1 BTU/(h-ft -ºF) is conservative. Based on review of the information provided, the staff accepts the PWROG conclusion that a thermal conductivity of 0.1 BTU/(h-ft-ºF) is conservative.

The PWROG does not believe that fibrous debris will adhere to the surface of the fuel rods. This was also an observation from the unheated fuel bundle mockup test as described in Reference 6. Although fibrous clumps were observed to form at the grid scraps, it was observed that fibers did not get captured at the smooth surfaces of the fuel rods. The NRC staff notes that the “fuel rods” in the demonstration observed in the reference were made of 3/8-inch plastic. Thus, the surface condition would not be representative of Zircalloy in texture, friction factor, or potential oxide buildup. Conclusions regarding adherence of fibrous material to fuel cladding can not be drawn from the demonstration referenced.

Testing of fibrous debris on heated surfaces is described in Reference 13. In these tests, an immersion heater, which was heated electrically or with a torch, was quenched in a slurry of glass wool and associated insulation blanket material which included the adhesive binder. Initial heater temperatures ranged from saturated to 2200 ºF. In other tests the electric heater was immersed within the slurry for hours. Fiber-to-metal or fiber-to-fiber adherence was not observed. The small amount of fiber which was present on the heater surface after the test was easily brushed away with a soft brush. Some sticking of the binder material was observed with sometimes a layer of fibers at the most one or two fiber diameters thick. The NRC staff notes that the mixture of material within the reactor core may be different from that of the test. Fibrous debris in the reactor sump water may be of a different nature than the glass fibers in the immersion heater tests.

The tests conducted and reported in Reference 13, referred to in Reference 15, were conducted in three ways: a rod heated to 2200 ºF was quickly quenched in a slurry of chopped blanket fibers in distilled water, a stainless steel strip was placed in the slurry and heated to nucleate boiling and held for three hours, and a stainless steel strip was placed in the slurry and heated to film boiling and held for three hours. The slurry was made from commercially available (1978 era) high temperature Fiberglas insulation with a density of 1 to 2.5 pounds per cubic feet, and a service temperature of up to 1000 ºF. The tests provided qualitative indications that the fibrous material would not adhere significantly to the heater surface. The NRC staff concluded in its SE on Reference 13 that the fiber deposits which occur on rods heated to film or nucleate boiling temperatures will not accumulate in sufficient thickness or quantity to measurably change either the flow of coolant or the heat transfer characteristics of the heated surface.

The third method for evaluating hot spots is described in Section 5.5 and Appendix E of the TR. This method uses steady state slab heat conduction methodology in the LOCADM computer code to calculate fuel rod cladding temperature as a result of crud deposits on the surface of a fuel at locations which are not within a spacer grid. The LOCADM computer code also calculates the amount of suspended and dissolved material in the core, and the plate out of that material on the fuel rods as discussed in Section 3.1 of this SE.

The NRC staff raised questions concerning the calculation of plate out of dissolved and suspended materials including fibrous debris upon the surface of fuel rods undergoing long-term boiling. The PWROG responded in Reference 3 that the deposition of small fibers that do not dissolve but are small enough to be transported through the sump screen and into the core cannot be ruled out. They also stated that the quantity of transported fines is expected to be small compared to both the total amount of debris and the amount of debris that dissolves or corrodes. The PWROG stated that quantity 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 input.

The PWROG presented a comparison of the LOCADM slab conduction methodology with the cylindrical conduction methodology of TR WCAP-16793, Appendix D, showing that similar results are obtained for thin crud layers. The NRC staff questioned whether for thick crud layers, the inaccuracies of the slab model for modeling cylindrical heat transfer might make this methodology unusable. In Reference 3, Table 34-1, the PWROG provided results of a comparison of using a slab geometry versus a cylindrical geometry heat transfer model. The slab geometry resulted in a temperature drop across the deposit that was 61 ºF greater than for the cylindrical geometry, thus, indicating that LOCADM is conservative. The NRC staff accepts this conclusion.

The PWROG has selected an average cladding temperature of 800 ºF as the acceptance basis for long-term cooling. The PWROG stated in Reference 3, that autoclave test data has demonstrated that oxidation and hydrogen pickup will be well behaved at and below the 800 ºF temperature and the reduction in cladding mechanical performance is small. The cladding specimens in the autoclave tests were for fresh material. The NRC staff requested data for specimens which have undergone prior exposure to LOCA heat-up and quench conditions. The PWROG response (Reference 3) was to refer to the autoclave test data and a literature review that indicates that susceptibility to localized accelerated corrosion occurs at temperatures in excess of 800 ºF. They do not expect cladding properties to degrade due to a 30-day exposure to a temperature of 800 ºF. While the NRC staff accepts that the autoclave results are sufficient to justify this temperature, the PWROG has not provided results from exposure of oxidized or pre-hydrided cladding material representative of temperature exposure in excess of 800 ºF. Lacking these test results, the NRC staff accepts a temperature limit of 800 ºF as the long-term cooling limit. Should a licensee calculate a temperature that exceeds this value, cladding strength data must be provided for oxidized or pre-hydrided cladding material that exceeds this temperature limit.

In Appendix A, the PWROG further stated that the maximum allowable fuel clad temperature for short transients, such as "hot leg switch over," and for localized "hot spots" is 2200 ºF during long-term cooling. The PWROG has clarified (Reference 3) that the reference to a temperature excursion as high as 2200 ºF during hot-leg switch-over is incorrect, and have stated that there are no known phenomena that would cause localized hot spots during long-term cooling. The PWROG will amend the text in the TR accordingly. The NRC staff accepts the PWROG response.

3.7 Application of TR WCAP-16793-NP, Revision 0

The PWROG concluded in TR WCAP-16793, Revision 0, that there is reasonable assurance acceptable long-term core cooling with debris and chemical products in the recirculating fluid has been demonstrated for all plants. This is based on the evaluations and sample analyses in the TR. The topics evaluated were:

1. Fuel clad temperature response to blockage at the inlet to the core.

2. Fuel clad temperature response to local blockages or chemical precipitation on fuel clad surfaces.

3. Chemical effects in the core region, including potential for plate-out on fuel cladding.

Following the procedures presented in the TR, using the standard methods discussed, a licensee is expected to be able to demonstrate that its plant will have adequate long-term core cooling in the presence of post-LOCA debris. A typical example of the amount of coolant flow needed to match the long-term boil-off for a Westinghouse 4-loop PWR is shown in Figure 5, taken from Reference 16.

The NRC staff believes that, for a utility to apply the conclusions of TR WCAP-16793, Revision 0, to a particular plant, certain conditions must be shown to be met regarding the type, amount and concentration of the chemicals and debris which might enter the reactor core during the long-term cooling period. The NRC staff requested that the PWROG provide such a list of acceptance criteria for each evaluation area of the TR which utilities could evaluate against the predicted condition of their plant following a large break LOCA as it entered into the long-term cool cooling period.

The PWROG response in Reference 3 is that a guidance document will be provided to the licensees on implementation of the TR. The draft guidance document, Reference 16, does not contain information such as acceptance criteria for evaluation areas. The draft document is a high-level discussion of the goals of long-term cooling with no details regarding how an analysis is to be performed, what information is needed, or how to determine acceptability of the results.

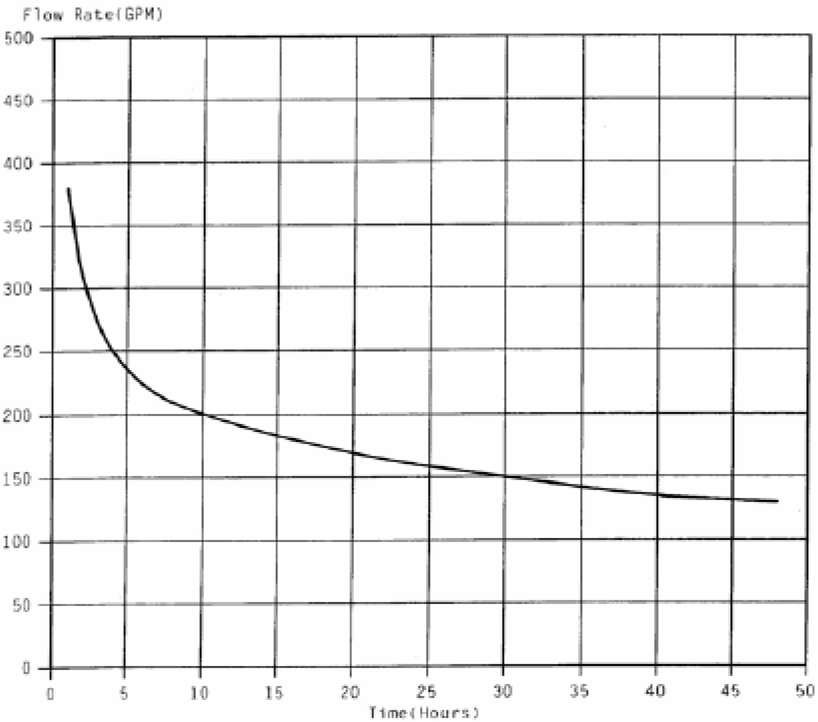

Figure 5
Typical Westinghouse 4-Loop PWR
Boil-off Curve

The PWROG also states in Reference 3, that licensees will be provided a copy of LOCADM with instructions on what inputs are needed to support the calculations. Licensees will also be provided a sample problem to confirm installation of LOCADM has been carried out successfully. The statement is made that licensees will have to perform a plant-specific analysis using plant-specific input. The analysis will then produce a comparison between the plant-specific cladding deposition and the maximum deposition of 50 mils, and the plant-specific cladding temperature and the 800 ºF acceptance limit.

The PWROG recognized in Reference 16, that some plants might not be bounded by the sample cladding deposit calculations presented in Section 5 of the TR, and indicated that plant-specific chemical deposition calculations may need to be performed using the method described in that section. The methodology in Section 5 includes both that of Westinghouse and AREVA NP, Inc. The methodology is fairly complex. Therefore, the NRC staff requested assurance that personnel performing these calculations would be trained and well qualified for making these evaluations. The PWROG responses in Reference 3 indicate a conviction that the methodology employed in LOCADM is sufficiently simple, and that on-line training is all that is necessary. The extensive training required of users of the large system analysis codes is not believed to be necessary.

The NRC staff is not convinced that low-level training on use of the TR is sufficient. The PWROG should provide training for utility representatives in the application of the TR for individual plants.

3.8 Acceptance Criteria (TR WCAP-16793, Revision 0, Section 3 and Appendix A)

The PWROG states in Section 3 of the TR that long-term core cooling acceptance bases are met when: (1) decay heat removal is provided such that core average cladding temperatures do not exceed 800 ºF, ensuring that rapid nodular corrosion and higher hydrogen pickup rates do not occur, and (2) boric acid concentration in the core region is prevented from exceeding the precipitation limit. The PWROG also states that since the fuel undergoing a large break LOCA will not be reused, the above noted acceptance criteria are conservative and, when used in conjunction with the engineering calculations discussed in the TR, provide reasonable assurance that long-term core cooling will be successfully maintained.

The NRC staff agrees that adherence to the methods and procedures discussed in this SE, including the conditions and limitations discussed in Section 4.0, will provide reasonable assurance of maintaining long-term core cooling following a postulated LOCA. However, the fuel cladding temperature limit of 800 ºF is to be the predicted peak cladding temperature rather than the core average temperature.

4.0 LIMITATIONS AND CONDITIONS

1. TR WCAP-16793-NP, Revision 0, states that licensees shall either demonstrate that previously performed (sump strainer) bypass testing, as cited in Section 2.1 of the TR, is applicable to their plant-specific conditions, or perform their own plant-specific testing. The NRC staff agrees with this stated position. (Section 3.1 of this SE)

2. There are very large margins between the amount of core blockage that could occur based on the fuel designs and the debris source term discussed in the TR and the blockage that would be required to degrade the coolant flow to the point that the decay heat could not be adequately removed. Plant-specific evaluations referencing TR WCAP-16793-NP should verify the applicability of the TR blockage conclusions to the licensees' plant and fuel designs. (Section 3.2 of this SE)

3. Should a licensee choose to take credit for alternate flow paths such as core baffle plate holes, it shall demonstrate that the flow paths would be effective and that the flow holes will not become blocked with debris during a LOCA and that the credited flow path would be effective. (Section 3.1 of this SE)

4. Existing plant analyses showing adequate dilution of boric acid during the long-term cooling period have not considered core inlet blockage. Licensees shall show that possible core blockage from debris will not invalidate the existing post-LOCA boric acid dilution analysis for the plant. (Section 3.2 of this SE)

5. The NRC staff expects the PWROG to revise TR WCAP-16793-NP, Revision 0, in the “A” version of the TR to address the NRC staff RAIs and the PWROG responses. In

addition, a discussion of the potential for fuel rod swelling and burst to lead to core flow blockage shall be included in the "A" version of the TR. (Section 3.4 of this SE)

6. TR WCAP-16793-NP, Revision 0, shall be revised in the "A" version of the TR to indicate that the licensing basis for Westinghouse two-loop PWRs is for the recirculation flow to be provided through the upper plenum injection (UPI) ports with the cold-leg flow secured. (Section 3.3 of this SE)

7. Individual UPI plants will need to analyze boric acid dilution/concentration in the presence of injected debris for a cold-leg break LOCA. (Section 3.3 of this SE)

8. TR WCAP-16793-NP, Revision 0, states that the assumed cladding oxide thickness for input to LOCADM will be the peak local oxidation allowed by 10 CFR 50.46, or 17 percent of the cladding wall thickness. The TR states that a lower oxidation thickness can be used on a plant-specific basis if that value is justified. The NRC staff does not agree with the flexibility in this approach. Licensees shall assume 17 percent oxidation in the LOCADM analysis. (Section 3.6 of this SE)

9. The NRC staff accepts a cladding temperature limit of 800 ºF as the long-term cooling acceptance basis for GSI-191 considerations. Should a licensee calculate a temperature that exceeds this value, cladding strength data must be provided for oxidized or pre-hydrided cladding material that exceeds this acceptance basis temperature. (Section 3.6 of this SE)

10. In the response to NRC staff RAIs, the PWROG indicated that if plant-specific refinements are made to the TR base model to reduce conservatisms, the LOCADM user shall demonstrate that the results still adequately bound chemical product generation. If a licensee uses plant-specific refinements to the TR base model that reduce the chemical source term considered in the downstream analysis, the licensee shall provide a technical justification that demonstrates that the refined chemical source term adequately bounds chemical product generation. This will provide the basis that the reactor vessel deposition calculations are also bounding. (Section 3.5 of this SE)

11. TR WCAP-16793-NP, Revision 0, states that the material with the highest insulating value that could deposit from post-LOCA coolant impurities would be sodium aluminum silicate. The TR recommends that a thermal conductivity of 0.11 BTU/(h-ft-ºF) be used for the sodium aluminum silicate scale and for bounding calculations when there is uncertainty in the type of scale that may form. If plant-specific calculations use a less conservative thermal conductivity value for scale [i.e., greater than 0.11 BTU/(h-ft-ºF)], the licensee shall provide a technical justification for the plant-specific thermal conductivity. This justification shall demonstrate why it is not possible to form sodium aluminum silicate or other scales with conductivities below the selected value. (Section 3.5 of this SE)

12. TR WCAP-16793-NP, Revision 0, indicates that initial oxide thickness and initial crud thickness could either be plant-specific estimates based on fuel examinations that are performed or default values in the LOCADM model. Consistent with Limitation and Condition Number 8, the default value for oxide used for input to LOCADM will be the

peak local oxidation allowed by 10 CFR 50.46, or 17 percent of the cladding wall thickness. The default value for crud thickness used for input to LOCADM is 127 microns, the thickest crud that has been measured at a modern PWR. Licensees using plant-specific values instead of the TR default values for oxide thickness and crud thickness shall justify the plant-specific values. (Section 3.5 of this SE)

13. As described in the Limitations and Conditions for TR WCAP-16530-NP (ADAMS Accession No. ML073520891), the aluminum release rate equation used in TR WCAP-16530-NP, provides a reasonable fit to the total aluminum release for the 30-day ICET tests but under-predicts the aluminum concentrations during the initial active corrosion portion of the test. To provide more appropriate levels of aluminum for the LOCADM analysis in the initial days following a LOCA, licensees shall apply a factor of two to the aluminum release as determined by the WCAP-16530-NP spreadsheet, although the total aluminum considered does not need to exceed the total predicted by the WCAP-16530-NP spreadsheet for 30 days. Alternately, licensees may choose to use a different method for determining the aluminum release, but in all cases licensees shall not use a method that under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1. (Section 3.5 of this SE)

14. In Appendix A of TR WCAP-16793-NP, Revision 0, the PWROG stated that the maximum allowable fuel clad temperature for short transients and for localized "hot-spots" is 2200 ºF during long-term cooling. However, in Reference 3, the PWROG clarified that the reference to a temperature excursion as high as 2200 ºF is incorrect and the PWROG will amend the text in the "A" version of the TR accordingly. (Section 3.6 of this SE)

5.0 CONCLUSION

The NRC staff has reviewed TR WCAP-16793-NP, which describes a methodology for consideration of particulates, fibrous and chemical debris affecting the long-term cooling at operating PWRs following a LOCA. In the course of the review of the TR, the NRC staff requested additional information in a number of areas. The PWROG responses to those RAIs (Reference 3) have been reviewed and found acceptable, subject to the conditions and limitations stated in Section 4.0 of this SE.

The NRC staff concludes, considering the conditions and limitations noted above, that application of the procedures and methods described in TR WCAP-16793-NP will provide an acceptable plant-specific evaluation of the plant's ability to adequately remove long-term decay heat from the core following a postulated LOCA.

6.0 REFERENCES

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2. Martin, T. O., NRC staff letter to Gresham, J. A., Westinghouse Electric Company, "Nuclear Regulatory Response to Westinghouse Letter LTR-NRC-06-46 Dated

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