WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President, Engineering

February 29, 2008
ET 08-0003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference:
1) Letter dated February 9, 2006, from USNRC to R. A. Muench, WCNOC
2) Letter ET 05-0018, dated August 31, 2005, from Terry J. Garrett, WCNOC, to USNRC
3) Letter WO 06-0028, dated May 31, 2006, from Stephen E. Hedges, WCNOC, to USNRC

Subject: Docket No. 50-482: Wolf Creek Nuclear Operating Corporation Response to Request for Additional Information RE: Response to Generic Letter 2004-02: "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors"

Gentlemen:

Pursuant to 10 CFR 50.54(f), this letter provides the Wolf Creek Nuclear Operating Corporation (WCNOC) response to the NRC request for additional information (RAI) regarding WCNOC's response to Generic Letter 2004-02 (Reference 1). Responses to the RAI questions are incorporated into a supplemental response to Generic Letter 2004-02 which is contained in Attachment I. Attachment I supersedes WCNOC's previous Generic Letter 2004-02 responses (References 2 and 3) in their entirety.

Attachment II lists each RAI request in Reference 1 and provides a reference to the applicable portion(s) of Attachment I or other appropriate information to address the request.

Attachment III lists commitments made to the NRC by this letter. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Richard Flannigan at (620) 364-4117.

Sincerely,

Terry J. Garrett

A116
NRR

TJG/rlt

Attachments: I Generic Letter 2004-02 Supplemental Response
II RAI Cross-Reference List
III List of Regulatory Commitments

cc: E. E. Collins (NRC), w/a
V. G. Gaddy (NRC), w/a
B. K. Singal (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By____________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 29th day of Feb., 2008.

Gayle Shepheard
Notary Public

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2011

Expiration Date 7/24/2011

Generic Letter 2004-02 Supplemental Response

Notes on Format and Content

Wolf Creek Nuclear Operating Corporation (WCNOC) used a standardized format in portions of this supplemental response to address each major section of Generic Letter 2004-02. There is a statement in each section as to whether information included in the section revises or supplements information that WCNOC previously provided to the NRC in response to Generic Letter 2004-02 (References 2 and 3), or whether the information previously supplied continues to apply. There is a statement in each section, if applicable, describing the use of NEI 04-07 guidance report (GR) (Reference 1) guidance, which is referred to in this attachment as NEI 04-07, Vol. 1 (GR). There is also a statement, if applicable, describing conformance to, or exceptions to NEI 04-07 NRC safety evaluation (SE) (Reference 1) requirements, which is referred to in this attachment as NEI 04-07, Vol. 2 (SE).

Each major section describes Generic Letter 2004-02 evaluations and other corrective actions that impact conformance to the regulatory requirements listed in Generic Letter 2004-02. Each section also includes the basis for methods and key assumptions not consistent with NRC-approved guidance or not previously reviewed by the NRC staff.

Summary-Level Description of WCNOC Approach

Wolf Creek Nuclear Operating Corporation's (WCNOC's) approach to resolving issues described in Generic Letter 2004-02 is to use the guidance and requirements of NEI 04-07 (Reference 1), as well as industry guidance, industry testing and plant-specific testing, to perform a comprehensive set of evaluations of the effects of design basis accident conditions on the ability of structures, systems and components, including the containment emergency sump strainers, to mitigate the consequences of the analyzed accidents and maintain long term core cooling in a manner consistent with governing regulatory requirements listed in Generic Letter 2004-02.

New sump strainers were installed in the existing emergency recirculation sump pits in the containment building to accommodate the water levels postulated for post-accident conditions. Each new sump strainer contains stacked plates arranged into modules that maximize the strainer surface area. The strainer plates have perforated stainless steel plate surfaces with 0.045 inch diameter holes to efficiently capture debris that enters the sump pits. While the original containment recirculation sump screens and trash racks had approximately 200 ft^2 of effective surface area per sump, the new replacement sump strainers have approximately 3300 ft^2 of effective surface area per sump.

Debris barrier plates have been installed in openings through the secondary shield wall that are near the emergency recirculation sumps. The barriers prevent the "short path" flow of debris-laden fluid directly to the sumps and force the fluid to take a longer "tortuous path" through shield wall openings farther away from the sumps to allow more time for the debris to settle out.

WCNOC has implement changes to programmatic controls for (1) design change process procedures, (2) containment entry and material control procedures, (3) clearance orders procedures, (4) work request procedures, and (5) scaffold construction and use procedures.

WCNOC has implemented changes to surveillance procedures to ensure that the installed replacement strainers will not have openings in excess of the maximum designed strainer opening.

WCNOC has implemented a containment latent debris assessment program, which utilizes swipe sampling to determine the amount of latent debris in the containment building. Housekeeping and foreign materials exclusion procedures have been revised to target containment building cleaning based on the results of the swipe sampling survey.

WCNOC has implemented a containment coatings assessment program for monitoring and assessing the containment building coatings, including administrative controls on conducting coating examinations, including deficiency reporting criteria and documentation requirements.

Interim compensatory measures implemented at Wolf Creek Generating Station (WCGS) in accordance with NRC Bulletin 2003-01, as described in Section 2 below, will remain in place at a minimum until all evaluations and corrective actions for GL 2004-02 are complete:

The approach described in this supplemental response incorporates numerous conservatisms and margins providing high confidence that issues described in Generic Letter 2004-02 have been adequately addressed, even considering that potential uncertainties may exist or could be identified in the future. A listing of significant conservatisms and margins is provided in the table below.

WCNOC Conservatisms and Margins Table

Item	Conservatism	Discussion
A value of 200 pounds of latent debris was used in the determination of debris generation.	Actual documented amount of latent debris is less than 70 pounds.	Margin of greater than 130 pounds of latent debris.
A distribution of 85% dirt/dust and 15% latent fibers is assumed for the latent debris in the containment building.	This mass fraction is recommended as a conservative assumption in Section 3.5.2.3 of NEI 04-07, Vol. 2 (SE).	
All debris that is not blown to upper containment	Conservatively assumed to flow outside the bio-shield wall	Some debris will be held up at various locations. Therefore, there will be less debris actually reaching the sump.
Transport fraction of fine debris generated by break	Conservatively assumed to be 100%	Some fine debris will settle out, prior to reaching the sump. Therefore, there will be less fine debris actually reaching the sump.

WCNOC Conservatisms and Margins Table (Con't)

Item	Conservatism	Discussion
Assumed that all transportable miscellaneous debris identified in the debris generation calc, including tags, labels, etc. would be transported to the sumps during recirculation	Conservative assumption	Some of the miscellaneous debris will either be held up or settle out prior to reaching the sump. Therefore, there will be less debris actually reaching the sump.
Assumed all debris blown upward would be subsequently washed back down by containment spray system flow with the exception of any pieces of fiberglass or metallic insulation debris held up on the grating.	Conservative assumption since some debris is blown up onto holdup areas protected from the containment spray path.	Some debris will be held up. Therefore, there will be less debris actually reaching the sump.
Assumed the debris generated by a small break loss of coolant accident (LOCA) blast would not be blown into upper containment.	Conservative assumption since no credit is taken for holdup of debris in upper containment since containment spray is not actuated in a small break LOCA. In addition, the small break LOCA has a correspondingly small zone of influence (ZOI).	Some debris (e.g., latent debris) will be held up. Therefore, there will be less debris actually reaching the sump.
With the exception of latent debris washed to the sumps and inactive cavities during pool fill-up, it was assumed that all latent debris is in lower Containment and would be uniformly distributed in the containment pool at the beginning of recirculation.	Conservative assumption since no credit is taken for debris remaining on structures and equipment above the pool water level	Some debris will be held up. Some debris will settle out. Therefore, there will be less debris actually reaching the sump.
Assumed that large piece debris that is not blown to upper containment would be distributed between the break location and the sumps at the beginning of recirculation.	Conservative assumption since it neglects the fact that some debris would be blown or washed to areas farther away from the sump during the blow-down and pool fill-up phases.	Some debris will land in stagnant areas and not be transported to the sump. Therefore, there will be less debris actually reaching the sump.

WCNOC Conservatisms and Margins Table (Con't)		
Item	**Conservatism**	**Discussion**
Water falling from the reactor coolant system (RCS) breach was assumed to do so without encountering any structures before reaching the containment pool.	Conservative assumption since any impact with structures would dissipate the momentum of the water and decrease the turbulent energy in the pool.	Some debris will be held up. Some debris will settle out. Therefore, there will be less debris actually reaching the sump.
The temperature of the water in the safety injection accumulators is assumed to be equal to the maximum initial containment air temperature of 120°F.	This approach is conservative because the density of water decreases with increasing temperature.	The actual temperature will normally be considerably lower. Therefore, there would be a larger mass of available water in the accumulators.
For a break in the RCS loop piping, it is assumed that the vessel, RCS loop piping and pressurizer surge line are refilled with emergency core cooling system (ECCS) inventory at the time of ECCS switchover to recirculation.	This is conservative because given the high temperatures within the steam generators and pressurizer and the high temperature in the containment building, it is probable that the ECCS inventory will not be drawn into these components until later in the postulated accident when the containment spray system switchover occurs.	This would increase the mass of water in the sump pool, at initial switchover.
General Observation: All debris that ends up in the recirculation pool takes up volume in the recirculation pool.	All debris in the recirculation pool, whether it has settled to the bottom or is suspended, displaces water.	No credit is taken in the water level calculation for the volume of debris in the recirculation pool. Therefore, the actual post LOCA debris-laden water level would be higher than that calculated, not considering the debris.
Maximum debris would be generated near the "D" loop area, but transport was assumed to originate from the "C" loop area, which is closer to the sumps.	Break in the Loop "D" area could result in lower transport fractions from inside the secondary shield wall than assumed with an origin in the loop "C" area.	Conservative transport values. A break in the "C" loop area would generate less debris.

Response to Specific Review Areas

1. Overall Compliance:

Activities are currently underway to ensure that the Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) recirculation functions under debris loading conditions at Wolf Creek Generating Station (WCGS) will continue to be in full compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of Generic Letter (GL) 2004-02. This will be achieved through assessment of the results of site-specific testing activities for potential impact on design inputs, assumptions and conclusions of WCNOC calculations and evaluations conducted in response to issues identified in GL 2004-02. As described in Reference 4 and approved in Reference 5, GL 2004-02 corrective actions will be completed and implemented by June 30, 2008. Following the implementation of calculations, evaluations and other changes described below, the ECCS and CSS recirculation functions will continue to support the 10 CFR 50.46 requirement for the ECCS to provide long-term cooling of the reactor core following a loss of coolant accident (LOCA), as well as the requirements of 10 CFR 50 Appendix A, General Design Criteria (GDC); GDC 35 for ECCS design, GDC 38 for containment heat removal systems, and GDC 41 for containment atmosphere cleanup. In addition, the CSS will continue to provide a mechanism to reduce the accident source term to support meeting the limits of 10 CFR Part 100.

2. General Description of and Schedule for Corrective actions:

Using the guidance and requirements of NEI 04-07 (Reference 1), as described in Section 3 of this response, WCNOC has performed a comprehensive set of evaluations of the effects of design basis accident conditions on the ability of structures, systems and components, including the containment emergency sump strainers, to mitigate the consequences of the analyzed accidents and maintain long term core cooling in a manner consistent with governing regulatory requirements.

A mechanistic analysis was performed using a computational fluid dynamics (CFD) model to simulate the development of flow patterns during the recirculation phase.

New sump strainers were installed in the existing emergency recirculation sump pits in the containment building to accommodate the water levels postulated for post-accident conditions. Each new sump strainer contains stacked plates arranged into modules that maximize the strainer surface area. The strainer plates have perforated stainless steel plate surfaces with 0.045 inch diameter holes to efficiently capture debris that enters the sump pits. While the original containment recirculation sump screens and trash racks had approximately 200 ft^2 of effective surface area per sump, the new replacement sump strainers have approximately 3300 ft^2 of effective surface area per sump.

Debris barrier plates have been installed in openings through the secondary shield wall that are near the emergency recirculation sumps. The barriers prevent the "short path" flow of debris-laden fluid directly to the sumps and force the fluid to take a "long path" through shield wall openings farther away from the sumps to allow more time for the debris to settle out.

Changes were implemented for design change process procedures to ensure that necessary engineering evaluations will be performed when preparing a change to the plant design that either directly or indirectly affects containment, ECCS, or CSS. Administrative controls were added to the plant modification process to require a specific response to the following design issues that could impact long term cooling following design basis accidents.

- Holdup points or restrictions that could affect flow areas in the containment building.
- Modified piping, tanks or equipment that could change containment flood levels.
- Impact of debris downstream of the strainers on equipment, valves, instruments, or nuclear fuel.
- Added or changed materials that could become post-accident debris.
- Addition or removal of aluminum or zinc from the containment building.
- Introduction of un-qualified coatings or impact on qualified coatings.

Changes were made to the containment entry and material control procedure to enhance requirements during plant operational modes 1 through 4 for control of materials during work activities conducted in the containment and for control of radiological postings.

Changes were made to the clearance order procedure to ensure that Generic Letter 2004-02 analyses and evaluations are considered prior to making future changes to existing requirements that clearance order tags are not installed on components inside the containment being removed from service (tagged out) during plant operational modes 1 through 4.

Changes to the work request procedure to ensure that Generic Letter 2004-02 analyses and evaluations are considered prior to making future changes to existing requirements that work request tags are not installed on components inside the containment.

Changes to the scaffold construction and use procedure to enhance requirements for control of scaffold tags and materials used during work activities conducted in the containment during plant operational modes 1 through 4.

WCNOC has implemented a containment latent debris assessment program, which utilizes swipe sampling to determine the amount of latent debris in the containment building. Housekeeping and foreign materials exclusion procedures have been revised to target containment building cleaning based on the results of the swipe sampling survey.

WCNOC has implemented a containment coatings assessment program for monitoring and assessing the containment building coatings, including administrative controls on conducting coating examinations, including deficiency reporting criteria and documentation requirements.

WCNOC has implemented changes to Technical Specifications surveillance procedures to ensure that the installed replacement strainers will not have openings in excess of the maximum designed strainer opening.

Interim compensatory measures implemented at WCGS in accordance with NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," as described in References 16 and 17, remain in effect to minimize interim risks associated with post-accident debris blockage while GL 2004-02 evaluations are being completed.

In accordance with NRC Bulletin 2003-01 requirements, the following measures will remain in place at a minimum until all evaluations and corrective actions for GL 2004-02 are complete:

1. Ensuring that alternative water sources are available to refill the Refueling Water Storage Tank (RWST) or to otherwise provide inventory to inject into the reactor core and spray into the containment atmosphere

2. Licensed operators, Operations, Engineering and Emergency Response organization training on sump clogging issues

3. More aggressive containment cleaning and increased foreign material controls

4. Ensuring containment drainage paths are unblocked

5. Ensuring sump screens are free of adverse gaps and breaches

As described in Reference 4 and approved in Reference 5, GL 2004-02 corrective actions will be completed and implemented by June 30, 2008. Plant-specific tests that support assumptions and corresponding conclusions contained in the GL 2004-02 evaluations were completed in January, 2008. Although the actual test runs at the vendor testing facility are completed, numerous additional actions are required prior to the vendor completing a final test report that meets the procurement specifications requirements, including those of 10 CFR 50, Appendix B for control of purchased services and procurement document control. Preliminary test results indicate adequate net positive suction head (NPSH) margins for the ECCS and CSS pumps during the recirculation mode of operation.

In addition, following receipt of the final test report from the vendor, numerous additional actions are required to complete formal verification of design inputs, assumptions and conclusions of calculations and evaluations conducted in response to issues identified in GL 2004-02. These activities include assessing the impact of the test results on strainer NPSH calculations, strainer bypass sampling impact on downstream effects analyses (in-vessel and ex-vessel), as well as potential impact on other Generic Letter 2004-02 corrective action evaluations. These activities also include compliance with 10 CFR 50, Appendix B requirements for design control, document control and quality assurance records.

Descriptions of the results of the testing and the implementation of calculations, evaluations and other changes described in Reference 4 will be submitted to the NRC as an update to this supplemental response to GL 2004-02 by June 30, 2008. The following additional actions listed in Reference 4 to be completed are reproduced below:

1. Using WCAP-16406-P (Reference 6), evaluate the effects of debris-laden fluid on systems and components downstream of the containment emergency sump strainers during the ECCS recirculation phase of design basis accidents.
2. Using NEI 04-07, evaluate the effects of design basis accident conditions on the ability of structures, systems and components upstream of the containment emergency sump strainers to mitigate the consequences of the analyzed accidents.
3. Using NEI 04-07, perform a debris generation calculation for the analyzed design basis accidents.
4. Using NEI 04-07, perform a debris transport calculation for the analyzed design basis accidents.

5. Evaluate the impact of chemical effects on containment emergency sump strainer head loss during design basis accident conditions.
6. Complete head loss testing of the containment emergency sump strainer.
7. Confirm that the available NPSH of the containment emergency sump strainers during design basis accident conditions is in excess of the required NPSH.
8. Complete the final site acceptance review of the Westinghouse evaluation team analysis summary report.
9. The following items will be completed:
   a. Modify safety injection system components, if required, based on the results of the downstream effects evaluation.
   b. Remove the containment spray system pump cyclone separators, if required, based on the results of the downstream effects evaluation.
   c. Implement all plant modifications and related administrative controls that support the NEI 04-07 analysis package.

3. Specific Information Regarding Methodology for Demonstrating Compliance:

3a. Break Selection

The objective of the break selection process is to identify the break size and location that present the greatest challenge to post-accident sump performance.

1. *Describe and provide the basis for the break selection criteria used in the evaluation.*
2. *State whether secondary line breaks were considered in the evaluation (e.g., main steam and feedwater lines) and briefly explain why or why not.*
3. *Discuss the basis for reaching the conclusion that the break size(s) and locations chosen present the greatest challenge to post-accident sump performance.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).1 (Reference 2). The break selection process described below conforms to Sections 3.3.4.1 and 4.2.1 of NEI 04-07, Vol. 2 (SE) (Reference 1).

NEI 04-07, Vol. 2 (SE), Section 3.3.5.2 advocates break selection at 5-ft intervals along a pipe in question but clarifies that "the concept of equal increments is only a reminder to be systematic and thorough." It further qualifies that recommendation by noting that a more discrete approach driven by the comparison of debris source term and transport potential can be effective at placing postulated breaks. The key difference among the many postulated breaks (especially large breaks) is not the exact location along the pipe, but rather the envelope of material targets affected by the break. Loop D was specifically chosen because a loop D break results in the largest production of NUKON™ insulation debris due to its proximity to the pressurizer and pressurizer surge line. Specific break locations were selected by modeling the various zones of influence (ZOIs) along the reactor coolant system piping and maximizing the debris generated by those ZOIs.

3a.1 Description and Basis for Break Selection Criteria

The break selection process consisted of determining the size and location of the high energy line breaks that would produce debris and potentially challenge the performance of the recirculation sump strainer. The break selection process required evaluating a number of potential break locations in order to identify the location that would be likely to present the greatest challenge to post-accident sump performance. The debris inventory and the transport path were both considered when making this determination.

A sufficient number of breaks in each high-pressure system that rely on recirculation were considered to ensure that the breaks that bound variations in debris generation by the size, quantity, and type of debris were identified. Piping under 2 inches in diameter was excluded when determining the limiting break conditions (reference Section 3.3.4.1, Item 7, of NEI 04-07, Vol. 2 (SE)). The following break locations were considered for further analysis at Wolf Creek Generating Station (WCGS):

1. Breaks in the reactor coolant system (RCS) with the largest potential for debris,
   - Loop "A" crossover leg
   - Loop "D" crossover leg at the steam generator (S/G)
   - Loop "D" crossover leg at the reactor coolant pump (RCP)
   - Loop "D" crossover leg at the mid-point
   - Loop "D" hot leg at primary shield wall

2. Large breaks with two or more different types of debris,
   - Loop "A" crossover leg
   - Loop "D" crossover leg at the steam generator
   - Loop "D" crossover leg at the RCP
   - Loop "D" crossover leg at the mid-point
   - Loop "D" hot leg at primary shield wall

3. Breaks in the most direct path to the sump,
   - As described below, there are no breaks evaluated outside the secondary shield wall near the sump area because no loss of coolant accident (LOCA) breaks outside the secondary shield wall are postulated in the WCGS licensing basis.

4. Large breaks with the largest potential particulate debris to fibrous insulation ratio by weight,
   - Reactor vessel cold leg nozzle

5. Breaks with the potential to generate a "thin bed" – high particulate with thin fiber bed
   - Alternate charging line at the loop "D" cold leg

3a.2 Secondary HELB Scenarios

Secondary side line breaks (main steam line or main feedwater line) were evaluated to determine potential impact of Generic Letter 2004-02 issues on for long term core cooling. As long as the reactor coolant system remains intact, decay heat removal via at least one steam generator will be possible. The WCGS USAR does not describe a sequence of events that results in ECCS recirculation for accidents or events other than LOCAs. Therefore, breaks on the secondary side are not included in the Generic Letter 2004-02 evaluations.

3a.3 Basis for Break Sizes and Locations Chosen

The WCGS updated safety analysis report (USAR) describes small break LOCAs as a rupture of the RCS pressure boundary with a total cross-sectional area less than one square foot, and large break LOCAs as a rupture equal to or greater than one square foot cross sectional break area.

Loss of reactor coolant boundary limits (isolation points) assumed in the WCNOC licensing bases are defined in USAR Figure 3.6-2. Four high energy line break (HELB) cases are characterized for RCS-attached piping based upon flow and valve position. HELB locations and break types are shown in USAR Figure 3.6-1. In each of the piping configurations depicted in USAR Figure 3.6-1, the applicable LOCA boundary (isolation point) is located within the secondary shield wall. Therefore, consistent with the WCGS licensing basis, HELBs do not occur outside the secondary shield wall. Piping breaks outside the secondary shield wall are not included in the Generic Letter 2004-02 evaluations.

Breaks from the largest diameter piping (hot legs and crossover legs) and associated larger zone of influence (ZOI), create the greatest quantity of debris. Breaks in loop "D" were selected because additional debris would be generated due to the break's impact on the pressurizer piping. A break in loop "A" was selected as the highest potential particulate debris to fibrous insulation ratio by weight.

Although debris generated from a small break LOCA scenario is much less than from a large break LOCA, the water level in the WCGS containment building following a small break LOCA may not be sufficient to completely submerge the containment emergency sump strainers. Specifically, a three inch pipe break or smaller could result in RCS pressure remaining too high to allow discharge of water from the safety-injection accumulators. In addition, containment pressure could be too low to actuate the containment spray system. A break in the three inch alternate charging line was selected for evaluation of partially submerged emergency sump strainers.

A summary of the specific break locations chosen for evaluation is shown in Figure 3a-1 and described below.

- Break No. 1 (Case 1a) – Break at loop "D" cross-over leg at the base of the steam generator:

  The 31" ID crossover line between the S/G and the RCP is the largest of the RCS lines and due to the truncated configuration of the reactor cavity wall, has the largest target area in the adjacent loop. With a seven pipe-diameter (7D) radius applicable for NUKON™ insulation, the ZOI extends approximately 18.1 feet from the centerline of the pipe.

- Break No. 2 (Case 1b) – Break at loop "D" cross-over leg at RCP:

  Loop "D" is the loop with the largest debris source term due to its proximity to the pressurizer area. The 7D ZOI for NUKON™ insulation extends approximately 18.1 feet from the centerline of the pipe.

- Break No. 3 (Case 1c) – Break at loop "D" cross-over leg at mid point:

  A break at the mid-point of the crossover leg in loop "D" generates the largest quantity of coatings debris.

- Break No. 4 – Break at loop "D" hot leg near primary shield wall:

  The crossover mid-point and primary shield wall breaks were analyzed to maximize coating debris. It was determined through preliminary analysis that coating debris generated by this break was bounded by a break at the mid-point of the crossover leg. Therefore, no further analysis was performed for this case.

- Break No. 5 (Case 2) – Reactor vessel cold leg loop "A" nozzle break:

  A break at the reactor vessel cold leg nozzle was chosen because it had the largest potential particulate debris to fibrous insulation ratio by weight. A break within the reactor cavity could result in a rapid pressurization of the volume above the 2.4 psig destruction pressure specified for Diamond Power Mirror® Reflective Metal Insulation (RMI-M). Therefore, all the RMI-M insulation within the cavity was assumed destroyed. Additionally, the insulation on the cold leg piping outside the affected reactor cavity penetration was assumed destroyed. Loop A was selected since its NUKON™ volume was the largest among the hot/cold legs in the four loops.

- Break No. 6 (Case 3) – Small break LOCA:

  A break in the alternate charging line at the loop "D" cold leg was assessed to provide a debris value associated with the resultant lower water level. This line was chosen because it generates the most debris compared to other RCS-attached lines with diameter of 3 inches and smaller.

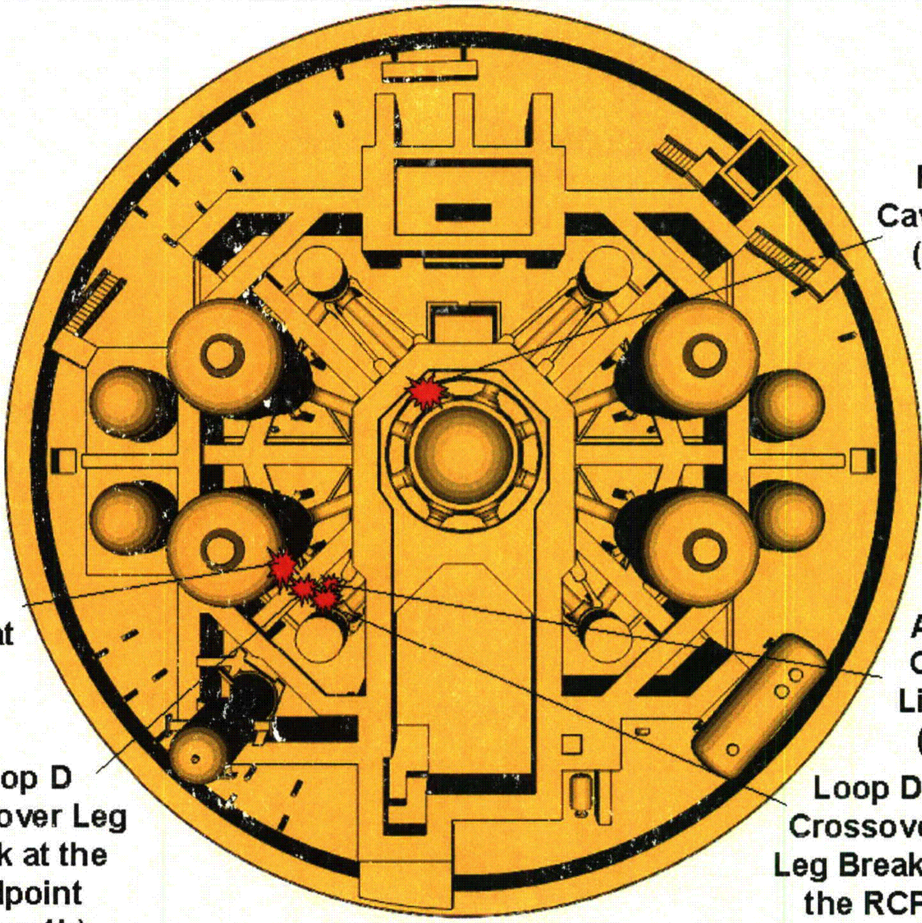


Figure 3a-1. Postulated Break Locations Evaluated

3b. Debris Generation/Zone of Influence (ZOI) (excluding coatings)

The objective of the debris generation/ZOI process is to determine, for each postulated break location: (1) the zone within which the break jet forces would be sufficient to damage materials and create debris; and (2) the amount of debris generated by the break jet forces.

1. *Describe the methodology used to determine the ZOIs for generating debris. Identify which debris analyses used approved methodology default values. For debris with ZOIs not defined in the guidance report/SE, or if using other than default values, discuss method(s) used to determine ZOI and the basis for each.*
2. *Provide destruction ZOIs and the basis for the ZOIs for each applicable debris constituent.*
3. *Identify if destruction testing was conducted to determine ZOIs. If such testing has not been previously submitted to the NRC for review or information, describe the test procedure and results with reference to the test report(s).*
4. *Provide the quantity of each debris type generated for each break location evaluated. If more than four break locations were evaluated, provide data only for the four most limiting locations.*
5. *Provide total surface area of all signs, placards, tags, tape, and similar miscellaneous materials in containment.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).2. (Reference 2) The debris generation/ZOI process described below conforms to Sections 3.4.2 of NEI 04-07, Vol. 2 (SE) (Reference 1), with the exception of ZOIs for Min-K, NUKON™ and Thermal-Wrap® (considered the same as NUKON™), as described below.

3b.1 Debris Generation/ZOI Methodology

The methodologies used by WCNOC to determine debris generation ZOIs are:

1. Apply jet impingement test results for Min-K and NUKON™ insulation. To be capable of simulating the postulated LOCA blow down, the initial conditions specified for the test facility were as follows:
    a. Initial temp of fluid source - 530°F ±25°F
    b. Initial pressure of fluid source – 2000 psia +0/-50 psi
    c. Nozzle size – 3 inches (nominal dimension); 3.5 inches (actual, measured dimension)
    d. Volume of fluid source – Sufficiently large as to allow for a 30-second blowdown simulation with a nominal 3-inch nozzle
2. Apply engineering judgment using materials properties comparisons for Thermal-Wrap® insulation. Thermal-Wrap and NUKON™ are both low-density fiberglass insulation with similar material properties. Thermal-wrap insulation is composed of 83-97% fibrous glass with 3-17% binder. NUKON™ insulation is composed of 85-96% fibrous glass with 4-15% binder. Due to the similarity in material properties between NUKON™ and Thermal-Wrap®, it is assumed that the material characteristics as well as destruction pressure and associated ZOI for Thermal-Wrap® are equal to those for NUKON™. This assumption is supported by the similarities in destruction data shown in Appendix II of NEI 04-07, Vol. 2 (SE) for both NUKON™ and Thermal-Wrap® insulation.
3. Use the value listed in Table 3-2 of NEI 04-07, Vol. 2 (SE) for RMI-M insulation, and
4. For debris materials not listed in NEI 04-07, not tested, and debris generation characteristics not well known, (Cerablanket®, Foamglas®, and Thermo-Lag), the maximum ZOI of 28.6D prescribed in Table 3-2 of NEI 04-07, Vol. 2 (SE) was used as a conservative estimate.
5. Robust barriers such as walls were considered in the debris generation analysis and resulted in the truncation of ZOI volumes. Large components such as steam generators and tanks were not considered to truncate the ZOI in the debris generation analysis. This is consistent with Section 3.4.2.3 of NEI 04-07, Vol. 2 (SE).

3b.2 Destruction ZOIs and Bases

The destruction ZOI is defined as the volume about the break in which the jet pressure is greater than or equal to the destruction damage pressure of the insulation, coatings and other materials impacted by the break jet. The size of the ZOI is defined in terms of pipe diameters of the piping assumed to break. The ZOI is defined as a spherical volume centered at the assumed piping break. Table 3b-1 describes the destruction pressures and associated ZOI radii used in the evaluation of impacted WCGS materials.

Table 3b-1. Destruction Zones of Influence

Material	Destruction Pressure (psig)	ZOI	Reference
RMI-M (with std. bands)	2.4	28.6	NEI 04-07, Vol. 2 (SE), Table 3.2
Jacketed NUKON™ (with std. Bands)	18.6	7.0*	Site-specific testing
Thermal-Wrap®	18.6	7.0*	Equivalent to NUKON™
Cerablanket®	N/A	28.6	Assumed max
Foamglas®	N/A	28.6	Assumed max
Fire Barrier (Thermo-Lag, Darmat KM1**)	N/A	28.6 (for Thermo-Lag only)	Assumed max (for Thermo-Lag only)
Min-K	N/A***	N/A***	Site-specific testing

* Jet impingement tests demonstrated a ZOI for NUKON™ of 5D. However, WCNOC used a value of 7D ZOI for NUKON™ in the debris generation calculations for additional conservatism.

** Even though Darmat KM1 fire barrier material is installed in the WCGS containment building, this material is not within the ZOI of any break locations evaluated. Therefore, no Darmat KM1 debris is generated or accounted for in the breaks analyzed.

*** Encapsulated Min-K insulation is located near the reactor vessel. Destructive testing of the simulated installed configuration demonstrated the encapsulation to be effective in precluding damage to the Min-K insulation. Therefore, no Min-K debris is generated or accounted for in the breaks analyzed.

3b.3 Destruction Testing

Testing was performed to determine the appropriate spherical-equivalent ZOIs for Min-K and NUKON™ insulation materials used inside containment. The testing was performed using a supply tank with subcooled fluid at 2000 pounds per square inch (psig) (+0/-50 psig) and 550°F (+/-25°F). The supply tank fluid volume was sufficiently large to allow for a 30-second blowdown with a nominal 3-inch nozzle. The initial fluid reservoir conditions were chosen so that test articles would be exposed to prototypical LOCA jet conditions in terms of pressure, temperature, time duration, and mass flux. To compensate for the fact that RCS conditions are 2250 psi, the test article was located such that the stagnation pressure at the point of jet impingement would be the same as using a supply tank at 2250 psi. The distance of the test article from the jet nozzle was calculated using the ANSI N58.2-1988 (Reference 15) jet expansion model. This testing was performed under the direction of Westinghouse Electric Company at Wyle Laboratories in Huntsville, Alabama.

The test program included two (2) types of insulation systems; encapsulated Min-K insulation and jacketed NUKON™ insulation. The purpose of the testing was to determine the behavior of the insulation systems under jet loads representative of those that would be generated under postulated large-break LOCA conditions. The specific test objectives of the test program were related to the type of insulation being tested:

1. For the encapsulated Min-K insulation system, demonstrate that the encapsulation protected the encased insulation from damage from jet impingement loads that the insulation system would experience due to a postulated LOCA in an as-installed configuration.

2. For the jacketed NUKON™ fiberglass insulation, determine the appropriate ZOI outside of which the woven fiberglass cloth-covered blanket containing the fiberglass insulation material will not experience sufficient damage such that the fiberglass material contained within the fiberglass cloth-covered pillow must be treated as debris generated by the postulated pipe break.

The encapsulated or jacketed insulation systems included in this test program were:

1. Min-K thermal insulation representing the as-installed configurations:
   a. Detector Well Panel, fully encapsulated and welded at seams
   b. Loop Piping Penetration Panel, fully encapsulated and welded at seams
   c. Reactor Pressure Vessel Top Head Panel, fully encapsulated and welded at seams

2. NUKON™ insulation representing the installed configurations:
   a. NUKON™ Insulation System consisting of fiberglass wool quilted with fiberglass scrim and covered with sewn fiberglass cloth, covered with a stainless steel jacket secured in place with latches at regular intervals.

Post-test observations from the testing of both encapsulated Min-K and stainless steel jacketed NUKON™ fiberglass insulation systems are summarized as follows:

1. The Detector Well encapsulated Min-K Panel was observed to remain intact with no loss of Min-K insulation material following the jet impingement at the distance from the jet nozzle that was tested.

2. The Reactor Pressure Vessel Top Head encapsulated Min-K Panel was observed to remain intact with no loss of Min-K insulation material following the jet impingement at the distance from the jet nozzle that was tested. See Figures 3b-1 and 3b-2 below for representative pre-test and post-test photos.

3. The Loop Piping Penetration encapsulated Min-K Panel was observed to remain intact with no loss of Min-K insulation material following the jet impingement at the distance from the jet nozzle that was tested.

4. For the 13 D ZOI test of jacketed NUKON™ insulation system, two of the three latches on the Stainless Steel Jacket were observed to have become detached after the jet impingement test; however the third latch held the jacket in place. No observable release or extrusion of fiberglass material from the woven fiberglass cloth-covered blanket was observed to have resulted due to jet impingement at the distance from the jet nozzle that was tested.

5. For the 10 D ZOI test of jacketed NUKON™ insulation system, on the 36 inch-long stainless steel jacket, the center latch was open and disengaged and the outer latches were open but engaged. The latches on the two 8 inch-long stainless steel jackets were engaged and closed. All of the stainless steel jacketing was observed to remain in place following this test.

   Two small tears in the outer 12-inch-long and 24-inch-long cloth-covered blankets were observed; these tears were evaluated to have been caused by movement of the stainless steel jacketing material. No loss of NUKON™ fiberglass insulation material from the woven fiberglass cloth-covered blankets, including the two small tears that were evaluated to have been caused by movement of the jacketing material, was observed to have resulted due to jet impingement at the distance from the jet nozzle that was tested.

6. For the 8 D ZOI test of jacketed NUKON™ insulation system, all of the stainless steel jacketing was observed to remain in place following this test. The latches on the 36-inch-long stainless steel jacket were open but remained engaged. The latches on the two 8-inch-long stainless steel jackets were engaged and closed. No loss of NUKON™ fiberglass insulation material from the woven fiberglass cloth-covered blankets was observed to have occurred due to jet impingement at the distance from the jet nozzle that was tested.

7. For the 6 D ZOI test of jacketed NUKON™ insulation system test, the 36 inch-long stainless steel jacket was ejected from the test fixture. The latches on the two 8-inch-long stainless steel jackets were engaged. The left side of the stainless steel jacket was dented by being punched up against the test fixture by the force of the jet.

   All of the NUKON™ insulation was saturated with water. The 6-inch right side outer layer “pillow” had a 1/2-inch hole. The 6-inch-long left side outer layer “pillow” had an 8-inch and 2-inch tear that was evaluated to have been caused by the test fixture. Both the 12-inch-long and the 24-inch-long outside layer “pillows” were off the pipe and lying on the ground in front of the test fixture. The 24-inch internal layer “pillow” remained on the pipe with the woven fiberglass cloth showing evidence of being stretched. The 12-inch-long internal “pillow” also remained on the pipe. No loss of NUKON™ fiberglass insulation material from the woven fiberglass cloth-covered blanket, including those “pillows” with holes, was observed to have resulted due to jet impingement at the distance from the jet nozzle that was tested.

8. For the 5 D NUKON™ insulation system, the 36-inch stainless steel jacket was ejected from the test fixture. On the 8-inch-long stainless steel jackets, both latches on the right jacket were closed and one disengaged. One latch on the left jacket was open and engaged on the opposite hook. The 6-inch-long NUKON™ blanket right side outer layer was observed to have a small hole. See Figure 3b-3 and 3b-4 below for representative pre-test and post-test photos.

Several significant observations are noted from the NUKON™ jet impingement tests, particularly those performed at small ZOI values:

1. While the stainless steel jacketing definitely protects the underlying NUKON™ insulation, the removal of the jacketing material by the impinging jet does not result in

the release of fibrous material from the woven fiberglass cloth-covered blanket or "pillow."

2. The direct impingement of the jet on a woven fiberglass cloth-covered blanket or "pillow" did not result in the failure of the woven fiberglass cloth-covered blanket material. Rather, the fabric stretched but did not release or allow the extrusion of fiberglass enclosed in the woven fiberglass cloth-covered blanket. This survivability of the woven fiberglass cloth-covered blanket was observed to a 5 D ZOI.

3. Small tears in the woven fiberglass cloth-covered blanket, evaluated to result from the movement of jacketing material resulting from forces exerted by jet impingement, did not result in the release or extrusion of the fiberglass material enclosed in the woven fiberglass cloth-covered blanket.

4. The test fixture was designed to represent the as-installed pipe insulation configurations; however, the bracing for the test rig was observed to result in some non-typical damage, but the observed insulation damage did not influence the test results. In only one case, the 5D ZOI test, did the interaction of the sacrificial end-pieces of NUKON™ with the test fixture result in the loss of a visually observable amount of fiberglass insulation material from the woven fiberglass cloth-covered pillow. This damage to the end-pieces in the test is not considered typical of plant behavior and is not expected to occur in the plant.

In summary, the testing performed demonstrates no debris generation from jet impingement loads typical of those resulting from a postulated large-break LOCA at the distances from the jet nozzle for any of the encapsulated Min-K insulation test articles considered in this program. Thus, the test observations of the encapsulated Min-K insulation under jet impingement loading provides the basis for the exclusion of the Detector Well Panels, the Reactor Pressure Vessel Top Head Panels and the Loop Piping Penetration Panels as debris sources within the containment building.

The testing also clearly demonstrates the acceptability of reducing the ZOI associated with the stainless steel jacketed NUKON™ thermal insulation from a spherical-equivalent ZOI of 17D to a value of 5D for piping and large components such as the Steam Generators and Pressurizer. However, for conservatism, a 7D ZOI was used for sump debris generation calculations.

Figure 3b-1 – Encapsulated Min-K Insulation Pre-Test Specimen

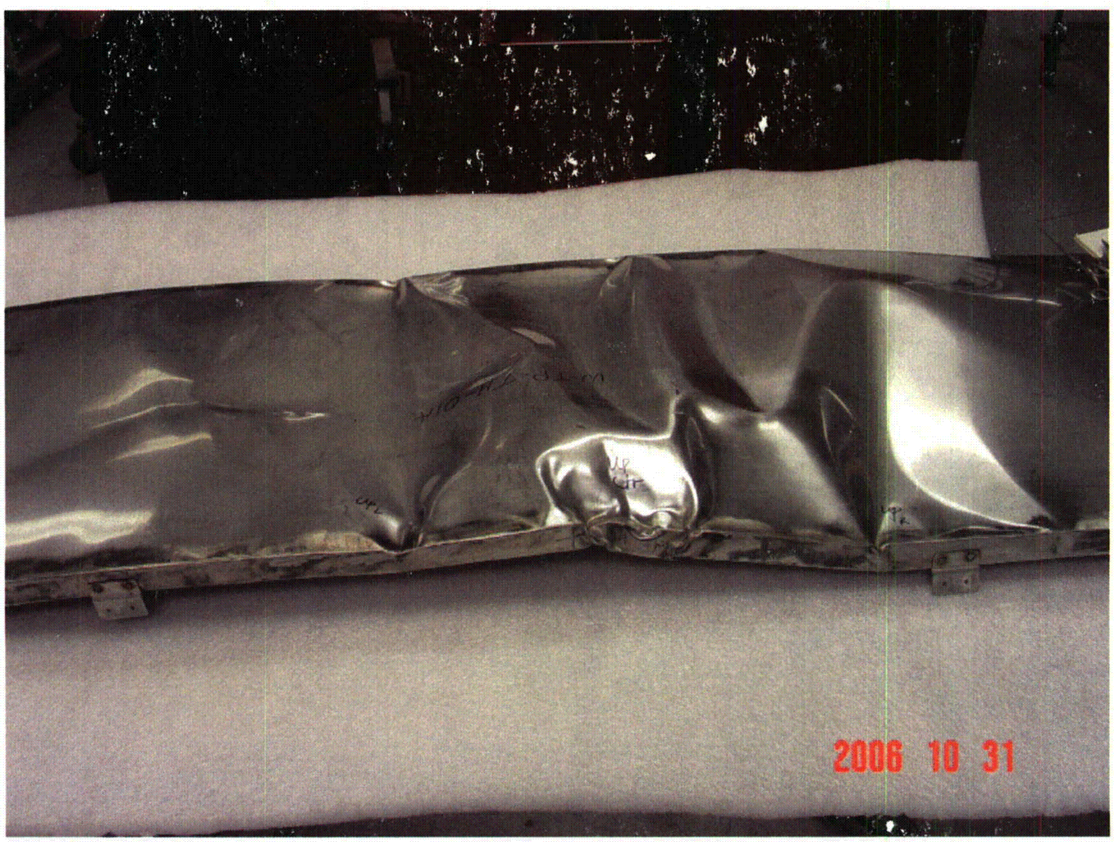


Figure 3b-2 – Encapsulated Min-K Insulation Post-Test Specimen

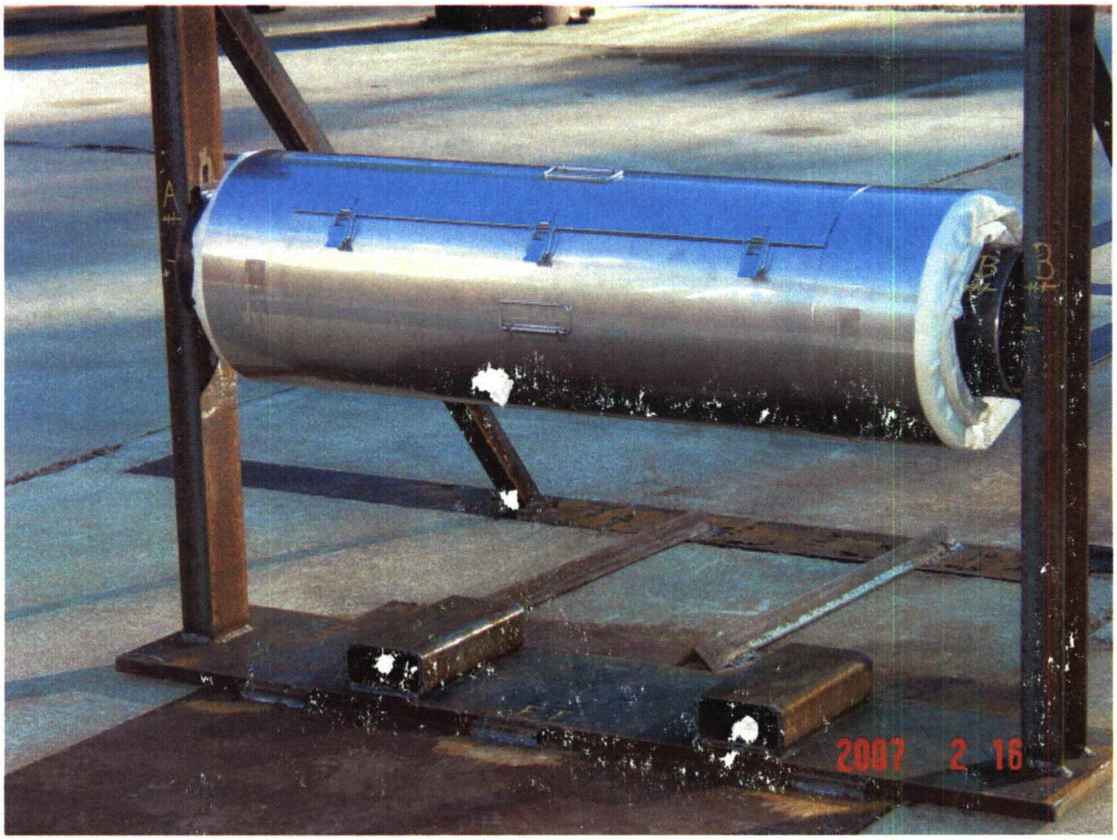


Figure 3b-3 – Jacketed NUKON Insulation 5D Pre-Test Specimen

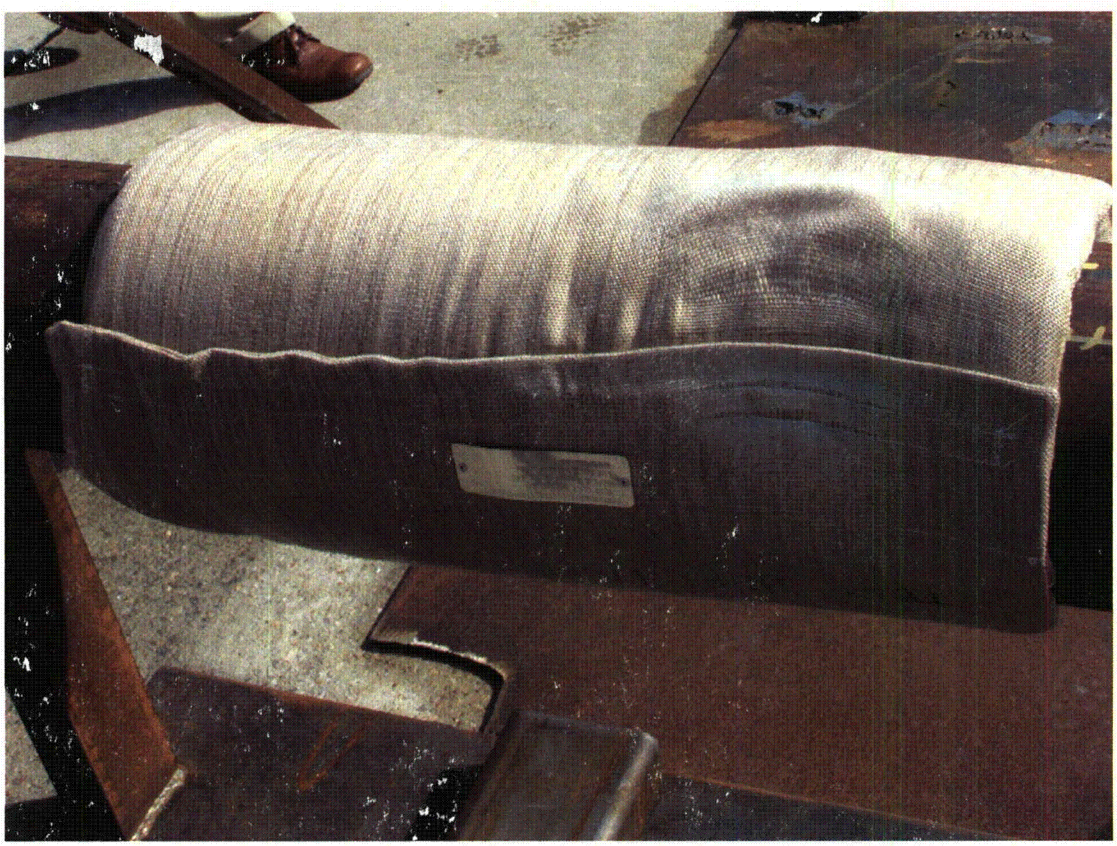


Figure 3b-4 – Jacketed NUKON Insulation 5D Post-Test Specimen

3b.4 Debris Types Generated for Each Break

Table 3b-2 provides the quantity of each debris type generated for the most limiting locations evaluated. Data for break #2 and break #3 described above are not included in Table 3b-2 since the results of the break evaluations showed that breaks #2 and #3 were bounded by the break #1 evaluation (Case 1a). Table 3b-2 includes data for break #1 (Case 1a), break #5 (Case 2) and break #6 (Case 3).

Table 3b-2. Debris Types Generated (Excluding Coatings)

Break Number	Debris Type	Debris	Amounts
Break #1 (Case 1a)	Fibrous	NUKON™ small fines	210.9 ft^3
		NUKON™ Large Pieces	140.6 ft^3
		NUKON™ Intact Blankets	142.1 ft^3
		Latent Fiber	12.5 ft^3
	Particulate	Thermo-Lag	25.3 lb
		Latent Particulate	170 lb
Break #5 (Case 2)	Fibrous	NUKON™ small fines	21.9 ft^3
		NUKON™ Large Pieces	14.6 ft^3
		NUKON™ Intact Blankets	0.0
		Cerablanket®	14.4 ft^3
		Latent Fiber	12.5 ft^3
	Particulate	Thermo-Lag	25.3 lb
		Foamglas®	5.6 lb
		Latent Particulate	170 lb
	RMI	RMI Small Pieces	4861 ft^2
		RMI Large Pieces	1620 ft^2
Break #6 (Case 3)	Fibrous	NUKON™ small fines	1.1 ft^3
		NUKON™ Large Pieces	0.7 ft^3
		NUKON™ Intact Blankets	2.3 ft^3
		Latent Fiber	12.5 ft^3
	Particulate	Latent Particulate	170 lb

3b.5 Surface Area of Miscellaneous Materials

Table 3b-3 lists the total surface area of all signs, placards, tags, tape and similar miscellaneous materials determined to be present in the WCGS containment for the areas analyzed.

Table 3b-3. Miscellaneous Debris Area

Debris Type	Unit Area (in^2)	Unit Quantity	Total Area (ft^2)
Tape	2" x 4"	45	2.5
Tape	2" x 10"	5	0.7
Fasteners	0.5" x 2"	90	0.6
Fasteners	1" x 5"	10	0.3
Personnel Protective Equip.	2" x 2"	50	1.4
High Temperature Phenolic Equip. Tags Note 1	3" x 5"	2363	246.1
Bakelite Equipment Tags	2.5" x 1"	850	14.8
Taped Equipment Labels	2" x 7"	656	63.8
Total			330.2

Note 1 - Equipment labels located inside a pipe break zone-of-influence (ZOI) were assumed to become dislodged and be transported to the containment flood pool during the accident blowdown phase (also see Section 3e-1). All other mechanically fastened equipment labels outside of the pipe break ZOI are assumed to remain attached. Site testing, using the containment recirculation water chemistry water conditions at 200°F, supports this assumption. The results of that testing provide confirmation that the equipment labels outside the ZOI would not degrade, thus remaining attached.

3c. Debris Characteristics

The objective of the debris characteristics determination process is to establish a conservative debris characteristics profile for use in determining the transportability of debris and its contribution to head loss.

1. *Provide the assumed size distribution for each type of debris.*
2. *Provide bulk densities (i.e., including voids between the fibers/particles) and material densities (i.e., the density of the microscopic fibers/particles themselves) for fibrous and particulate debris.*
3. *Provide assumed specific surface areas for fibrous and particulate debris.*
4. *Provide the technical basis for any debris characterization assumptions that deviate from NRC-approved guidance.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).3. (Reference 2) The debris characteristics determination process generally conforms, with some exceptions described below, to Sections 3.4.3 and 4.2.2 of NEI 04-07, Vol. 2 (SE) (Reference 1).

3c.1 Debris Size Distribution

Table 3c-1 shows the assumed size distribution in the debris generation calculation for each debris type. The debris distribution percentages are consistent with Section 3.4.3 of NEI 04-07, Vol. 2 (SE).

Section 3.4.3.3 of NEI 04-07, Vol. 1 (GR) (Reference 1) classifies the destroyed insulation debris in two categories of small fines and large pieces. Small fines fibrous debris classification is considered debris that passes through a 4" x 4" opening. Debris that does not pass through a 4" x 4" opening is considered large pieces.

Table 3c-1. Assumed Size Distribution by Insulation Debris Type

Insulation Debris Type	Size	Distribution
NUKON™	Small fines[1]	60% (5.0 L/D) 0% (7.0-5.0 L/D)
NUKON™	Large Pieces (>4" on a side)	40% (5.0 L/D) 0% (7.0-5.0 L/D)
NUKON™	Intact (covered) Blankets	100% (7.0-5.0 L/D)
RMI (inner foils)	4" and smaller	75%
RMI (inner foils)	4" to 6"	25%

1. A representative amount of "small fines" was checked to determine how much "fines" was contained in the "small fines" due the debris processing activities. The results showed that approximately 30% of the "small fines" are "fines".

3c.2 Fibrous and Particulate Debris Characteristics

Table 3c-2 shows the debris amounts, bulk densities, material densities and characteristic diameters for fibrous debris other than latent debris. The values were obtained from NEI 04-07, Vol. 1 (GR) Table 3-2, which has been recognized by NEI 04-07, Vol. 2 (SE), Section 3.4.3.6, and are discussed further below.

Table 3c-2. Fibrous Material Characteristics

Debris material	As-Fabricated Density (lb/ft^3)	Microscopic Density (lb/ft^3)	Characteristic Diameter (μm)
NUKON™	2.4	159	7.0
Thermal-Wrap™ *	2.4	159	7.0
Cerablanket®.**	8	158	3.2

*Material characteristics for Thermal-Wrap™ are assumed to be equivalent to NUKON™.

**100% of Cerablanket® inside the ZOI is assumed to fail as small fines. This is consistent with Section 3.4.3.3 of NEI 04-07, Vol. 2 (SE).

Table 3c-3 shows the debris amounts, solid density and diameter for particulate debris other than latent debris.

Table 3c-3. Particulate Debris Characteristics

Debris Material	Microscopic Density (lb/ft^3)	Characteristic diameter (μm)
IOZ Coating (inside ZOI)	457	10
Unqualified IOZ (outside ZOI)	457	10
Carboline 191 Epoxy (inside ZOI)	104	10
Carboline 890 Epoxy (inside ZOI)	112	10
Carboline 195 Epoxy (inside ZOI)	228.5	10
Carboline 4674 Acrylic (inside ZOI)	86	10
Unqualified Epoxy (Outside ZOI)	112	10
Unqualified Alkyd (Outside ZOI)	98	10
Thermo-Lag	43.6	10

3c.3 Fibrous and Particulate Debris Surface Areas

The specific surface areas for fibrous and particulate debris were generally used in the preliminary prediction of head loss with the NUREG/CR-6224, "Correlation and Dearation Software Package". Since the head loss across the installed recirculation strainers was determined via testing, the NUREG/CR-6224 prediction was used for comparison only.

3c.4 Basis for Debris Characteristic Assumptions

The debris characteristics assumptions are consistent with NEI 04-07, Vol. 2 (SE), with the exception of Cerablanket®. Physical properties for Cerawool® listed in NEI 04-07, Vol. 1 (GR) Table 3-2 were assumed for Cerablanket® installed in the reactor cavity at WCGS. Product information lists densities from 3 to 8 lb/ft^3. The highest fabricated density was assumed for conservatism.

3d. Latent Debris

The objective of the latent debris evaluation process is to provide a reasonable approximation of the amount and types of latent debris existing within the containment and its potential impact on sump screen head loss.

1. *Provide the methodology used to estimate quantity and composition of latent debris.*
2. *Provide the basis for assumptions used in the evaluation.*
3. *Provide results of the latent debris evaluation, including amount of latent debris types and physical data for latent debris as requested for other debris under c. above.*
4. *Provide amount of sacrificial strainer surface area allotted to miscellaneous latent debris.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).4 (Reference 2). The latent debris evaluation process described below conforms to Sections 3.5 and 4.2.2.2 of NEI 04-07, Vol. 2 (SE) (Reference 1). The documented amount of latent debris in the containment building from the baseline survey was estimated to be is less than 70 pounds. However, a value 200 pounds of latent debris was used in the debris generation calculation to provide margin for future assessments and for additional conservatism.

3d.1 Methodology used to estimate quantity and composition of latent debris:

A Containment Latent Debris Sampling Plan was developed for WCGS. Representative sample areas are selected prior to beginning the sampling process. Sample area categories include floor areas, wall areas, top surface of major equipment and top surfaces of major piping. Clean Masolin cloths are bagged and pre-weighed, prior to being used to swipe the selected sample areas. The bagged clothes are weighed again after the sample areas have been swiped. The weights from the sample areas are then multiplied by the total area for each representative area category and added together to give the total weight of latent debris in containment.

Conservatism was assured in the determination of latent debris loads by overestimating the surface areas of floors, major equipment, piping, HVAC ductwork, and electrical raceways.

The composition of the latent debris mixture is assumed to be 85% dirt/dust and 15% latent fibers, consistent with Section 3.5.2.3 of NEI 04-07, Vol. 2 (SE).

3d.2 Technical basis for assumptions used in the evaluation:

Representative sampling methodology, latent debris mixture, and characteristics assumptions are consistent with Sections 3.5.2.2 and 3.5.2.3 of NEI 04-07, Vol. 2 (SE).

3d.3a Results of latent debris evaluation:

The Containment Latent Debris Sampling Plan was implemented at WCGS for the first time during Refueling Outage XIV in April, 2005. Its purpose was to obtain a baseline amount of latent debris existing in the containment building. The results indicated that approximately 62 pounds of latent debris was present in the containment building.

For analysis purposes, latent debris amount in containment was increased from the 62 lbs observed in Refueling Outage XIV to 200 lbs in the analysis to provide additional margin for the WCNOC containment latent debris assessment program. A distribution of 85% dirt/dust and 15% fibers is assumed, consistent with Section 3.5.2.3 of NEI 04-07, Vol. 2 (SE). The debris distribution used for this baseline analysis, therefore, was 170 lbs dirt/dust and 30 lbs latent fiber.

3d.3b Fibrous and Particulate Latent Debris Characteristics

Table 3d-1 shows the bulk density, material density and characteristic diameter for fibrous latent debris. These values are consistent with Section 3.5.2.3 of NEI 04-07, Vol. 2 (SE).

Table 3d-1. Fibrous Material Characteristics for Latent Debris

Debris material	As-Fabricated Density (lb/ft^3)	Microscopic Density (lb/ft^3)	Characteristic diameter (μm)
Latent Fiber	2.4	94	7.0

Table 3d-2 shows the solid density and characteristic diameter for particulate latent debris. These values are consistent with Section 3.5.2.3 of NEI 04-07, Vol. 2 (SE).

Table 3d-2. Particulate Debris Characteristics for Latent Debris

Debris Material	Microscopic Density (lb/ft^3)	Characteristic diameter (μm)
Latent Particulate (dirt/dust)	169	17.3

3d.4 Amount of strainer surface area allotted for miscellaneous latent debris:

WCNOC did not allot sacrificial strainer surface area to account for miscellaneous latent debris, because sump strainer qualification testing included miscellaneous latent debris.

3e. Debris Transport

The objective of the debris transport evaluation process is to estimate the fraction of debris that would be transported from debris sources within containment to the sump suction strainers.

1. *Describe the methodology used to analyze debris transport during the blowdown, washdown, pool-fill-up, and recirculation phases of an accident.*
2. *Provide the technical basis for assumptions and methods used in the analysis that deviate from the approved guidance.*
3. *Identify any computational fluid dynamics codes used to compute debris transport fractions during recirculation and summarize the methodology, modeling assumptions, and results.*
4. *Provide a summary of, and supporting basis for, any credit taken for debris interceptors.*
5. *State whether fine debris was assumed to settle and provide basis for any settling credited.*
6. *Provide the calculated debris transport fractions and the total quantities of each type of debris transported to the strainers.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).5 (Reference 2).

In summary, the WCGS containment is considered a "mostly uncompartmentalized containment", which is defined as a containment that has partial robust structures surrounding the steam generators. The debris transport evaluation process information described below provides the methodology used to estimate the fraction of debris that is transported from the debris source (break location) to the sump strainers. This methodology is based on Sections 3.6 and 4.2.4 of NEI 04-07, Vol. 1 (GR) and the associated sections in NEI 04-07, Vol. 2 (SE) (Reference 1). Exceptions taken to the methodologies suggested by NEI 04-07, Vol. 2 (SE) are identified and justified below.

3e.1 Methodology Used to Analyze Debris Transport

The four major phases of debris transport are blowdown, washdown, pool fill-up, and recirculation phases of an accident.

Blowdown Phase Transport – The vertical and horizontal transport of debris to all areas of the containment building by the break jet forces.

Washdown Phase Transport – The vertical (downward) transport of debris by the containment spray system flow and the break flow.

Pool Fill-up Phase Transport – The transport of debris by break flow and containment spray system flow from the Refueling Water Storage Tank (RWST) to regions that may be active or inactive during recirculation. The areas below the containment floor elevation that fill up with the water/debris mixture and then remain stagnant for the remainder of the analyzed accident, are referred to as inactive areas of the pool. Other areas of the pool are referred to as active areas.

Recirculation Phase Transport – The horizontal transport of debris from the active portions of the recirculation pool to the sump strainers by the flow through the Emergency Core Cooling System (ECCS).

Each phase of debris transport was analyzed for each type of debris generated and a logic tree was developed to determine the total transport to the sump strainers. The purpose of this approach was to break a complicated transport problem down into specific smaller problems that were more easily analyzed.

The size distribution and characterization for the specific debris types determined in the debris generation calculation were used as input to the debris transport calculation. The debris transport logic tree shown in Figure 3e-1 below is based on the un-quantified logic tree that appears in NEI 04-07, Vol. 1 (GR). Unlike the logic tree that appears in the GR, the logic tree shown below contains entry locations for erosion and direct transport to the sumps during pool fill-up. Also, the logic tree is expanded to account for a more refined debris size distribution. Some examples of specific logic trees used in the calculation are provided in Section 3e.6 below.

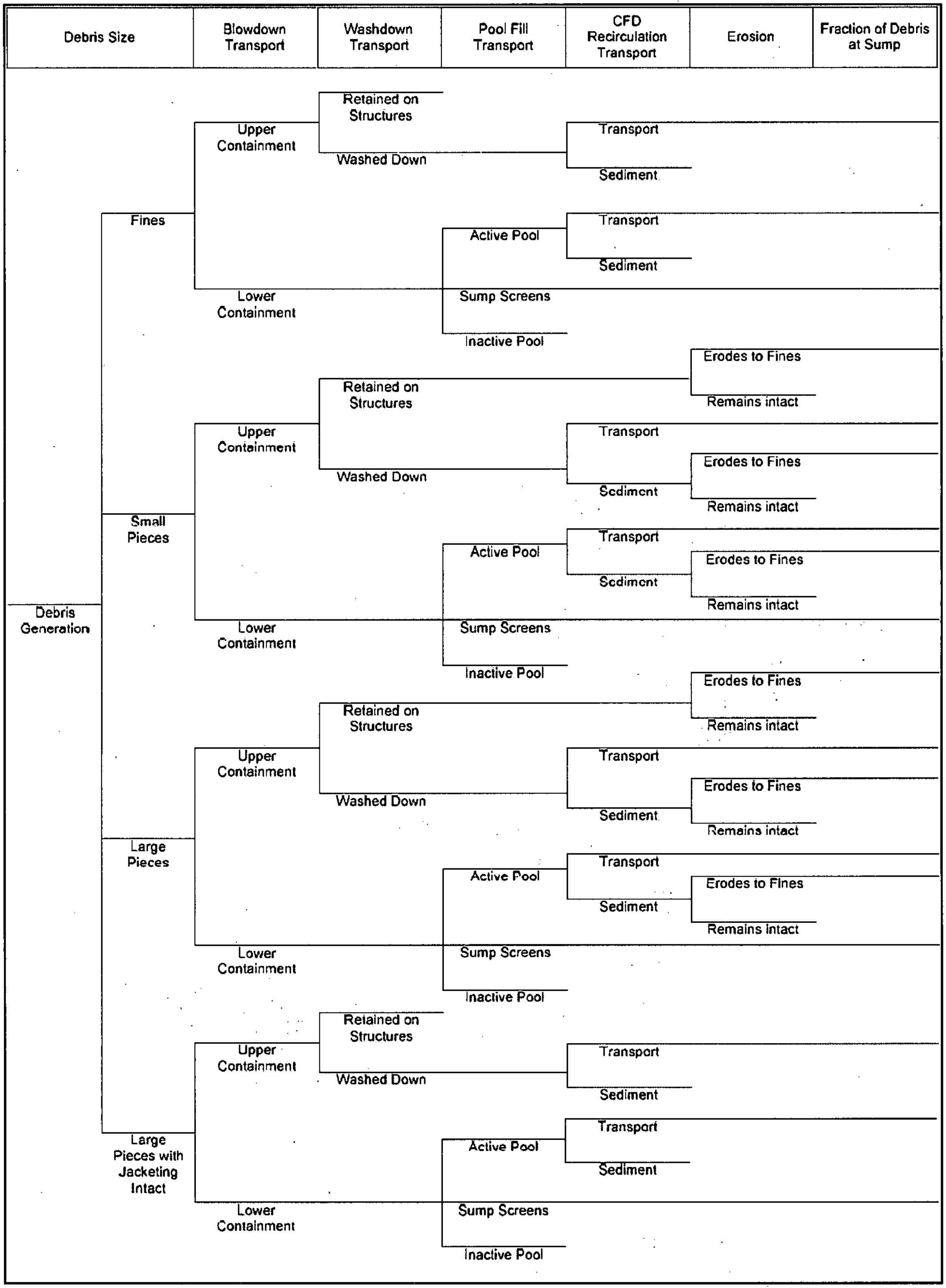


Figure 3e-1. Generic Debris Transport Logic Tree

Based on containment building design drawings, a three-dimensional model was built using computer aided drafting (CAD) software to facilitate determination of transport flow paths. Potential upstream blockage points including screens, fences, grating, drains, etc. that could lead to water holdup are addressed. Debris types, size distributions, and quantities from the debris generation calculation were compiled for each postulated break location. The fraction of debris blown into upper containment was determined based on the volumes of upper and lower containment. The quantity of debris transported to inactive areas or directly to the sump strainers was calculated based on the volume of the inactive and sump cavities proportional to the water volume at the time these cavities are filled. Using this methodology, the location and quantity of each type and size of debris at the beginning of the recirculation phase was determined.

A computational fluid dynamics (CFD) model (using Flow-3D® Version 9 software) was developed to simulate the development of flow patterns during the recirculation phase. The mesh in the CFD model was nodalized to sufficiently resolve the features of the CAD model, while keeping the cell count low enough for the simulation to run in a reasonable amount of time. The boundary conditions for the CFD model were set based on the configuration of the WCGS containment building and ECCS systems during the recirculation phase. The containment spray system flow was included in the CFD calculation with the appropriate flow rate and kinetic energy to accurately model the effects on the containment pool. At the postulated break locations, mass sources were added to the model to introduce the appropriate flow rate and kinetic energy associated with the break flow. Mass sinks were added at the sump locations with a total flow rate equal to the sum of the break and spray flow. A renormalization group theory turbulence model, was selected for the CFD calculations. After running the CFD calculations, the mean kinetic energy and other relevant parameters were checked to verify that the model had been run long enough to reach steady-state conditions. Transport metrics for each significant debris type were determined for each significant debris type present in Containment. Other miscellaneous debris (i.e., as listed in Table 3b-3) was conservatively assumed to transport 100%.

A graphical determination of the transport fraction of each type of debris was made using the velocity and turbulent kinetic energy (TKE) profiles from the CFD model output, along with the determined initial distribution of debris. The recirculation transport fractions from the CFD analysis were gathered to input into the logic trees. The quantity of debris that could experience erosion due to the break flow or spray flow was determined. Finally, the overall transport fraction for each type of debris was determined by combining each of the previous steps in logic trees.

A rectangular mesh was defined in the CFD model that was fine enough to resolve important features, but not so fine that the simulation would take prohibitively long to run. A 6-inch cell length was chosen as the largest cell size that could reasonably resolve the concrete structures that are found in the WCGS containment area. For the large break LOCA CFD model, the cells nearest the floor (both inside and outside the primary shield wall) were set to 3-inches tall in order to closely resolve the vicinity of settled debris. For the small break LOCA CFD model, the vertical cell size was set to 3 inches for each cell. The total cell count in the large break LOCA model was 1,019,200 and the total cell count in the small break LOCA model was 470,400.

The large break and small break LOCA CFD models were started from stagnant conditions at pool depths of 2.09 ft and 0.741 ft respectively, and were run long enough for steady-state conditions to develop. The models were run for a total of 5 minutes of real time for the large break LOCA case, and 16-1/2 minutes of real time for the small break LOCA case. The velocity and TKE results for both of these cases reflected steady-state conditions.

Even though the break locations chosen for evaluation were on loops "A" and "D," debris transport evaluations were performed assuming the break locations were in the vicinity of loop "C." It is assumed that debris transport evaluations for the break locations modeled for the large break LOCA and small break LOCA in the vicinity of loop "C" are conservative compared to breaks in other locations inside the secondary shield wall. This assumption is reasonable since the debris barriers at the entrances to loops "A" and "D" limit the amount of debris going through the entrance while forcing the break flow to take a longer path through the other two entrances. Since the "A" and "D" debris barriers are perforated plates and since the amount of flow that passes through the debris barriers is not modeled, it is expected that the actual velocity out the "B" and "C" loop openings would be conservatively lower than the flow determined from the transport analysis. Also, since all of the debris not blown to upper containment was conservatively assumed to wash outside the primary shield wall, the actual location of the modeled break flow does not significantly affect debris transport.

It was assumed that the reflective metal insulation (RMI) debris generated by the reactor cavity break did not transport to the sump strainers. This assumption is reasonable since most of the RMI generated by this break remained in the reactor cavity and settled to the floor, as shown in Figure 3e-2 below. Any RMI blown out of the cavity would not likely transport, since the energy from the break flow would be largely dissipated when it reaches the main pool.

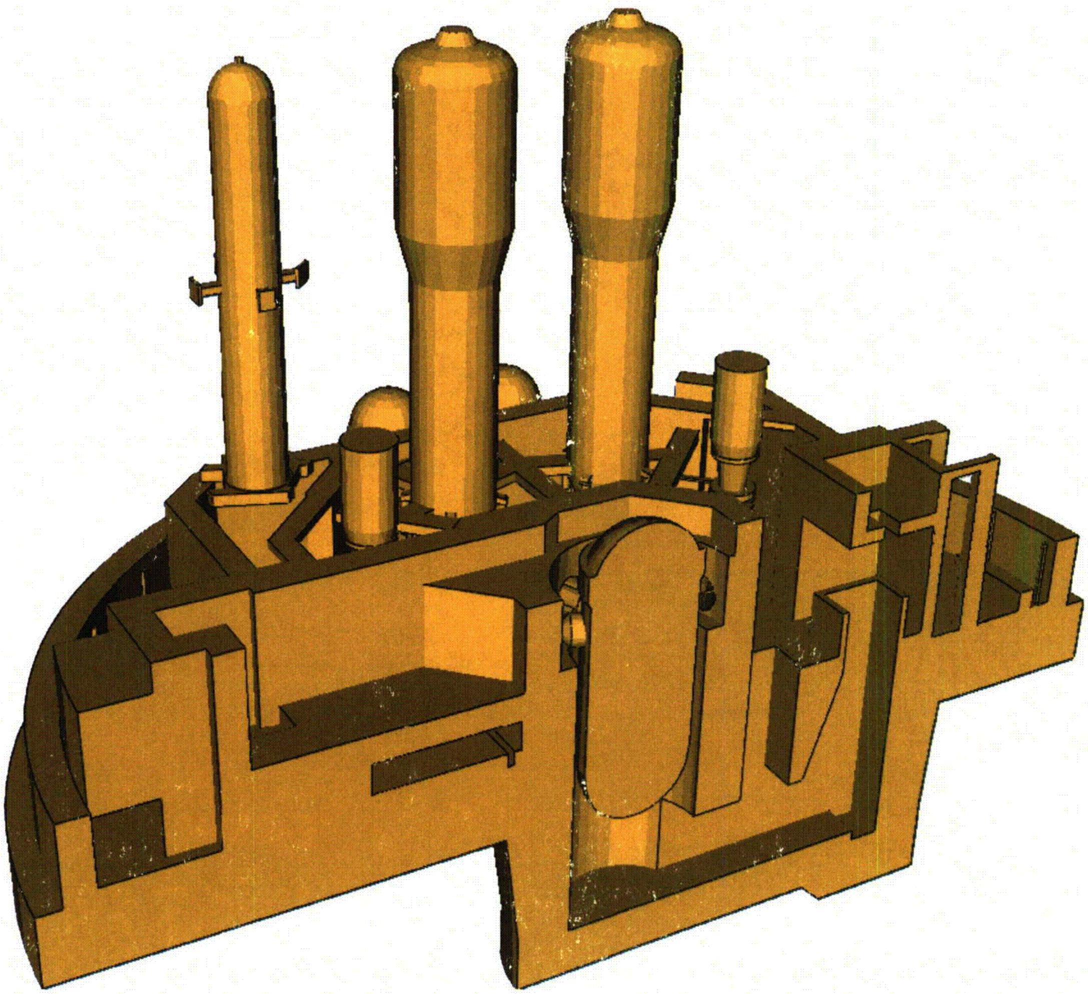


Figure 3e-2. Lower Containment Building Cross-Section

It was conservatively assumed that the fiberglass small fines debris transports similarly to individual fibers. This is a conservative assumption since individual fibers are readily transportable.

It was assumed that the large pieces of fiberglass debris (larger than 6") can be treated as 6" pieces. This is a conservative assumption since smaller pieces of fiberglass debris transport more readily. It was also assumed that intact pieces of fiberglass did not transport in the containment pool. See Section 3.e.2 below.

It was assumed that ¼"-4" pieces of RMI and fiberglass debris can be treated as ½" pieces and 4"-6" pieces can be treated as 2" pieces for recirculation transport. This is conservative since smaller pieces of RMI debris transport more readily.

It was assumed that the RMI debris does not break down into smaller pieces following the initial generation. This assumption is reasonable since RMI is a metallic insulation that is not subject to erosion by the flow of water.

It was assumed that the settling velocity of particulate debris (insulation, dirt/dust and coatings) can be calculated using Stokes' Law. This assumption is reasonable since the particulate debris is generally spherical, small in size and would settle slowly (within the applicability of Stokes' Law). Although non-uniformities in the particulate debris could cause it to settle slightly slower than a perfect sphere, this potential non-conservatism is offset by using a lower water temperature for the calculation, which results in a slower settling velocity.

Based on fibrous debris testing documented in NUREG/CR-2791, (Reference 10) it was assumed that NUKON™ debris does not float on the containment pool. Test data documented in NUREG/CR-2791 shows that fiberglass insulation sinks more readily in hotter water. Therefore, given the initial high temperature of the containment pool at WCGS, no floating debris was assumed in the transport analysis. During recent vendor testing activities, a small amount of fiber was observed on the water surface, but the amount was very small and is considered to have an insignificant impact on the results of the transport analysis.

Due to a lack of test data showing the turbulence and tumbling velocity metrics for miscellaneous debris, it was conservatively assumed that all transportable miscellaneous debris identified in the debris generation calculation including tags, labels, etc. were transported to the sumps during recirculation. The transportability of miscellaneous debris was tested during clean strainer head loss testing and most of the miscellaneous debris was shown to settle out before reaching the sump strainers. This is discussed further in Section 3f.4.

It was assumed that floor drains in upper containment become clogged with debris and that spray water was forced to flow off the floors through grated openings to the pool. This is a conservative assumption since the drains discharge at the two normal sumps and the turbulence of the spray water would be largely dissipated before reaching the pool (compared to the sprays falling at freefall or terminal velocity through the grated openings).

It was assumed that fines generated by the large break LOCA blast were transported to upper containment in proportion to the volume of upper containment compared to the volume of the entire containment area. This is a reasonable assumption since fine debris generated by the LOCA jet would be easily entrained and carried with the blowdown flow. Note that the break jet would not necessarily be directed toward upper containment. However, as the lower containment pressurizes, a significant portion of the blowdown flow would move toward upper containment.

It was assumed that a fraction of small and large piece debris was transported to upper containment in proportion to the containment volumes.

It was assumed that intact pieces of fiberglass did not transport in the containment pool. Large pieces of fiberglass insulation with the fiberglass blanket cloth intact are essentially the original full insulation blankets that have been blown off of piping or equipment. Given the size of these pieces and the potential for the jacketing to get caught on miscellaneous piping or equipment, the intact blankets would not be likely to transport far. Since the "A" and "D" loops have debris barriers installed, in order for intact pieces of fiberglass to transport to the sumps, it would have to be transported out the "B" and "C" doors, around the annulus and over the curb. Given the distance and the numerous obstructions along this path, none of the intact pieces of fiberglass were assumed to transport.

It was conservatively assumed that all debris blown upward would be subsequently washed back down by the containment spray system flow with the exception of any pieces of fiberglass or RMI debris held up on grating. Note that debris blown up into holdup areas protected from the containment spray path (on the primary shield walls, the shield walls around the pressurizer and on the bottom side of over head floor slabs) was conservatively assumed to all be washed back down with the containment spray, with the exception of the debris held up on grating. The fraction of debris washed down to various locations was determined on the spray flow split.

The large piece debris blown to upper containment was assumed to remain in upper containment. As discussed in NEI 04-07, Vol. 1 (GR), the shallow flow on the operating deck is not conducive to transporting large pieces of debris on the operating deck floor. Also, any debris landing on grating will not pass through the grating.

During pool fill-up, it was assumed that a fraction of the debris was transported to inactive areas, as well as directly to the sump strainers as the sump cavities fill with water. These fractions were based on the ratio of the cavity volumes to the pool volume at the point in time when the cavities were filled.

It was assumed that debris generated by the small break LOCA blast was not blown into upper containment. This is a conservative assumption since no credit was taken for holdup of debris in upper containment. Note that this assumption is also reasonable, since a small break would be less likely to blow debris into upper containment than a large break.

With the exception of latent debris washed to the sumps and inactive cavities during pool fill-up, it was conservatively assumed that all latent debris was in lower containment and was uniformly distributed in the containment pool at the beginning of recirculation. This is a conservative assumption since no credit was taken for debris remaining on structures and equipment above the pool water level.

It was assumed that the debris washed down from upper containment by the spray flow remained in the general vicinity of the location where it was washed down until recirculation was initiated. This is a reasonable assumption since pool velocities are low in the area outside the secondary shield wall during pool fill-up after the inactive and sump cavities have been filled. Also, this assumption is somewhat conservative since the local turbulence caused by the sprays would increase the potential for debris to be transported from these locations.

With the exception of debris washed directly to the sump strainers or to inactive areas, it was assumed that the fine debris that was not blown to upper containment was uniformly distributed in the recirculation pool at the beginning of recirculation. This assumption is based on the fact the settling velocity of fine debris is quite small, so the fine debris tends to remain in suspension during pool fill-up and is mixed throughout the pool by the break and spray flows.

It was assumed that small and large piece debris that was not blown to upper containment was distributed between the break location and the sumps at the beginning recirculation. This is a conservative assumption since it neglects the fact that some debris would be blown or washed to areas farther away from the sump during the blowdown and pool fill-up phases.

The water falling from the reactor coolant system (RCS) breach was assumed to do so without encountering any structures before reaching the containment pool. This is a conservative assumption since any impact with structures dissipates the momentum of the water and decreases the turbulent energy in the pool.

It was assumed that the miscellaneous penetrations in the secondary shield wall would be excluded from the CFD model. It is possible that some small piece debris could be washed through these penetrations and transport to the sumps. Note, however, that most of these penetrations are at or above the minimum water level elevation. Therefore, since pieces of debris would have to be carried in suspension to pass through the penetrations, only a small quantity of debris will likely to pass through these penetrations. Since modeling a higher water level or realistically modeling the flow through the penetrations would reduce the pool velocities, and therefore reduce debris transport in the annulus, it is conservative to exclude these penetrations from the analysis.

3e.2 Technical Basis for Analysis Assumptions that Deviate from NEI 04-07, Vol. 2 (SE)

Due to limited test data, the Section III.3.3 NEI 04-07, Vol. 2 (SE) stipulates the use of a 90% erosion fraction for fiberglass in the containment pool. Vendor erosion testing, which was confirmed to apply to the transport analysis, showed that the erosion fraction would be less than 10%. Therefore, an erosion factor of 10% was used in the transport analysis.

3e.3 Computational Fluid Dynamics Codes

The CFD calculation for recirculation flow in the Containment pool was performed using Flow-3D® Version 9.0 using the vendor's modified subroutine. Flow-3D® is a commercially available general-purpose computer code for modeling the dynamic behavior of liquids and gasses influenced by a wide variety of physical processes. The program is based on the fundamental laws of conservation of mass, momentum, and energy. It has been constructed for the treatment of time-dependent multi-dependent multi-dimensional problems and is applicable to most flow processes including containment recirculation pool modeling. Flow-3D® is configuration-controlled under the vendor's QA program, which contains a varied collection of exacting test problems. Version 9.0 (with the modified subroutine) has been validated and verified under the vendor's QA program. The subroutine modification to the standard Flow-3D code was to enable the introduction of containment sprays at the appropriate source locations near the surface of the pool and still model the appropriate flow rates and velocities for each spray location. Using the modified version of Flow-3D, up to 10,000 mass source particles were able to be placed in discreet locations.

Various volume and area calculations were performed using AutoCAD® 2007, AutoCAD® 2008, and Inventor® 2008. These software packages are commercially available computer codes. The CAD software used in the transport analysis is configuration-controlled under the vendor's QA program similar to the Flow-3D® software. AutoCAD® 2007, AutoCAD® 2008 and Inventor® 2008 have been validated and verified under the vendor's QA program.

3e.4 Debris Interceptors

While WCGS does not have debris interceptors, debris barriers have been installed in all openings through the secondary shield wall near the emergency recirculation sumps. The barriers prevent the flow of debris-laden fluid directly to the sumps and force the fluid to take a longer "tortuous path" through shield wall openings farther away from the sumps. Debris barriers have been installed in the Loop A and Loop D passageway entrances through the secondary shield wall, as well as in drain trenches and other openings in the secondary shield wall near the sumps. The debris barriers at the Loop A and Loop D entrances and the drain trenches are fabricated using perforated plate with 1/8" holes to restrict passage of debris while allowing water to pass through the barrier.

3e.5 Fine Debris Settling Credit

Fine debris is assumed to settle according to Stokes' Law. The use of Stoke's law is reasonable since the particulate debris is generally spherical, small in size and would settle slowly (within the applicability of Stokes' Law). Although non-uniformities in the particulate debris could cause it to settle slightly slower than a perfect sphere, this potential non-conservatism is offset by using a lower water temperature for the calculation, which results in a slower settling velocity.

3e.6 Debris Transport Fractions and Total Debris Transported

The following four logic trees are examples of debris transport fractions for NUKON™ debris transport from the Case 1a break evaluation, a break in loop "D" cross-over leg at the base of the steam generator. Case 1a break evaluation included evaluations of transport for both one train operation and two train operation. The logic trees below show transport fraction results of the portion of the evaluation that applies to debris transport to both sumps, considering two sump operation. Figure 3e-3 shows fine debris transport to alpha sump. Figure 3.e-4 shows large debris transport to alpha sump. Figure 3e-5 shows fine debris transport to bravo sump. Figure 3e-6 shows large debris transport to bravo sump. All four logic trees shown below are from the debris transport calculation. Refinements from NEI 04-07, Vol. 1 (GR) that impact the logic trees below are discussed in Section 3e.1.

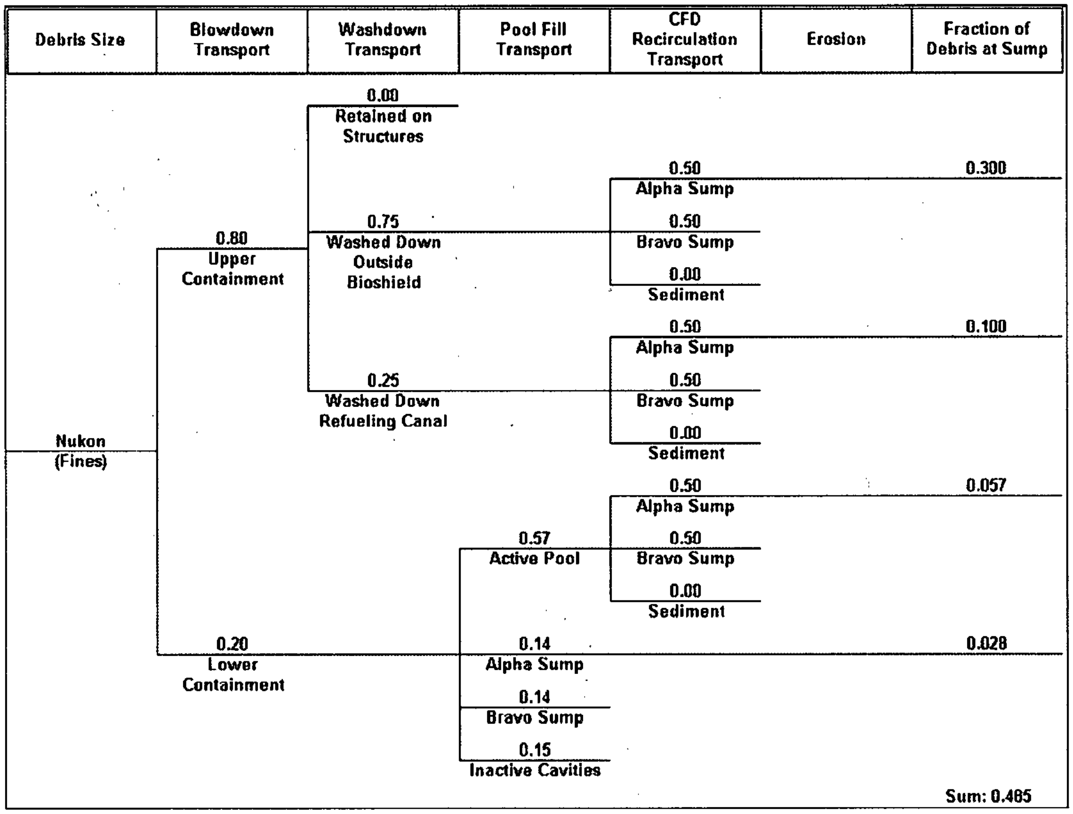


Figure 3e-3. Logic tree for NUKON™ fine debris to Alpha Sump

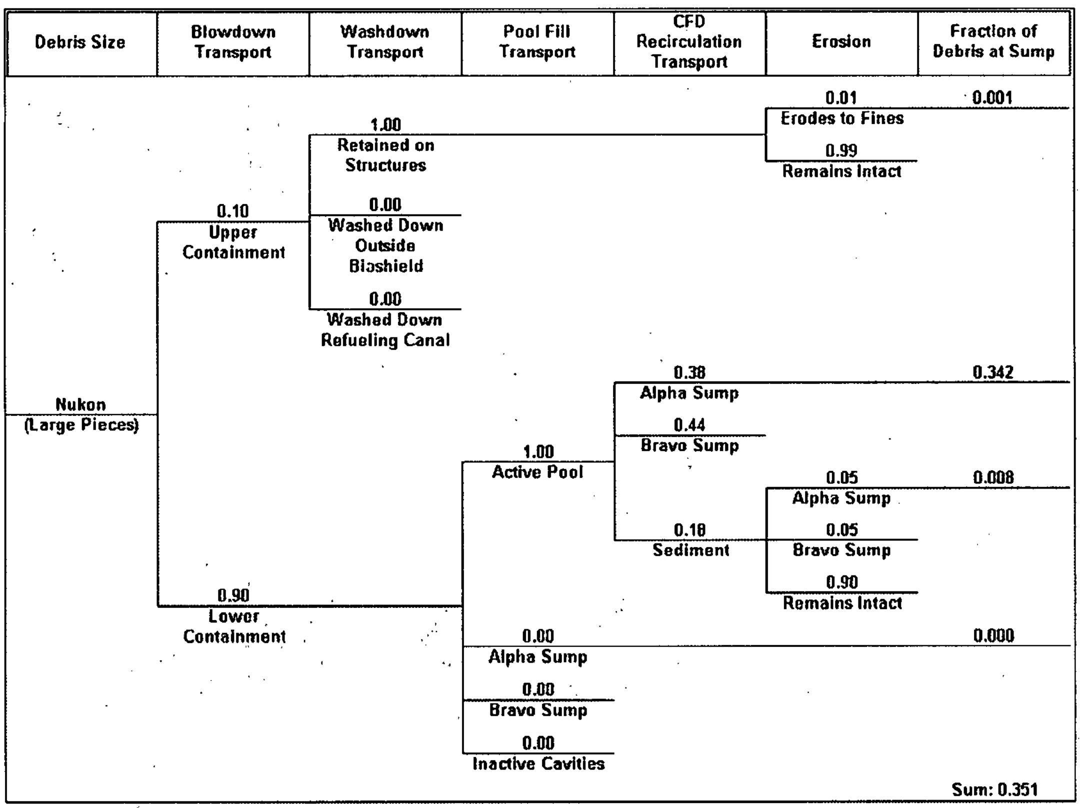


Figure 3e-4. Logic tree for NUKON™ large pieces debris to Alpha Sump

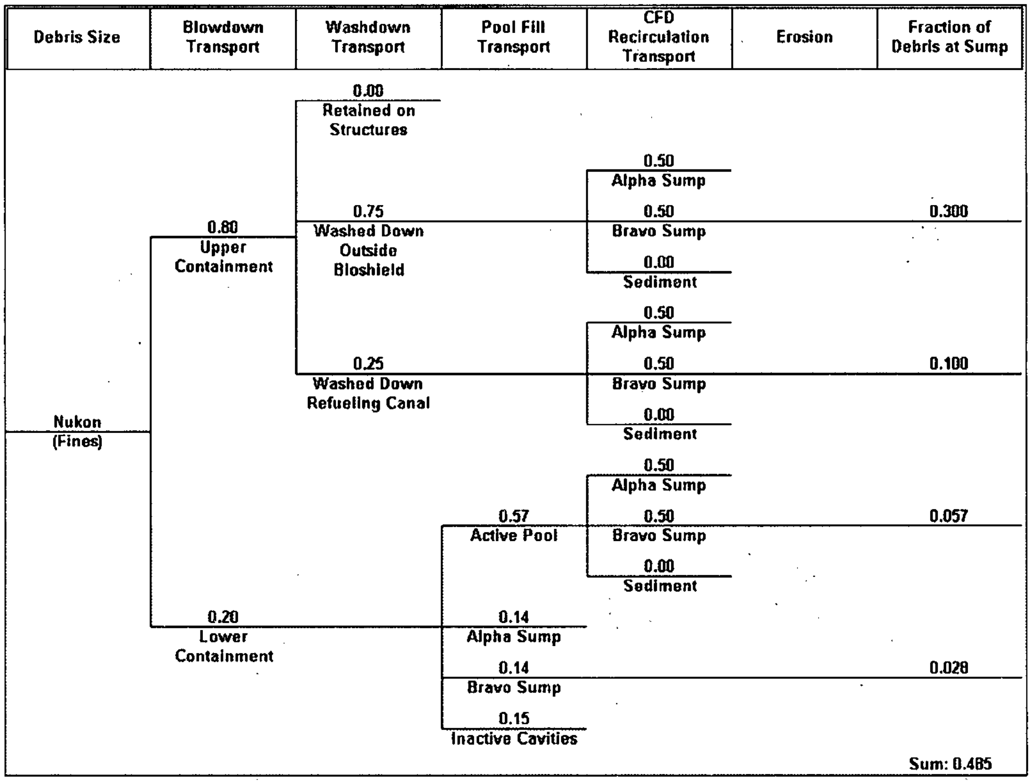


Figure 3e-5. Logic tree for NUKON™ fine debris to Bravo Sump

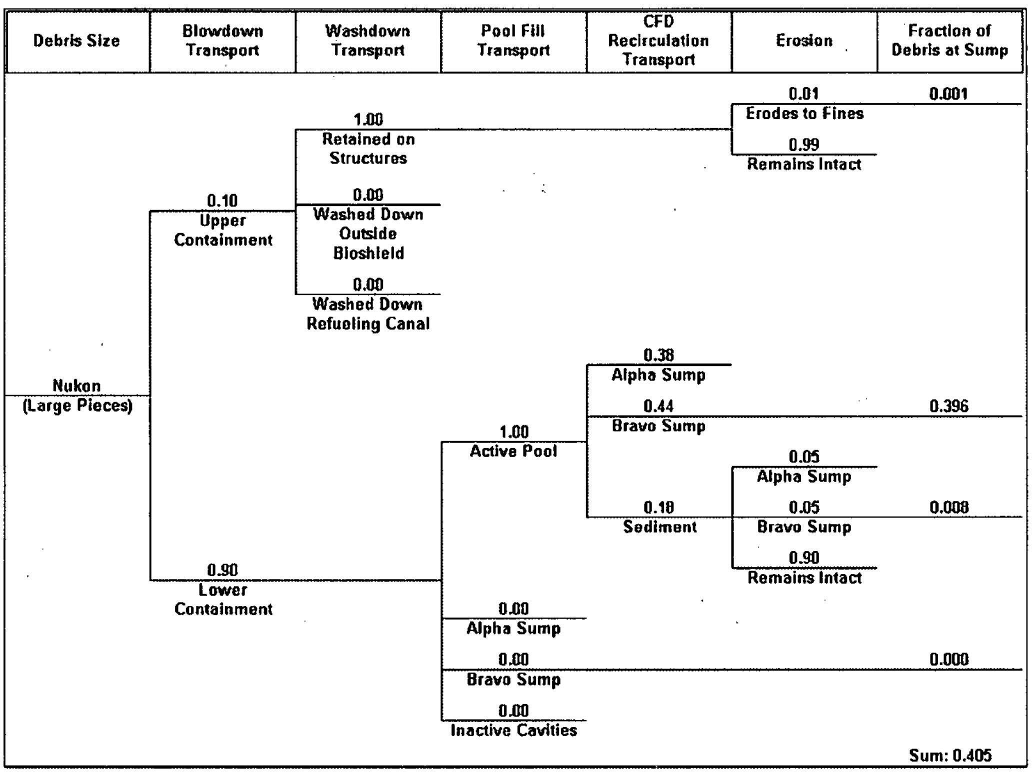


Figure 3e-6. Logic tree for NUKON™ large pieces debris to Bravo Sump

Tables 3e-1 through 3e-5 provide the total quantities of each type of debris transported to the sumps for the following evaluations:

1. Case 1a – Break at loop "D" cross-over leg at the base of the steam generator (two train operation)

2. Case 1a – Break at loop "D" cross-over leg at the base of the steam generator (one train operation)

3. Case 2 – Reactor vessel cold leg loop "A" nozzle break (two train operation)

4. Case 2 – Reactor vessel cold leg loop "A" nozzle break (one train operation)

5. Case 3 – Small break LOCA (two train operation)

Table 3e-1. Loop "D" Cross-over at SG with Two Train Operation

Material	Debris Generated	% Debris to Sumps A/B	Debris Transported to Sumps A/B
NUKON™ Small Fines*	210.9 ft^3	49%/49%	103.3 ft^3 / 103.3 ft^3
NUKON™ Large Pieces	140.6 ft^3	34%/40% (intact) 1%/1% (eroded to fines)	47.8 ft^3 / 56.2 ft^3 (intact) 1.4 ft^3/1.4 ft^3 (eroded to fines)
NUKON™ Intact Blankets	142.1 ft^3	0% / 0%	0.0 / 0.0
Latent Fiber	12.5 ft^3	43% / 43%	5.4 ft^3 / 5.4 ft^3
Thermo-Lag	25.3 lb	49% / 49%	12.4 lb / 12.4 lb
Latent Particulate	170 lb	43%/ 43%	73.1 lb / 73.1 lb
OEM Unqualified IOZ	15 lb	50% / 50%	7.5 lb / 7.5 lb
OEM Unqualified Alkyd	53 lb	50% / 50%	26.5 lb / 26.5 lb
OEM Unqualified Epoxy	18 lb	50% / 50%	9 lb / 9 lb
OEM Unqualified Margin	14 lb	50% / 50%	7 lb / 7 lb
Unqualified and Degraded IOZ (From degraded chips)	31.9 lb	50% / 50%	16 lb / 16 lb
Degraded Particulate Chips	2.24 lb	50% / 50%	1.12 lb / 1.12 lb
Degraded Fine Chips	6.72 lb	36% / 5%	2.42 lb / 0.34 lb
Degraded Small Chips	2.26 lb	1% / 3%	0.02 lb / 0.07 lb
Degraded Large Chips	2.49 lb	1% / 2%	0.02 lb / 0.05 lb
Degraded Curled Chips	4.37 lb	42% / 54%	1.84 lb / 2.36 lb
Carbozinc 11 IOZ (Inside ZOI)	78.9 lb	49% / 49%	38.7 lb / 38.7 lb
Carboline 191 HB Epoxy (Inside ZOI)	25.4 lb	49% / 49%	12.4 lb / 12.4 lb
Carboline 195 Epoxy (Inside ZOI)	21.4 lb	49% / 49%	10.5 lb / 10.5 lb
Amercoat 66 Epoxy (Inside ZOI)	0		
Carboline 4674 Acrylic (Inside ZOI)	10.3 lb	49% / 49%	5 lb / 5 lb
RMI Small Pieces	0 ft^2	34% / 35%	0 / 0
RMI Large Pieces	0 ft^2	36% / 44%	0 / 0
Misc. Debris	330.2	50% / 50%	165.1ft^2 / 165.1ft^2

* The "small fines" size classification contains approximately 30% "fines". See Section 3c.1.

Table 3e-2. Loop "D" Cross-over at SG with One Train Operation

Material	Debris Generated	% Debris to Sump A or B	Debris Transported to Sump A or B
NUKON™ Small Fines*	210.9 ft^3	94%/94%	198.2 ft^3 / 198.2 ft^3
NUKON™ Large Pieces	140.6 ft^3	2%/6% (intact) 9%/9% (eroded to fines)	2.8 ft^3 / 8.4 ft^3 (intact) 12.7ft^3/12.7ft^3 (eroded to fines)
NUKON™ Intact Blankets	142.1 ft^3	0% / 0%	0.0 / 0.0
Latent Fiber	12.5 ft^3	71% / 71%	8.9 ft^3 / 8.9 ft^3
Thermo-Lag	25.3 lb	94% / 94%	23.8 lb / 23.8 lb
Latent Particulate	170 lb	71%/ 71%	120.7 lb / 120.7 lb
OEM Unqualified IOZ	15 lb	100% / 100%	15.0 lb / 15.0 lb
OEM Unqualified Alkyd	53 lb	100% / 100%	53.0 lb / 53.0 lb
OEM Unqualified Epoxy	18 lb	100% / 100%	18.0 lb / 18.0 lb
OEM Unqualified Margin	14 lb	100% / 100%	14.0 lb / 14.0 lb
Unqualified and Degraded IOZ (From degraded chips)	31.9 lb	100% / 100%	31.9 lb / 31.9 lb
Degraded Particulate Chips	2.24 lb	100% / 100%	2.24 lb / 2.24 lb
Degraded Fine Chips	6.72 lb	7% / 7%	0.47 lb / 0.47 lb
Degraded Small Chips	2.26 lb	2% / 0%	0.05 lb / 0.00 lb
Degraded Large Chips	2.49 lb	1% / 0%	0.02 lb / 0.00 lb
Degraded Curled Chips	4.37 lb	85% / 88%	3.71 lb / 3.85 lb
Carbozinc 11 IOZ (Inside ZOI)	78.9 lb	94% / 94%	74.2 lb / 74.2 lb
Carboline 191 HB Epoxy (Inside ZOI)	25.4 lb	94% / 94%	23.9 lb / 23.9 lb
Carboline 195 Epoxy (Inside ZOI)	21.4 lb	94% / 94%	20.1 lb / 20.1 lb
Amercoat 66 Epoxy (Inside ZOI)	0		
Carboline 4674 Acrylic (Inside ZOI)	10.3 lb	94% / 94%	9.7 lb / 9.7 lb
RMI Small Pieces	0 ft^2	28% / 22%	0 / 0
RMI Large Pieces	0 ft^2	29% / 24%	0 / 0
Misc. Debris	330.2	100% / 100%	330.2ft^2 / 330.2ft^2

* The "small fines" size classification contains approximately 30% "fines". See Section 3c.1.

Table 3e-3. Reactor Cavity Nozzle Break with Two Train Operation

Material	Debris Generated	% Debris to Sumps A/B	Debris Transported to Sumps A/B
NUKON™ Small Fines*	21.9 ft^3	50%/50%	11.0 ft^3 / 11.0 ft^3
NUKON™ Large Pieces	14.6 ft^3	34%/40% (intact) 1%/1% (eroded to fines)	5.0 ft^3 / 5.8 ft^3 (intact) 0.1 ft^3 / 0.1 ft^3 (eroded to fines)
NUKON™ Intact Blankets	0.0 ft^3	0% / 0%	0.0 / 0.0
Cerablanket	14.4 ft^3	50% / 50%	7.2 ft^3 / 7.2 ft^3
Latent Fiber	12.5 ft^3	50% / 50%	6.3 ft^3 / 6.3 ft^3
Thermo-Lag	25.3 lb	50% / 50%	12.4 lb / 12.4 lb
Foamglas	5.6 ft^3	50% / 50%	2.8 ft^3 / 2.8 ft^3
Latent Particulate	170 lb	50%/ 50%	85.0 lb / 85.0 lb
OEM Unqualified IOZ	15 lb	50% / 50%	7.5 lb / 7.5 lb
OEM Unqualified Alkyd	53 lb	50% / 50%	26.5 lb / 26.5 lb
OEM Unqualified Epoxy	18 lb	50% / 50%	9 lb / 9 lb
OEM Unqualified Margin	14 lb	50% / 50%	7 lb / 7 lb
Unqualified and Degraded IOZ (From degraded chips)	31.9 lb	50% / 50%	16 lb / 16 lb
Degraded Particulate Chips	2.24 lb	50% / 50%	1.12 lb / 1.12 lb
Degraded Fine Chips	6.72 lb	36% / 5%	2.42 lb / 0.34 lb
Degraded Small Chips	2.26 lb	1% / 3%	0.02 lb / 0.07 lb
Degraded Large Chips	2.49 lb	1% / 2%	0.02 lb / 0.05 lb
Degraded Curled Chips	4.37 lb	42% / 54%	1.84 lb / 2.36 lb
Carbozinc 11 IOZ (Inside ZOI)	52.4 lb	50% / 50%	26.2 lb / 26.2 lb
Carboline 890 (Inside ZOI)	55.2 lb	50% / 50%	27.6 lb / 27.6 lb
Carboline 4674 Acrylic (Inside ZOI)	37.0 lb	50% / 50%	18.5 lb / 18.5 lb
RMI Small Pieces	4,861 ft^2	0% / 0%	0 / 0
RMI Large Pieces	1,620 ft^2	0% / 0%	0 / 0
Misc. Debris	330.2	50% / 50%	165.1 ft^2 / 165.1 ft^2

* The "small fines" size classification contains approximately 30% "fines". See Section 3c.1.

Table 3.e-4 Reactor Cavity Nozzle Break with One Train Operation

Material	Debris Generated	% Debris to Sumps A/B	Debris Transported to Sumps A/B
NUKON™ Small Fines*	21.9 ft^3	100%/100%	21.9 ft^3 / 21.9 ft^3
NUKON™ Large Pieces	14.6 ft^3	2%/6% (intact) 9%/9% (eroded to fines)	0.3 ft^3 / 0.9 ft^3 (intact) 1.3 ft^3 / 1.3 ft^3 (eroded to fines)
NUKON™ Intact Blankets	0.0 ft^3	0% / 0%	0.0 / 0.0
Cerablanket	14.4 ft^3	100% / 100%	14.4 ft^3 / 14.4 ft^3
Latent Fiber	12.5 ft^3	100% / 100%	12.5 ft^3 / 12.5 ft^3
Thermo-Lag	25.3 lb	100% / 100%	25.3 lb / 25.3 lb
Foamglas	5.6 ft^3	100% / 100%	5.6 ft^3 / 5.6 ft^3
Latent Particulate	170 lb	100% / 100%	170 lb / 170 lb
OEM Unqualified IOZ	15 lb	100% / 100%	15 lb / 15 lb
OEM Unqualified Alkyd	53 lb	100% / 100%	53.0 lb / 53.0 lb
OEM Unqualified Epoxy	18 lb	100% / 100%	18.0 lb / 18.0 lb
OEM Unqualified Margin	14 lb	100% / 100%	14 lb / 14 lb
Unqualified and Degraded IOZ (From degraded chips)	31.9 lb	100% / 100%	31.9 lb / 31.9 lb
Degraded Particulate Chips	2.24 lb	100% / 100%	2.24 lb / 2.24 lb
Degraded Fine Chips	6.72 lb	7% / 7%	0.47 lb / 0.47 lb
Degraded Small Chips	2.26 lb	2% / 0%	0.05 lb / 0.0 lb
Degraded Large Chips	2.49 lb	0% / 0%	0.0 lb / 0.0 lb
Degraded Curled Chips	4.37 lb	85% / 88%	3.71 lb / 3.85 lb
Carbozinc 11 IOZ (Inside ZOI)	52.4 lb	100% / 100%	52.4 lb / 52.4 lb
Carboline 890 (Inside ZOI)	55.2 lb	100% / 100%	55.2 lb / 55.2 lb
Carboline 4674 Acrylic (Inside ZOI)	37.0 lb	100% / 100%	37.0 lb / 37.0 lb
RMI Small Pieces	4,861 ft^2	0% / 0%	0 / 0
RMI Large Pieces	1,620 ft^2	0% / 0%	0 / 0
Misc. Debris	330.2	100% / 100%	330.2 ft^2 / 330.2 ft^2

* The "small fines" size classification contains approximately 30% "fines". See Section 3c.1.

Table 3e-5. SBLOCA: Loop D Alternate Charging line (Two Train Operation)

Material	Debris Generated	% Debris to Sumps A/B	Debris Transported to Sumps A/B
NUKON™ Small Fines*	1.1 ft^3	46%/46%	0.5 ft^3 / 0.5 ft^3
NUKON™ Large Pieces	0.7 ft^3	0%/0% (intact) 5%/5% (eroded to fines)	0 ft^3 / 0 ft^3 (intact) 0 ft^3/ 0 ft^3 (eroded to fines)**
NUKON™ Intact Blankets	2.3 ft^3	0% / 0%	0.0 / 0.0
Latent Fiber	12.5 ft^3	46% / 46%	5.8 ft^3 / 5.8 ft^3
Latent Particulate	170 lb	46%/ 46%	78.2 lb / 78.2 lb
OEM Unqualified IOZ	15 lb	50% / 50%	7.5 lb / 7.5 lb
OEM Unqualified Alkyd	53 lb	50% / 50%	26.5 lb / 26.5 lb
OEM Unqualified Epoxy	18 lb	50% / 50%	9.0 lb / 9.0 lb
OEM Unqualified Margin	14 lb	50% / 50%	7.0 lb / 7.0 lb
Unqualified and Degraded IOZ (From degraded chips)	31.9 lb	50% / 50%	16.0 lb / 16.0 lb
Degraded Particulate Chips	2.24 lb	50% / 50%	1.12 lb / 1.12 lb
Degraded Fine Chips	6.72 lb	0% / 0%	0 lb / 0 lb
Degraded Small Chips	2.26 lb	0% / 0%	0 lb / 0 lb
Degraded Large Chips	2.49 lb	0% / 0%	0 lb / 0 lb
Degraded Curled Chips	4.37 lb	27% / 19%	1.18 lb / 0.83 lb
Carbozinc 11 IOZ (Inside ZOI)	0 lb	0% / 0%	0 lb / 0 lb
Carboline 191 HB Epoxy (Inside ZOI)	0 lb	0% / 0%	0 lb / 0 lb
Carboline 195 Epoxy (Inside ZOI)	0 lb	0% / 0%	0 lb / 0 lb
Amercoat 66 Epoxy (Inside ZOI)	0 lb	0% / 0%	0 lb / 0 lb
Carboline 4674 Acrylic (Inside ZOI)	0 lb	0% / 0%	0 lb / 0 lb
RMI Small Pieces	0 ft^2	0% / 0%	0 / 0
RMI Large Pieces	0 ft^2	0% / 0%	0 / 0
Misc. Debris	330.2	50% / 50%	165.1ft^2 / 165.1ft^2

* The "small fines" size classification contains approximately 30% "fines". See Section 3c.1.

** Even though there is a small percentage of eroded fines transported to the sumps, the small volume (0.035 ft^3) is neglected in the analysis.

3f. Head Loss and Vortexing

The objectives of the head loss and vortexing evaluations are to calculate head loss across the sump strainer and to evaluate the susceptibility of the strainer to vortex formation.

1. *Provide a schematic diagram of the emergency core cooling system (ECCS) and containment spray systems (CSS).*
2. *Provide the minimum submergence of the strainer under small-break loss-of-coolant accident (SBLOCA) and large-break loss-of-coolant accident (LBLOCA) conditions.*
3. *Provide a summary of the methodology, assumptions and results of the vortexing evaluation. Provide bases for key assumptions.*
4. *Provide a summary of the methodology, assumptions, and results of prototypical head loss testing for the strainer, including chemical effects. Provide bases for key assumptions.*
5. *Address the ability of the design to accommodate the maximum volume of debris that is predicted to arrive at the screen.*
6. *Address the ability of the screen to resist the formation of a thin bed or to accommodate partial thin bed formation.*
7. *Provide the basis for the strainer design maximum head loss.*
8. *Describe significant margins and conservatisms used in the head loss and vortexing calculations.*
9. *Provide a summary of the methodology, assumptions, bases for the assumptions, and results for the clean strainer head loss calculation.*
10. *Provide a summary of the methodology, assumptions, bases for the assumptions, and results for the debris head loss analysis.*
11. *State whether the sump is partially submerged or vented (i.e., lacks a complete water seal over its entire surface) for any accident scenarios and describe what failure criteria in addition to loss of net positive suction head (NPSH) margin were applied to address potential inability to pass the required flow through the strainer.*
12. *State whether near-field settling was credited for the head-loss testing and, if so, provide a description of the scaling analysis used to justify near-field credit.*
13. *State whether temperature/viscosity was used to scale the results of the head loss tests to actual plant conditions. If scaling was used, provide the basis for concluding that boreholes or other differential-pressure induced effects did not affect the morphology of the test debris bed.*
14. *State whether containment accident pressure was credited in evaluating whether flashing would occur across the strainer surface, and if so, summarize the methodology used to determine the available containment pressure.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Sections 2(c).7, 2(d)(ii) and 2(d)(iii). (Reference 2)

3f.1 Schematic Diagrams

Schematic diagrams of the emergency core cooling system (ECCS) and the containment spray system (CSS) are provided in Figures 3f-1 and 3f-2. The highlighted flow paths in Figure 3.f-1 depicted possible flow paths associated with containment recirculation sump ECCS operation, but do not depict specific operating conditions. For example, hot leg (HL) and cold leg (CL) recirculation are not aligned at the same time.

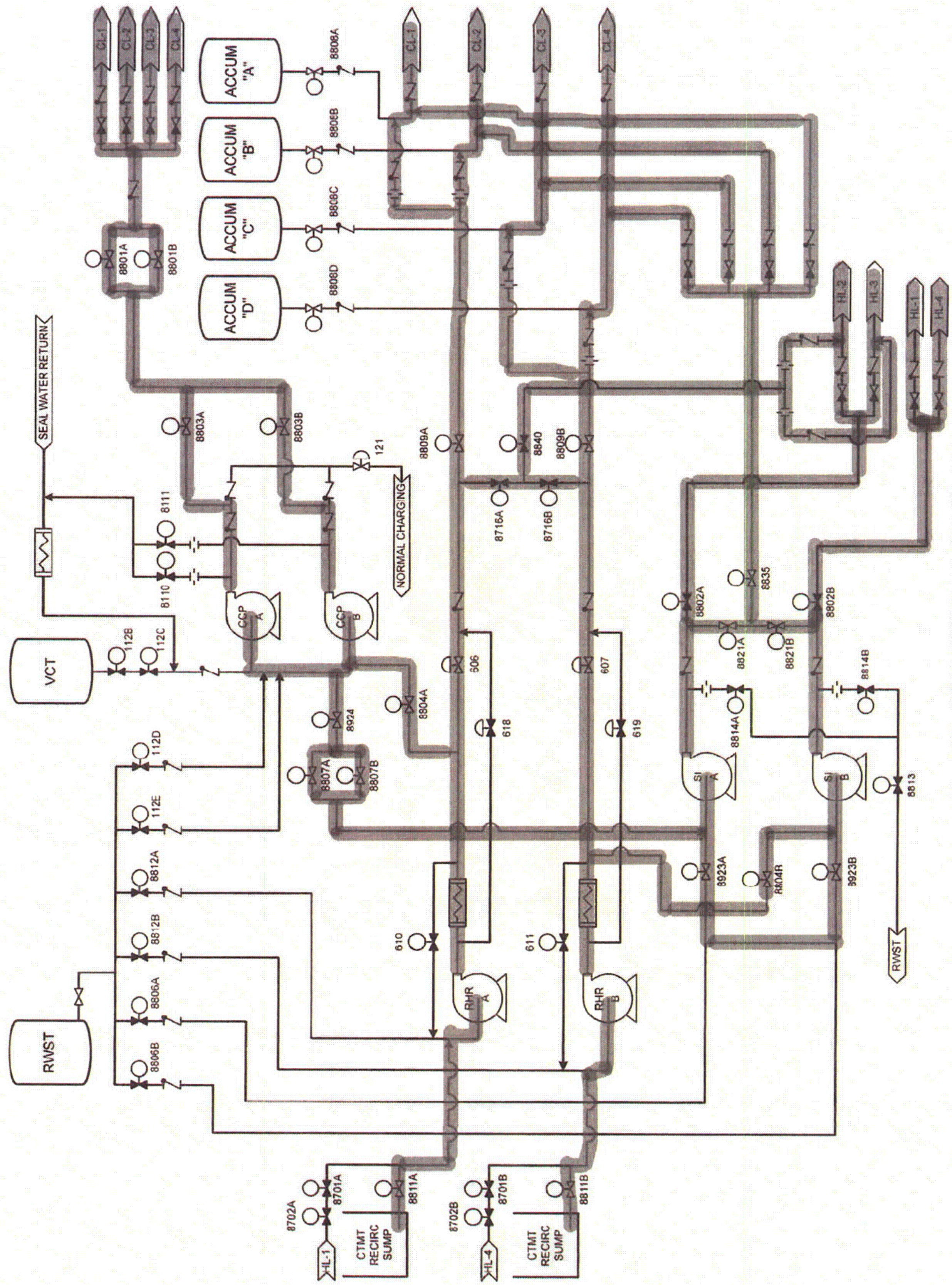


Figure 3f-1. WCGS ECCS Process Flow Diagram

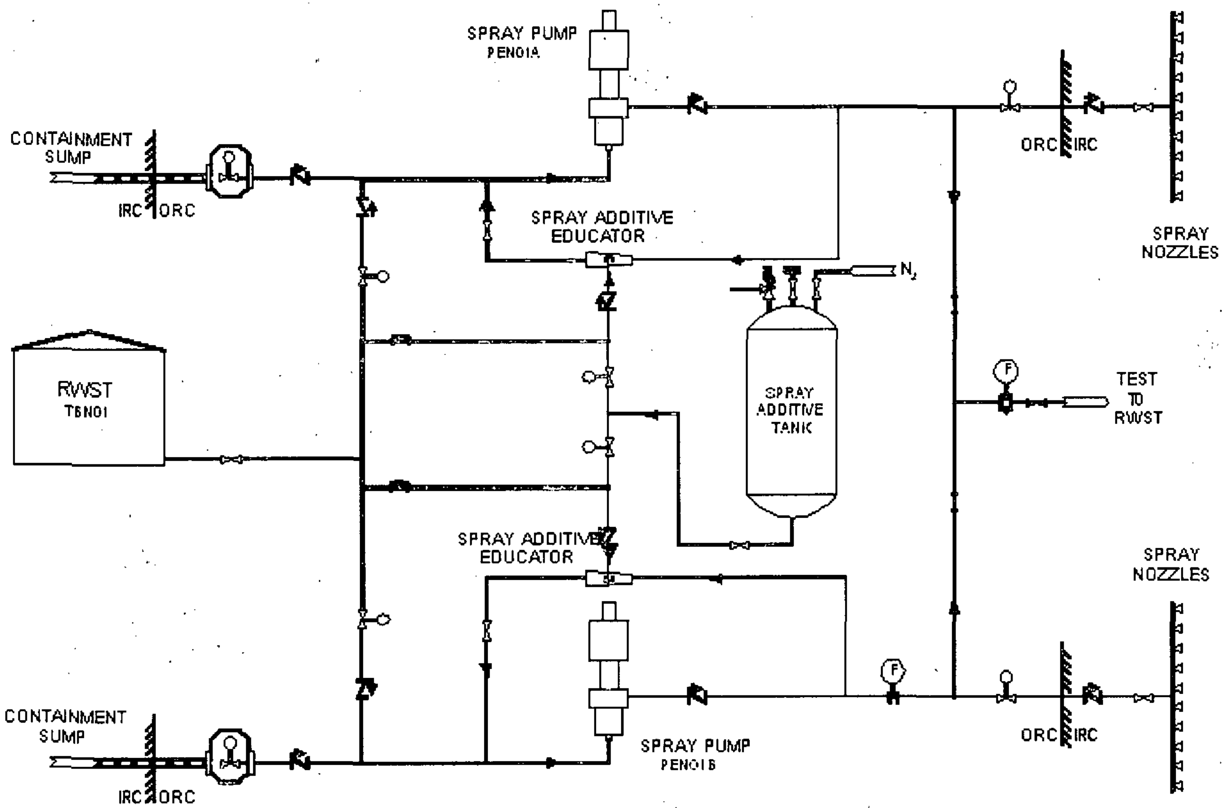


Figure 3f-2. WCGS CSS Process Flow Diagram

3f.2 Minimum Strainer Submergence

For the large break LOCA condition, the containment recirculation sump strainers will be completely submerged, with greater than 8" of water above the top of the strainers, at the time of ECCS switchover to recirculation. For the small break LOCA condition, the replacement sump strainer will be partially submerged, with water level less than approximately 6" below the top of the strainer, at the time of ECCS switchover to recirculation. A section of the minimum containment water level calculation specifically addresses a small break LOCA condition when the safety injection accumulators do not discharge and the containment spray system does not activate.

3f.3 Vortexing Evaluation

A calculation will be performed for vortexing, air ingestion, and void fraction after the final test report from the January 2008 strainer testing has been completed. As described in Section 2, WCNOC will update this response following completion of the actions associated with the vendor testing.

In addition to the vendor performed calculation for vortexing, air ingestion, and void fraction, the following items are considered additional means by which vortexing is precluded at the strainers:

Vortex breakers are installed at the piping entrances into the ECCS/CSS system piping, as shown in Figure 3f-3. The purpose of the vortex breakers is to prevent vortexing at the pipe entrances. This design feature has been in place since initial operation.

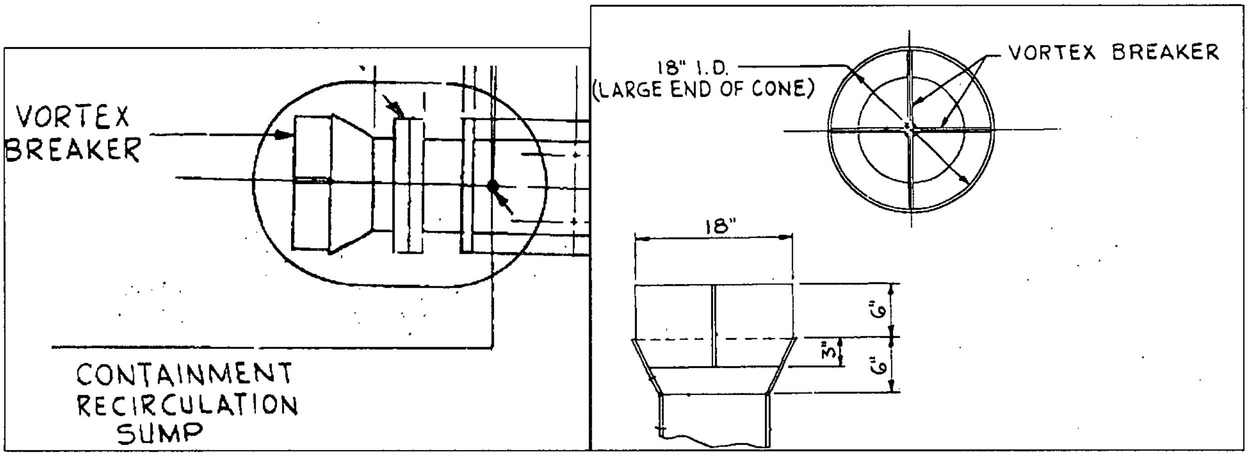


Figure 3f-3. Vortex Breakers

3f.4, 3f.5, 3f.6, 3f.7, 3f.8, 3f.9, 3f.10, and 3f.11

As described in Section 2, WCNOC will update these responses following completion of the actions associated with the vendor testing.

3f.12 Credit for Near Field Settling

The strainer head loss testing did credit near-field settling. The test flume was constructed with the flume walls arranged such that the velocity fields around the testing strainer were representative or bounding of the expected velocity fields in the containment during LOCA recirculation. Two computational fluid dynamics (CFD) analyses were used to accurately model the debris transport and to model the flow/turbulence of the flow patterns approaching the strainer.

A refined CFD analysis near sump screen structure was performed to model the flow patterns approaching the sump to define approach flow velocities for test flume. The CFD analysis completed for the transport calculation modeled the sump as a mass sink, which was not based on the actual strainer design configuration.

The refined analysis was required because the debris transport CFD simulation utilized a relatively coarse mesh in the vicinity of the sump pits. The boundary condition at the sump pit in the debris transport CFD model was representative of the plant configuration prior to installation of the Performance Contracting, Inc. (PCI) Sure-Flow® strainer arrays. The flow patterns approaching the perimeter of the pit were strongly influenced by the outflow boundary condition used in the debris transport CFD analyses, and the uniform flow characteristics of the PCI Sure-Flow® strainer modules influence the approach flow patterns to the sump.

Inflow boundary conditions for the new near field approach CFD model were taken directly from the results database of the debris transport CFD model (flow and velocity). Water surface elevation was assumed constant.

In the debris transport CFD, water falling near the "A" sump was assumed to fall as individual large raindrops with a terminal velocity of approximately 29 ft/s. As this results in only a disturbance at the water surface with minimal energy penetrating the water column, the equivalent flow falling from above was introduced at the new near field approach CFD model inflow boundaries to conservatively increase the transport velocities leading to the sumps.

Flow patterns from the refined new near field approach CFD model reflect the presence of the PCI Sure-Flow® Strainer Array. Flow patterns approaching the sump pit from the containment periphery were similar in the two CFD calculations.

The new near field approach CFD flow patterns and velocities are more representative of the post LOCA containment flow patterns because the boundary conditions associated with the PCI Sure-Flow® Strainer have been included in the analysis, and the computational grid has been greatly refined in the vicinity of the sumps resulting in a more realistic prediction of approach flow patterns near the sump pit and screens.

3f.13 Head Loss Test Scaling

As described in Section 2, WCNOC will update this response following completion of the actions associated with the vendor testing.

3f.14 Crediting Containment Accident Pressure

Conservatively, no credit was taken for any over-pressurization (i.e., greater than atmospheric pressure) of the Containment during the LOCA or post-LOCA period with regard to head-loss, vortex, air ingestion or void fraction determination.

3g. Net Positive Suction Head (NPSH)

The objective of the NPSH section is to calculate the NPSH margin for the ECCS and CSS pumps that would exist during a loss-of-coolant accident (LOCA) considering a spectrum of break sizes.

1. *Provide applicable pump flow rates, the total recirculation sump flow rate, sump temperature(s), and minimum containment water level.*
2. *Describe the assumptions used in the calculations for the above parameters and the sources/bases of the assumptions.*
3. *Provide the basis for the required NPSH values, e.g., three percent head drop or other criterion.*
4. *Describe how friction and other flow losses are accounted for.*
5. *Describe the system response scenarios for LBLOCA and SBLOCAs.*
6. *Describe the operational status for each ECCS and CSS pump before and after the initiation of recirculation.*
7. *Describe the single failure assumptions relevant to pump operation and sump performance.*
8. *Describe how the containment sump water level is determined.*
9. *Provide assumptions that are included in the analysis to ensure a minimum (conservative) water level is used in determining NPSH margin.*
10. *Describe whether and how the following volumes have been accounted for in pool level calculations: empty spray pipe, water droplets, condensation and holdup on horizontal and vertical surfaces. If any are not accounted for, explain why.*

11. Provide assumptions (and their bases) as to what equipment will displace water resulting in higher pool level. *12. Provide assumptions (and their bases) as to what water sources provide pool volume and how much volume is from each source.* *13. If credit is taken for containment accident pressure in determining available NPSH, provide description of the calculation of containment accident pressure used in determining the available NPSH.* *14. Provide assumptions made which minimize the containment accident pressure and maximize the sump water temperature.* *15. Specify whether the containment accident pressure is set at the vapor pressure corresponding to the sump liquid temperature.* *16. Provide the NPSH margin results for pumps taking suction from the sump in recirculation mode.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(d)(i) (Reference 2). The NPSH margin evaluation process described below conforms to Sections 3.7 and 4.2.5 of NEI 04-07, Vol. 2 (SE) (Reference 1).

3g.1 Pump Flow Rates, Recirculation Sump Flow Rate, Sump Temperature, and Minimum Containment Water Level

For two train operation, the maximum flow rates are conservatively assumed for head loss testing purposes to be:

- Residual Heat Removal (RHR) Pump flow rate on recirculation is 4800 gpm for each pump
- Containment Spray (CS) Pump flow rate on recirculation is 3950 gpm
- Total flow rate through a strainer is 8750 gpm
- Total sump flow rate is 17,500 gpm

For single train operation, the maximum flow rates are conservatively assumed for head loss testing purposes to be:

- Residual Heat Removal (RHR) Pump flow rate on recirculation is 4880 gpm
- Containment Spray (CS) Pump flow rate on recirculation is 3950 gpm
- Total flow rate through a strainer is 8830 gpm considering RHR and CS flows.

NOTE: Single train RHR flow rate is higher than two train operation since one RHR pump supplies the RCS as well as both safety injection and centrifugal charging pumps during single train operation.

Large break LOCA minimum water level at ECCS switchover is elevation 2002' 1.02"

Large break LOCA minimum water level at CS switchover is elevation 2002' 5.18"

Small break LOCA minimum water level elevation 2000' 8.89"

A sump temperature of 212°F was used for the NPSH calculation.

ECCS Switchover is expected to occur between 17 and 27 minutes.

3g.2 Assumptions and Assumptions Sources/Bases

The basis for the flow rates used in the NPSH analysis were chosen to bound preoperational testing flow rates.

3g.3 Basis for Required NPSH Values

The NPSH required values are based on the pump curves provided by the pump vendor.

3g.4 Friction and other Flow Losses

The friction and flow loss values are based on standard industry accepted estimates of piping friction and fitting head loss.

3g.5 System Response Scenarios for LBLOCAs and SBLOCAs

For LOCAs, there are two modes of operation: the injection mode and the recirculation mode of operation.

During a large break LOCA, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. Once RCS pressure is less than approximately 600 psig, the four accumulator tanks will inject into the RCS. A safety injection signal (SIS) is generated when the low pressurizer pressure SI setpoint is reached. A containment spray actuation signal (CSAS) is generated when the containment pressure setpoint is reached. Upon receipt of an SIS and CSAS, the ECCS and CSS are activated, commencing the injection mode of ECCS and CSS operation. This mode of operation consists of both trains of the ECCS pumps running (2 charging pumps, 2 safety injection pumps, and 2 residual heat removal pumps) and both containment spray pumps taking suction from the RWST and delivering water to the reactor coolant system (RCS).

Continued operation of the ECCS pumps supplies water during long-term cooling. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation mode of operation in which spilled borated water is drawn from the containment recirculation sumps and returned to the RCS cold legs. The containment spray pumps are manually aligned to the containment recirculation sumps and continue to operate to further reduce containment pressure.

Approximately 10 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs to control the boric acid concentration in the reactor vessel.

For a small break LOCA, information provided by Westinghouse indicates that for "a typical Westinghouse 4-loop PWR with a larger dry containment, such as WCGS or Callaway, an equivalent 3-inch diameter break or smaller may result in the RCS pressure equilibrating at 1000 psi to about 1200 psi. At this pressure, the safety injection accumulators will not discharge. For breaks of greater than about an equivalent 3-inch diameter, the plant would undergo a sufficiently rapid depressurization that the safety injection accumulators would discharge".

During a small break LOCA scenario, a safety injection signal will start both trains of charging, safety injection (SI) and residual heat removal (RHR) pumps in the injection mode from the RWST to the RCS (cold leg injection). The charging pumps will inject water immediately. If RCS pressure continues to decrease below the shut-off head of the SI pumps (approximately 1550 psig), they will also start injecting into the RCS. As the control room operators progress through emergency procedures, they may shut-off the RHR pumps based on RCS pressure (stable or increasing). The SI accumulators are also aligned to inject into the RCS when the RCS pressure drop below the accumulator pressure. The RHR pumps will not start injecting until RCS pressure drops to below the shutoff head of the RHR pumps (approximately 325 psig). If the combination of the charging pumps and SI pumps does not equal the break flow, RCS pressure will continue to decrease. If RCS pressure stabilizes somewhere above the shut-off head of the RHR pumps, they may be turned off. Containment Spray is not expected to actuate during a small break LOCA.

The objective of the control room operators is to cool down and depressurize the RCS, so that the RHR pumps may be aligned from the RCS hot legs and recirculated back to the cold legs. But, if the RWST is pumped down to the ECCS switchover level, the RHR pumps will be aligned to take a suction from the containment emergency recirculation sumps and supply suction to the SI and charging pumps. If RCS pressure is below the shut off head of the RHR pumps, the RHR pumps will also inject into the RCS. The recirculating water will be cooled as it is pumped through the RHR heat exchangers.

3g.6 Operational Status of ECCS and CSS Pumps

Prior to the recirculation phase of the analyzed postulated accident, both safety trains of ECCS pumps are running, which includes two charging pumps, two safety injection pumps, and two residual heat removal pumps. In addition, both containment spray pumps are running. Following initiation of the recirculation phase, the status remains as described above.

3g.7 Single Failure Assumptions

A review of single failure assumptions associated with the analyses of design basis accident scenarios described in this response determined that there are no single failure assumptions that caused the results of the NPSH analyses to be non-conservative.

3g.8 Determining Containment Sump Water Level

The quantity of water added to containment from the Refueling Water Storage Tank (RWST) was calculated for each of the breaks. The quantity of water in the containment building that would not contribute to the water level is also calculated. Water volume not contributing to the containment building water level includes:

1. Steam holdup in the Containment atmosphere,
2. Water volume required to fill the RHR and Containment Spray piping that is empty prior to the LOCA,

3. Additional mass of water that must be added to the RCS due to the increase in the water density at the lower sump water temperature (versus the RCS temperature prior to the LOCA),
4. Water film on surfaces,
5. Water volume required to fill the RCS steam space,
6. Water in transit from the Containment Spray nozzles and the break to the Containment Sump, and
7. Holdup volumes throughout Containment, as described in section 3l.

Given the net mass of water added to the Containment floor based on the considerations described above, the post-LOCA containment building water level was calculated using a correlation between available water volume and containment building space available above the floor level elevation.

3g.9 Assumptions to Ensure a Conservative Water Level is Used

The RWST, RCS and the safety injection accumulators inventories were assumed to be the same density as pure water. This assumption is reasonable since the boric acid concentration is small (i.e., less than or equal to 2500 ppm).

The total RHR and Containment Spray piping hold-up volume calculated was increased by a conservative 5% to account for additional volume that was not considered (such as higher cross-sectional area for valves, other fittings, and drain lines).

3g.10 Accounting for Volumes in Pool Level Calculations

The following volumes have been accounted for in the pool level calculation:

1. Steam holdup in the Containment atmosphere,
2. Water volume required to fill the RHR and Containment Spray piping that is empty prior to the LOCA,
3. Additional mass of water that must be added to the RCS due to the increase in the water density at the lower sump water temperature (versus the RCS temperature prior to the LOCA),
4. Water film on surfaces,
5. Water volume required to fill the RCS steam space,
6. Water in transit from the Containment Spray nozzles and the break to the Containment Sump, and
7. Holdup volumes throughout Containment, as described in section 3l.

Example values of volumes for empty spray pipe, condensation and holdup on horizontal and vertical surfaces are described below.

1. 260.6 ft^3 was allotted for the "A" train containment spray line and 259.0 ft^3 was allotted for the "B" train containment spray line.
2. 1592 ft^3 of water vapor is allotted for a large break LOCA at ECCS Switchover with minimum safeguards (i.e., one train of equipment assuming to be operating at minimum acceptable operation).

3. 1164 ft^3 of water volume is allotted for wetted surfaces during a Large Break LOCA at ECCS Switchover with minimum safeguards.

3g.11 Assumptions/ Bases for Water Displacement by Equipment

Equipment below 2000' elevation that displaces water: (Reactor cavity & Incore tunnel):

1. Incore tubes – 30.05 ft^3
2. Reactor vessel – 2078 ft^3
3. Incore tunnel beams – 21.9 ft^3

Equipment between elevation 2000'-0" and 2000'-6" that displaces water:

1. Concrete walls – 3071 ft^2
2. Floor @ 2001'-4" – 4710 ft^2
3. PRT supports 36.7 ft^2
4. RCDT supports – 5.55 ft^2
5. Recirculation sump curbs – 33.8 ft^2
6. Incore sump curbs – 18.9 ft^2
7. Area of Incore tunnel – 188.56 ft^2

3g.12 Assumptions/ Bases for Water Sources Providing Pool Volume

The initial RCS volume in minimized by assuming a minimum Pressurizer volume of 38%. This assumption is based on the nominal pressurizer span at 100% power and Tavg = 570.7°F.

To conservatively minimize the mass of water contained in the safety injection accumulators that could flow into the containment building, the temperature is assumed to be equal to the maximum initial containment air temperature of 120°F, consistent with the accident analysis. This approach is conservative because the density of water decreases with increasing temperature.

The RWST, RCS and the SI accumulator inventory was assumed to be the same density as pure water. This assumption is reasonable since the boric acid concentrations are small.

Mass input to sump from the containment water level calculation at ECCS switchover (or RHR swapover) accident scenario:

- RCS blowdown 549,054 lbm
- SI Accumulators 200,006 lbm
- Initial Containment vapor 732 lbm
- RWST Input 1,889,072 lbm

3g.13 Credit for Containment Accident Pressure

Conservatively, no credit was taken for any over-pressurization (i.e., greater than atmospheric pressure) of the Containment during the LOCA or post-LOCA period with regard to head-loss, vortex, air ingestion or void fraction determination.

3g.14 Assumptions that Minimize Containment Accident Pressure and Maximize Sump Water Temperature

Conservatively, no credit was taken for any over-pressurization (i.e., greater than atmospheric pressure) of the Containment during the LOCA or post-LOCA period with regard to head-loss, vortex, air ingestion or void fraction determination.

The pressure drop across the strainer in the worse case, will be the lowest temperature seen during the event time frame. Soon after an event occurs, the temperature inside the containment building reaches the highest temperature for the event. When RHR and CSS are operating the temperature slowly decreases. The conservative sump temperature will be used in the NPSH margin determination to be completed along with other future actions associated with the vendor testing, as described in Section 2.

3g.15 Containment Accident Pressure and Sump Liquid Temperature Vapor Pressure

Credit is not taken in the NPSH analysis for post-accident containment overpressure. The vapor pressure of the containment sump liquid vapor pressure is assumed equal to atmospheric pressure.

3g.16 ECCS and CCS NPSH Margin

As described in Section 2, WCNOC will update this response following completion of the actions associated with the vendor testing.

3h. Coatings Evaluation

The objective of the coatings evaluation section is to determine the plant-specific ZOI and debris characteristics for coatings for use in determining the eventual contribution of coatings to overall head loss at the sump screen.
1. *Provide a summary of type(s) of coating systems used in containment, e.g., Carboline CZ 11 Inorganic Zinc primer, Ameron 90 epoxy finish coat.*
2. *Describe and provide bases for assumptions made in post-LOCA paint debris transport analysis.*
3. *Discuss suction strainer head loss testing performed as it relates to both qualified and unqualified coatings and what surrogate material was used to simulate coatings debris.*
4. *Provide bases for the choice of surrogates.*
5. *Describe and provide bases for coatings debris generation assumptions. For example, describe how the quantity of paint debris was determined based on ZOI size for qualified and unqualified coatings.*
6. *Describe what debris characteristics were assumed, i.e., chips, particulate, size distribution and provide bases for the assumptions.*
7. *Describe any ongoing containment coating condition assessment program.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).6 (Reference 2). The coatings evaluations described below conforms to Sections 3.4.2.1 and 3.4.3 of NEI 04-07, Vol. 2 (SE) (Reference 1).

Initially, all coatings within the zone of influence (ZOI) of a loss of coolant accident (LOCA) break were assumed to fail and be transported to the sump strainers. In addition, all unqualified coatings inside Containment were also assumed to fail and be transported to the sump strainers. To reduce the amount of assumed debris, from failed coatings, reaching the sump strainers, a WCGS-specific calculation was performed to quantify the amount of unqualified coating inside the Containment Building that could reach the recirculation sump strainers. Pertaining to coatings, WCNOC is committed to Regulatory Guide 1.54-1973. R.G. 1.54 refers to ANSI N101.4 as being acceptable. Therefore, any surface coating on components in Containment was considered to be qualified if its documentation stated that the coating application met the requirements of R.G. 1.54 and/or ANSI N101.4. This calculation researched the coating specifications for various components located inside Containment to determine if they could be considered qualified. The total unqualified coating surface area was quantified and the amount that was prevented from reaching the sump strainers (i.e. insulated) was deducted. The total amount of coatings surface area that could fail and reach the sump strainers was determined to be 5000 ft^2. This number includes margin and 1000 ft^2 for qualified but degraded coating. The weight of the debris, including margin, from failed coatings caused by a LOCA environment was determined to be 250 pounds.

In lieu of the 10D destruction ZOI stipulated by NEI 04-07, Vol. 2 (SE), a 4D ZOI for coatings has been used for all qualified coatings systems (with the exception of untopcoated inorganic zinc systems) based on Wyle Labs testing (Reference 8). This same reference specifies the use of a 5D ZOI for untopcoated inorganic zinc primers.

3h.1 Types of Coatings used in Containment

Design Basis Accident (DBA) Qualified/Acceptable Coatings are:

- Carboline CZ 11 is an inorganic zinc (IOZ) coating used as a primer in Containment on structural steel, handrails, ladders and other ferrous field-coated substrates.
- Carboline 191 HB is an epoxy coating used in Containment on concrete and steel surfaces.
- Carboline 195 is a primer-surfacer used in Containment on concrete walls and floors.
- Carboline 4674 is a zinc-free silicone acrylic applied on major NSSS equipment in Containment.
- Amercoat 66® is a high build polyamide-cured epoxy used in Containment for protection of steel, concrete or Dimetcote surfaces.
- Carboline 890 is an epoxy coating used in Containment on steel surfaces.

DBA-unqualified coating systems are inorganic zinc, epoxy and alkyds.

3h.2 Bases for Assumptions Made in Debris Transport Analysis

The post-DBA debris evaluations of all coatings were all based on NEI-04-07 and/or testing as discussed below.

The debris generation assumption for qualified coatings in the zone of influence (ZOI) of the LOCA are based on testing performed on representative coating systems as documented in WCAP-16568-P (Reference 8). This testing concluded that a spherical ZOI of 4D is conservative for the qualified coating systems, with the exception of untopcoated inorganic zinc (IOZ) systems. A 5D ZOI is utilized for untopcoated IOZ primers.

Unqualified coatings under intact insulation were not assumed to fail. However, the unqualified coatings under destroyed insulation were assumed to fail within a 10D ZOI.

For debris generation and transport analysis, 10-micron particles were assumed for qualified epoxy coatings within the 4D ZOI. Qualified coatings outside the 4D ZOI were not assumed to fail.

For debris generation and transport analysis, 10-micron particles were assumed for DBA-unqualified coatings within a 10D ZOI. In addition, 100% of the DBA-unqualified and degraded coatings outside the 4D ZOI were assumed to fail as 10-micron particles [except where based on testing and plant specific conditions as described below]. Coatings (unqualified) under intact insulation were not assumed to fail. However, the coatings under destroyed insulation were assumed to fail within a 10D ZOI.

Testing performed for Comanche Peak Steam Electric Station by Keeler & Long (Reference 11) has been reviewed and found applicable to the degraded DBA-qualified epoxy and inorganic zinc coatings applied at WCNOC. In the test, epoxy topcoat / inorganic zinc primer coating system chips, taken from the Comanche Peak Unit 1 containment after 15 years of nuclear service, were subjected to DBA testing in accordance with ASTM D 3911-03. In addition to the standard test protocol contained in ASTM D 3911-03, 10 μm filters were installed in the autoclave recirculation piping to capture small, transportable particulate coating debris generated during the test.

The data in this report shows that inorganic zinc predominantly fails in a size range from 9 to 89 microns with the majority being between 14 and 40 microns. Therefore, a conservative size of 10 microns was assumed for transport and head loss analysis of inorganic zinc. The data in this report also showed that DBA-qualified epoxy that has failed as chips by delamination tend to remain chips in a LOCA environment. The data showed that almost all of the chips remained larger than 1/32-inch diameter. Therefore, a chip diameter of 1/32 inch was used for transport for Phenoline 305 epoxy coatings shown to fail as chips by delamination. Consistent with manufacturer's publish data sheets and material safety data sheets (MSDSs), Carboline Phenoline 305 is conservatively representative of the other DBA-qualified/Acceptable epoxy coatings found in US nuclear power plants (Reference 9). This includes Mobil 78, Mobil 89, Amercoat 66, Keeler & Long 6548/7107 and Keeler & Long D-1 and E-1.

For original equipment manufacturer (OEM) coatings, industry testing documented in EPRI test report 1011753 (Reference 12), was used to determine that 10 microns is a very conservative assumption for particle sizes. None of the OEM coatings failed as chips. Therefore, a 10-micron particle size was used for transport and head loss analyses. This report also showed that, on average, much less than half of OEM coatings detached and failed during testing. Based on the EPRI test results and the conservative assumption of 10-micron particle size, 100% failure of all OEM coatings is highly conservative.

Based on the review of the EPRI report 1011753, it has been determined that a reduction in the failure percentage for the OEM epoxy could be justified. The failure percentage for OEM epoxy of 50% was used in the transport analysis. The failure percentage bounds the worst performing sample for this type in the EPRI test data. The failure percentage of all other OEM coatings is assumed at 100%.

It is assumed that unqualified coatings debris outside the ZOI that are in the following categories are not transported to the sumps:

a) coatings within an inactive sump,

b) coatings covered by intact insulation, or

c) coatings otherwise isolated from spray.

Based on industry test results for degraded qualified coatings, the debris generation calculation assumes that the epoxy chips fail as flat chips in sizes of fines, 1/64 inch, 1/8 inch, ¼ inch, ½ inch, 1 inch and 2 inches. It is assumed that this size distribution is fine particulate (smaller than 1/64 inch), fine chips (1/64 inch in size), small chips (3/8 inch in size), and large chips (1.5 inch in size) in order to simplify the analysis and bound the size distribution.

The debris generation calculation states that a portion of all of the epoxy chips also fail as curled chips, based on test data available to the industry. It is assumed that all curled chips in the size distribution could be simplified into a 1.5 inch chip size. The tumbling velocity for this size, curled chip was determined using a correlation of the NUREG/CR-6916 (Reference 13) data for the curled chips tested (1 inch to 2 inch). Note that curled chips tumble along the floor more easily than flat chips which is likely due to the larger frontal area exposed to flow for the curled chips. Therefore, since the slightly smaller 1.5 inch chips tend to have a smaller frontal area, this is considered to be a reasonable and conservative assumption.

It is assumed that the different density and thickness of chips postulated in the debris generation calculation is represented with a single chip density of 94 lb/ft^3 and a thickness of 6 mils. This assumption is conservative because it yields the lowest transport metrics, which bounds all other chip sizes.

It is assumed that the unqualified coatings debris is initially distributed in the locations where they are applied or in the locations where they are washed down. This is a reasonable assumption since unqualified coatings would fail gradually and would likely fail after recirculation has been initiated.

3h.3 Coatings information for Strainer Head Loss Testing

The coating debris amounts transported to the sumps from a LOCA are shown above in Tables 3e-1 through 3e-5. Quantities for the testing were then based on a scaling factor derived from the test module size to total strainer size ratio. The surrogate materials used in head loss testing are shown in Table 3h-1.

Table 3h-1. Coating Surrogates for Strainer Performance Test

Coating Debris	Coating Density, lb/ft^3	Surrogate Used	Surrogate Density, lb/ft^3
IOZ	457	Tin Powder	455
Carboline 191 HB Epoxy	104	Powder (walnut shells)	74.9 to 93.6
Carboline 4874 Acrylic	86	Powder (walnut shells)	74.9 to 93.6
Carboline 195 Surfacer Epoxy	115	Powder (walnut shells)	74.9 to 93.6
Carboline 890 Epoxy	112	Powder (walnut shells)	74.9 to 93.6
OEM Unqualified Alkyd	105	Powder (walnut shells)	74.9 to 93.6
OEM Unqualified Epoxy	111.5	Powder (walnut shells)	74.9 to 93.6
Epoxy, Fine Chips (1/64")	94	Chips (Carboline CarboGuard 890/891, similar type coatings)	94
Epoxy Curled Chips	94	Chips (Carboline CarboGuard 890/891, similar type coatings)	94

3h.4 Basis for Choice of Surrogates

Assumptions made and/or data used to justify use of surrogates listed in above table:

- Particles of "like" size, shape and density will perform in the same way as other particles of "like" size, shape and density.
- Particles of similar size that are less dense will suspend more easily, and when added to the debris mix at the postulated mass of the actual coating material is bounding and conservative for these tests.
- Particles of smaller sizes will bound particles of larger sizes. This is because smaller particles can fill more of the interstitial spaces between fibers than will larger particles; which will increase head loss on a relative scale.
- Zinc has a specific density of 457 lb/ft^3 and tin has a specific density of 455.1 lb/ft^3.
- Walnut shells have a density range of 74.9 to 93.6 lb/ft^3.

Coating chips were manufactured from dried Carboline CarboGuard 890/891 epoxy coating, or similar type coating, with a density of approximately 94 lbs/ft^3.

Walnut shell flour (based on density, size, shape, texture, etc.) is a bounding and conservative surrogate material for coatings with densities above 75 lbs/ft^3 such as epoxy, enamel, acrylic, and alkyd coatings. Coating chips from Carboline CarboGuard 890/891 epoxy coating, or similar type coating, is a suitable surrogate based on density. For inorganic zinc coatings (including primers), the use of tin powder is an acceptable surrogate.

3h.5 Basis for Coatings Debris Generation Assumptions

A 4D destruction ZOI for coatings has been assumed for all qualified coatings systems (with the exception of untopcoated IOZ systems) based of Wyle Labs testing documented in WCAP-16586-P (Reference 8). WCAP-16568-P supports the use of a 5D ZOI for untopcoated IOZ primers.

Where specific epoxy/phenolic coating systems could not be identified from documentation reviewed, the density for Carboline 890 was used as representative. The density of Carboline 890, determine from vendor information was considered conservative since it exceeds the value (94 lb/ft^3) recommended in Table 3-3 of NEI 04-07, Vol. 1 (GR). (Reference 1) and the values determined for the Carboline systems.

Qualified coatings outside of the ZOI are considered to remain intact for this baseline evaluation. All degraded qualified coatings outside of the ZOI are assumed to fail with the IOZ primer portion becoming particulate and the epoxy portion becoming chips as follows:

a. IOZ - 10μm particulate, equivalent in size to the average zinc particle in IOZ coatings or the pigment used in epoxy coatings (consistent with Section 3.4.3.6 of NEI 04-07, Vol. 2 (SE). (Reference 1).
b. Epoxy – as chips with the debris characteristics shown in Table 3h-2.

Table 3h-2. Degraded Qualified Coatings Size Distribution

Range (inches)	1.0 – 2.0	0.5 – 1.0	0.25 – 0.50	0.125 – 0.25	<0.125*
Weight Distribution (fraction)**	0.32	0.09	0.04	0.05	0.37 / 0.12

*Chips <0.125 inches have an additional size distribution. Approximately 37% are assumed to be 15.6 mils (chips) and approximately 12% are assumed to be 6 mils (particulate). Chips <0.125 inch are assumed to not curl.

**Chips larger than 0.5 inches will be assumed to be 57% curled and 43% flat (TXU Paint Chip Characterization). It is assumed that all other chip sizes will be 8.2% curled and 91.8% flat to bound the overall fraction of chips found to be curled. (the IOZ portion of the chip fails as particulate)

All unqualified coatings inside and outside of the ZOI are assumed to fail as particulate. Section 3h.2 above addresses failure percentages for OEM coatings.

3h.6 Coatings Debris Characteristics

Unqualified Coatings:
The method used to reduce the amount of indeterminate coatings assumed to fail in the containment building was based on the information contained in EPRI Report 1011753. An overall assessment of the failure rate of the specified coatings was used to determine an

appropriate reduction for each type. EPRI performed testing on 37 samples of coatings on common vendor-supplied equipment. The data reported suggested the following:

a, Generally, less than 100% of the coatings fail, and,

b. The failed coating debris has a size distribution associated with it that was larger than the 10-20 micron size identified in NEI 04-07 Vol. 1 (GR), and accepted in NEI 04-07, Vol. 2 (SE).

The method used to reduce indeterminate coatings load was as follows:

a. Categorize the indeterminate coatings by coatings type.

b. Relate those coatings types of coatings tested under the EPRI program.

Then using the data determined above,

a. An appropriate conservative percentage of coatings failures for that type of coating was identified, and

b. An appropriate debris size distribution was identified, if it was supported by the data.

Overall, OEM unqualified coatings were found to fail at an average rate of 20.4%, which is significantly less than the 100% failure rate described in Section 3.4.2.1 of NEI 04-07, Vol. 2 (SE). Per the EPRI Report, a coating failure was defined as the detachment of the coating from the surface to which it was applied. Therefore, failure rate is defined as the amount of surface area that experiences a detachment of its coating.

<u>3h.7 Containment Coatings Condition Assessment Program</u>

The acceptability of visual inspection as the first step in monitoring of Containment Building coatings is validated by EPRI Report No. 1014883 (Reference 14). Monitoring of Containment Building coatings is conducted at a minimum, once each fuel cycle in accordance with plant procedures. Monitoring involves conducting a general visual examination of all accessible coated surfaces within the containment building, followed by additional nondestructive and destructive examinations of degraded coating areas as directed by the plant Protective Coatings Specialist. Examinations and degraded coating areas are conducted by qualified personnel as defined in plant procedures as recommended by ASTM D 5163-05a. Detailed instructions on conducting coating examinations, including deficiency reporting criteria and documentation requirements are delineated in plant procedures.

3i. <u>Debris Source Term</u>

In responding to GL 2004 Requested Information Item 2(f), provide the following:

1. *A summary of the containment housekeeping programmatic controls in place to control or reduce the latent debris burden. Specifically for RMI/low-fiber plants, provide a description of programmatic controls to maintain the latent debris fiber source term into the future to ensure assumptions and conclusions regarding inability to form a thin bed of fibrous debris remain valid.*
2. *A summary of the foreign material exclusion programmatic controls in place to control the introduction of foreign material into the containment.*
3. *A description of how permanent plant changes inside containment are programmatically controlled so as to not change the analytical assumptions and numerical inputs of the*

licensee analyses supporting the conclusion that the reactor plant remains in compliance with 10 CFR 50.46 and related regulatory requirements.

4. *A description of how maintenance activities including associated temporary changes are assessed and managed in accordance with the Maintenance Rule, 10 CFR 50.65.*
5. *Summarize the application of the suggested design and operational refinements given in Section 5 of NEI 04-07, Vol. 1 (guidance report) and Section 5.1 of NEI 04-07, Vol. 2 (SE).*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(f) (Reference 2). The debris source term controls described below conforms to Section 5.1 of NEI 04-07, Vol. 2 (SE) (Reference 1).

3i.1 Containment Latent Debris Housekeeping Programmatic Controls

WCNOC implemented a containment latent debris assessment program, which utilizes swipe sampling to determine the amount of latent debris in containment, consistent with Section 5.1 of NEI 04-07, Vol. 2 (SE). Housekeeping and foreign material exclusion program procedures have been revised to target the Containment cleaning effort from the results of the swipe sampling survey and to enhance the cleanliness requirements in Modes 1 through 4.

Plant preventative maintenance documents specify that the latent debris sampling is conducted every other outage consistent with section 5.1 of NEI 04-07, Vol. 2 (SE). Procedures also provide guidance to conduct the latent debris sampling process in the containment when major work activities are performed, e.g., steam generator replacement.

3i.2 Foreign Material Exclusion Programmatic Controls

The foreign materials exclusion program provides the guidance that the containment building is considered a system during plant modes 1 through 4 and refers to the containment entry and materials control procedure. The containment entry and materials control procedure provides guidance following containment entries during plant modes 1 though 4 to thoroughly clean the immediate work area and other areas where debris may have migrated during the work activity. A containment inspection surveillance is then conducted following containment entries during plant modes 1 though 4 to ensure cleanliness. During plant modes 5, 6, and defueled, the containment entry and materials control procedure also provides guidance for general containment cleaning.

3i.3 Plant Modification Process Programmatic Controls

Administrative controls for the plant modification process require a specific response to the following questions pertaining to ECCS and CSS recirculation functions:

- Does the change add any potential recirculation water holdup point or restricted flow areas upstream of the containment sump strainers inside the Containment Building?
- Does the change modify piping, storage tank capacity or add or remove equipment such that Containment flood levels are changed?
- Does the change add or modify any component in the ECCS or CSS recirculation flow path (equipment, valves, instruments, nuclear fuel, etc.) that may cause flow restrictions or blockage of flow paths or suffer abrasive damage due to post-LOCA debris?

- Does the change add or modify material (i.e., insulation, lead shielding blankets, tags/labels, etc.) inside the Containment that could become loose debris during a pipe break or Containment flooding event?
- Does the change Add or remove aluminum or zinc from Containment?
- Does the change add any safety or non-safety component/subcomponent (including steel supports, piping etc.) inside Containment that has not been previously painted with a qualified coating?
- Does the change require the post installation application of a coating of any component or subcomponent located inside Containment?
- Does the change modify any existing or add new coatings inside the Containment; steel or concrete?

Any yes answer to the above questions requires the Engineer to consult with the designated expert, to determine a solution.

In addition to the design change controls, WCNOC procedurally tracks all transient materials taken inside Containment. During normal operations, all items taken into Containment are logged. At the completion of the Containment entry the items are accounted for. At the completion of work activities inside Containment during normal operations, the work area is thoroughly cleaned and inspected (including the area below the work activity, if the work was performed on grating), prior to leaving Containment. Items left in Containment during normal operations are either stored in a container with a latchable door or cover, or secured to a structural member to prevent possible transport to the sumps.

3i.4 Assessment and Management of Maintenance Activities

Procedures are in place to control maintenance activities and evaluate temporary changes that have the potential to affect the debris source term.

The containment entry and material control procedures contain requirements for control of materials during work activities conducted in the containment building. Following maintenance activities in the containment building, procedures that control the containment cleanliness verification process specifically require both general area and target area cleaning.

Changes implemented as temporary alterations in support of maintenance that impact plant design are required to be developed in accordance with the same design change procedures that are used for all plant modifications. As described in Section 2, the plant modification procedures contain administrative controls that specifically address potential impacts of debris on the ECCS performance.

3i.5a Summary of the application of the following refinement, if applicable: Recent or planned insulation change-outs in the containment which will reduce the debris burden at the sump strainers.

There are no planned insulation change-outs that would reduce the debris burden at the sump strainers.

3i.5b Summary of the application of the following refinement, if applicable:
Any actions taken to modify existing insulation (e.g., jacketing or banding) to reduce the debris burden at the sump strainers.

There are no planned actions to modify existing insulation (e.g., jacketing or banding) to reduce the debris burden at the sump strainers.

3i.5c Summary of the application of the following refinement, if applicable:
Modifications to equipment or systems conducted to reduce the debris burden at the sump strainers.

There are no planned modifications to equipment or systems to reduce the debris burden at the sump strainers.

3i.5d Summary of the application of the following refinement, if applicable:
Actions taken to modify or improve the containment coatings program.

There are no planned actions to modify the existing containment coatings to reduce the debris burden at the sump strainers.

3j. Screen Modification Package

The objective of the screen modification package section is to provide a basic description of the sump screen modification.

1. *Provide a description of the major features of the sump screen design modification.*
2. *Provide a list of any modifications, such as reroute of piping and other components, relocation of supports, addition of whip restraints and missile shields, etc., necessitated by the sump strainer modifications.*

3j.1 Description of Major Features of Sump Strainer Design Modification

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(a). (Reference 2)

Description of sump strainer design modification:
New sump strainers were installed in the two Containment Recirculation Sumps during WCGS Refueling Outage XV. The Performance Contracting, Inc. (PCI) Sure-Flow™ strainers were installed in the sump pits to accommodate the post-accident containment water levels. Figure 3j-1 shows an isometric view of the location of the sumps in the containment building below two safety infection accumulator tanks located in the lower left portion of the figure. Figure 3j-2 shows a closer view of the sump pits.

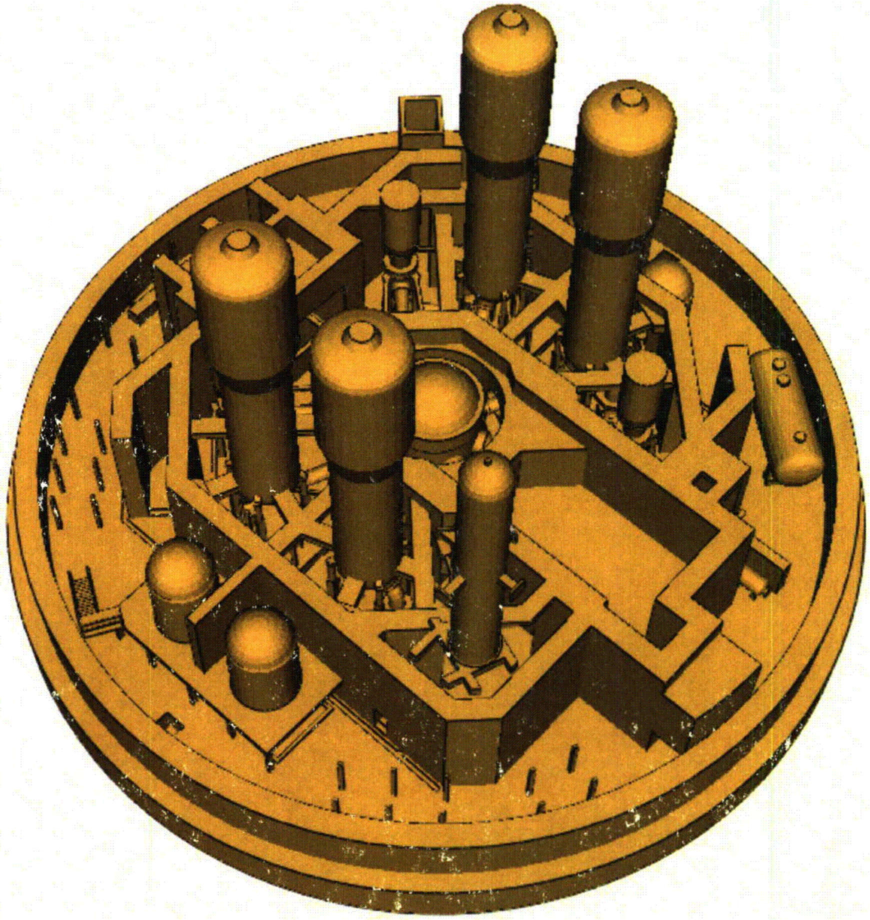


Figure 3j-1. Isometric View of Lower Containment

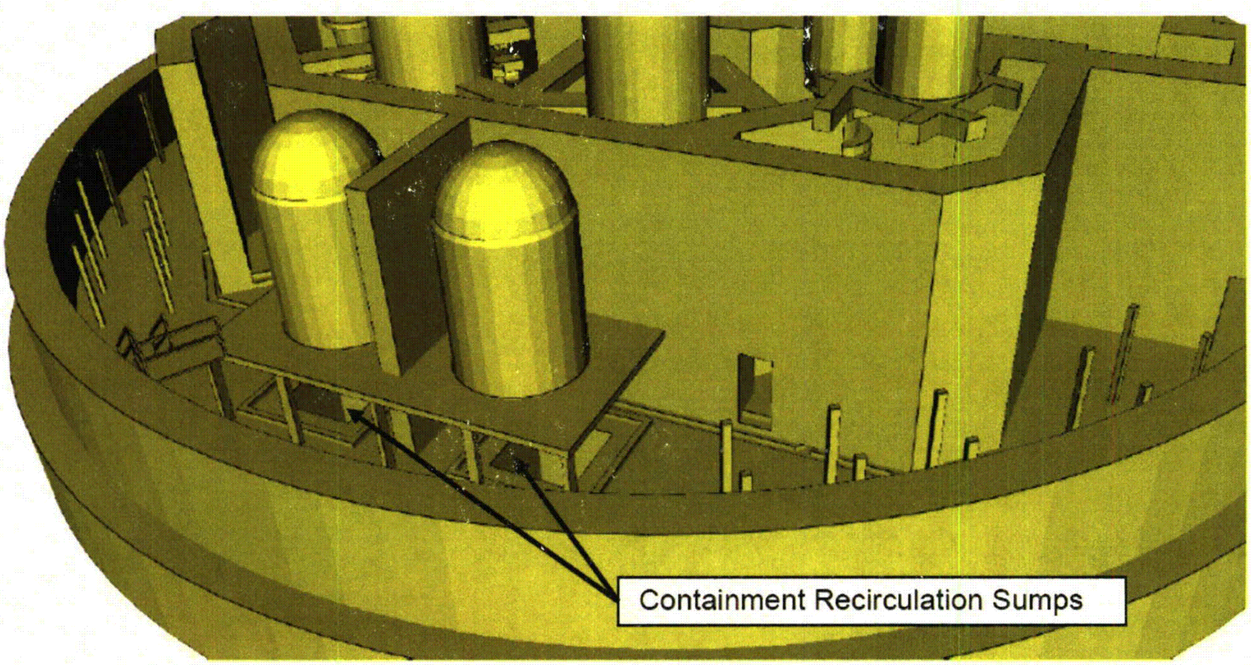
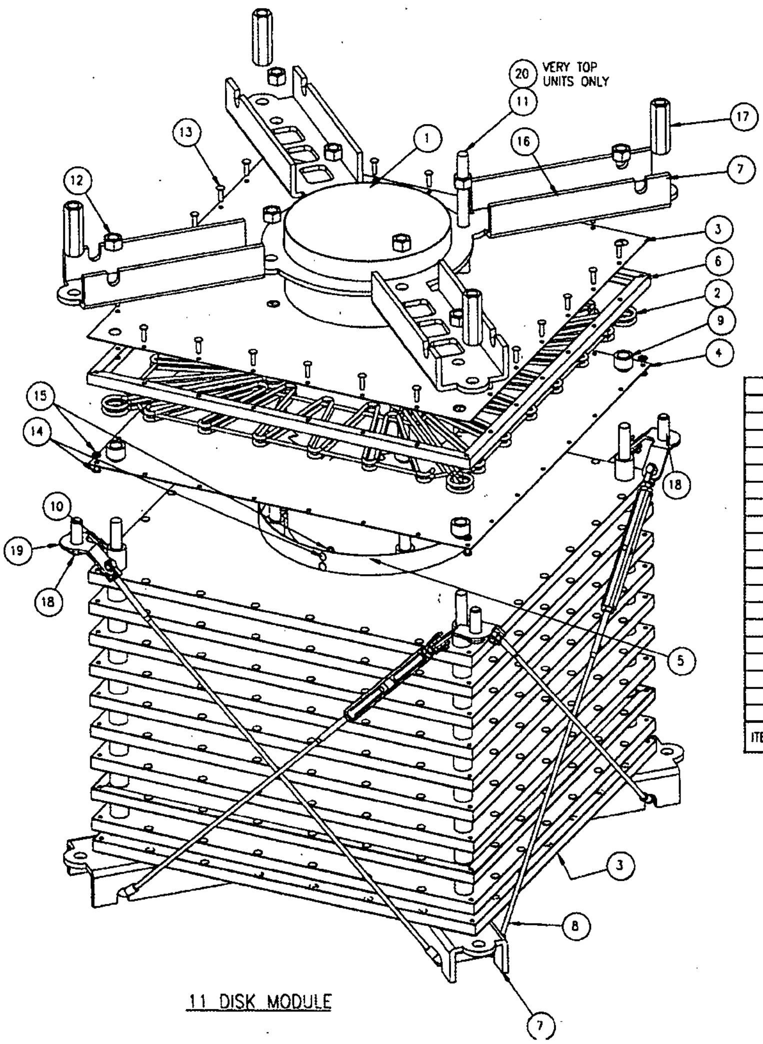


Figure 3j-2. Close-up View of Lower Containment

Each new sump strainer is made up of 72 modules. Eight modules are seven plate/disks high and 64 modules are eleven plates/disks high (Refer to Figure 3j-3). The modules are arranged in a square matrix of 16 modules on each level, except for the bottom level that has only eight modules (Refer to Figures 3j-4 and 3j-5). Each stack of modules (see Figure 3j-5) is an integrated unit that equalizes the flow rate and corresponding pressure drop across the perforated plate at each level and allows for a distributed pressure drop across the column. The strainers are installed on a strainer substructure assembly, which is installed at the bottom of the containment recirculation sump pit. The strainers superstructure consists of four vertical supports on the 2000' elevation concrete pad. These supports are inside the sumps 6" concrete curb. A series of horizontal channels connect to the four vertical supports and provide lateral restraint for the module stacks. The strainers are robust so as to also serve as the trash racks, as described in a license amendment application (Reference 18) and approved in the associated NRC safety evaluation (Reference 19). The strainers have 0.045" holes in the perforated stainless steel plate surfaces. The materials for the strainer supports, both the lower support platform and the superstructure, are also stainless steel.



ITEM No.	DESCRIPTION
20	TENSION ROD-LONG, 1/2"ø-13UNC x 18 3/4" LG., SS
19	SEISMIC CABLE BRACKET, 11ga., SS
18	BOLTS, 1/2"-13 UNC X 1 1/4", SS
17	COUPLING 1/2-13UNC X 2 1/2" LG. SS
16	I.D. TAG, SS
15	BACKER WASHER, SS
14	3/16"ø, CLOSED END RIVETS W/ GRIP RANGE OF 0.020"-0.125", SS
13	3/16"ø, OPEN END, BLIND RIVETS W/ GRIP RANGE OF 0.626"-0.75", SS
12	HEAVY HEX NUT, 1/2"-13 UNC, SS
11	TENSION ROD, 1/2"ø-13UNC x 18 1/8" LG., SS
10	SPACER, 1" x 1/2"ø SCH 80, SS PIPE
9	SPACER, 1/2" x 1/2"ø SCH 80, SS PIPE
8	SEISMIC CABLE ASSEMBLY, SS
7	EXTERNAL RADIAL/DEBRIS STOP, 1/4" THK., SS
6	RIM DISK, 22ga. PERF PLATE, SS
5	GAP DISK, 22ga. PERF PLATE, SS
4	INTERMEDIATE DISK FACE, 22ga. PERF PLATE, SS
3	FIRST & LAST DISK FACE, 22ga. PERF PLATE, SS
2	GRILL, MAD OF 7 ga. & 8 ga. WIRES, SS
1	CORE TUBE, 7.5"ø, 16 ga., SS

Figure 3j-3. 11 Disk Sump Strainer Module Detail

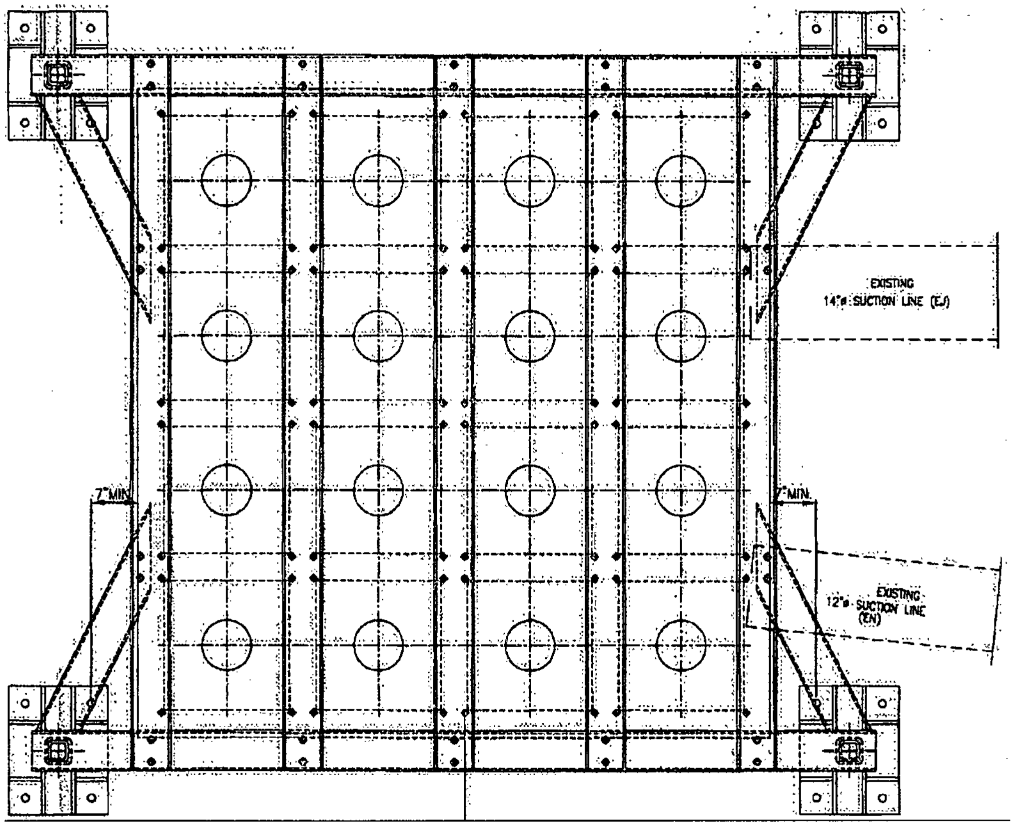


Figure 3j-4. Sump Strainer Plan View

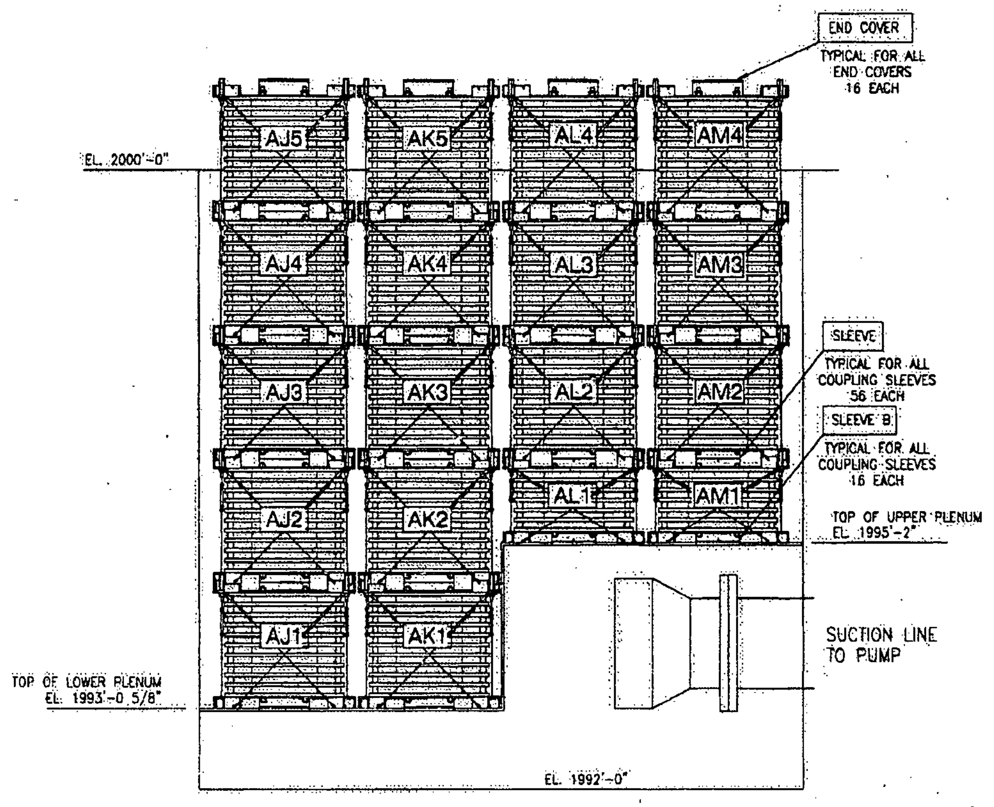


Figure 3j-5. Sump Strainer Section View

The original containment recirculation sump screens and trash racks had approximately 200 ft^2 of effective surface area per sump. The new replacement sump strainers have approximately 3300 ft^2 of effective surface area per sump that can handle the amount of debris generated and carried to the sumps. A significant design feature of the new PCI Sure-Flow® strainers ensures uniform flow rate through all sections of the modules. This ensures that during post-accident operation, debris is not preferentially distributed to certain areas of the strainer. Additionally, as a result of the increased surface area, the approach velocity of the recirculation coolant flow at the sump strainer face will be less than 0.01 feet per second.

3j.2 Other Sump Strainer-related Design Modifications

As mentioned above in Section 3e.4 above, debris barriers have been installed in all openings through the secondary shield wall near the emergency recirculation sumps. The barriers prevent the flow of debris-laden fluid directly to the sumps and force the fluid to take a "long path" through shield wall openings farther away from the sumps. Using perforated plates with a hole size of 1/8 inch, the debris barriers are designed to restrict passage of debris while allowing water to pass through the barrier. While not specifically credited in the debris transport analysis, the effect of any water flow through the "A" and "D" debris will lower pool velocity out the "B" and "C" loop openings. This is conservative compared to the current transport analysis which assumes all water flows out the "B" and "C" loop openings. Debris barriers have been installed in the Loop A and Loop D passageway entrances through the secondary shield wall, as well as in drain trenches and other openings in the secondary shield wall near the sumps. Blockage of small and large piece debris through the Loop A and Loop D passageways and other openings was included in the transport modeling.

As a result of the new strainer design, sump level indication was also replaced. The new instrumentation provides an indication of strainer differential pressure. The differential pressure measurement provides the control room operators a qualitative indication of how well the strainer is performing during the recirculation functions of the Emergency Core Cooling System and Containment Spray System following all postulated accidents for which the operation of these systems is required.

3k. Sump Structural Analysis

The objective of the sump structural analysis section is to verify the structural adequacy of the sump strainer including seismic loads and loads due to differential pressure, missiles, and jet forces.

Provide the information requested in GL 2004-02 Requested Information Item 2(d)(vii).

GL 2004-02 Requested Information Item 2(d)(vii) Verification that the strength of the trash racks is adequate to protect the debris screens from missiles and other large debris. The submittal should also provide verification that the trash racks and sump screens are capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under flow conditions.

1. *Summarize the design inputs, design codes, loads, and load combinations utilized for the sump strainer structural analysis.*
2. *Summarize the structural qualification results and design margins for the various components of the sump strainer structural assembly.*
3. *Summarize the evaluations performed for dynamic effects such as pipe whip, jet impingement, and missile impacts associated with high-energy line breaks (as applicable).*
4. *If a backflushing strategy is credited, provide a summary statement regarding the sump strainer structural analysis considering reverse flow.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(d)(vii) (Reference 2). The sump structural analysis process described below conforms to Section 7.1 of NEI 04-07, Vol. 2 (SE) (Reference 1).

3k.1 Structural Analysis Design Input, Design Codes, and Load Combinations

The WCGS sump strainer structural qualification analysis evaluated the strainer modules as well as the supporting structures associated with the strainers. The governing code for qualification of the strainer is the WCNOC code of record, the American Institute of Steel Construction (AISC), 7th edition. In circumstances where the AISC code does not provide adequate guidance for the particular component, other codes or standards are used for guidance. The evaluations were performed using a combination of manual calculations and finite element analysis using the GTSTRUDL Computer Program and the ANSYS Computer Program.

The strainers are designed for the following load combinations:

Seismic loads – The strainers are designed to meet Category I Seismic Criteria. Both the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) loads are developed from response spectra curves that envelope the response spectra curves for WCGS. The structures are considered "Bolted steel structures" and the damping values for seismic loads are taken from Regulatory Guide 1.61 as 4% for the OBE and 7% for the SSE.

Live Loads – Live loads include the weight of the debris accumulated on the strainer and the differential pressure across the strainer perforated plates in the operating condition.

Thermal Loads – Thermal expansion is considered in the design and layout of the structures. The strainers themselves are free to expand in the vertical direction as the superstructure is designed with a sliding connection allowing the strainer modules to expand upward without constraint. In the lateral direction, the seismic supports are gapped leaving enough of a gap to accommodate the thermal growth of the strainers and their supports without restraint. The design temperature for the strainers is 268 °F, which is the maximum calculated containment sump water temperature during a large break LOCA. The maximum air temperature inside containment can reach as high as 320 °F, however this is a very short term spike and the structure would not have time to heat up to this temperature before the containment air temperature would fall back down to lower levels. Therefore, the use of the maximum water temperature for material properties and thermal expansion is appropriate.

Hydrodynamic loads - Hydrodynamic loads on the strainers from the motion of the water surrounding the strainer during a seismic event were also considered.

A structural and seismic evaluation was also performed on the instrument support elements associated with the new strainers. The evaluations were performed using manual calculations.

3k.2 Structural Qualification and Design Margins

The structural qualification design margins for the various components of the sump strainer structural assembly are listed in Table 3k-1.

Table 3k-1. Sump Strainer Structural Assembly Components Design Margins

(Table 3k-1) **Strainer Component**	**Interaction Ratio[1]**
External Radial Stiffener (Including Collar and Plates)	0.23 / 0.89
Tension Rods	0.48 / 0.56
Spacers	0.79 / 0.87
Edge Channels	0.08 / 0.69
Cross Bracing Cables	0.15 / 0.40
Hex Couplings	0.20 / 0.73
Core Tube	0.03 / 0.10
Substructure Angle Iron Support Legs	0.32 / 0.62
Substructure Angle Iron Framing (including coped sections and angle braces)	0.52 / 0.77
Substructure Channels (including coped sections)	0.53 / 0.81
Cover Plates	0.08 / 0.24
Superstructure Square Tubing Support Legs	0.22 / 0.62
Superstructure Channels	0.15 / 0.24
Perforated Plate (DP Case)	0.47 / 0.39
Perforated Plate (Seismic Case)	0.66 / 0.55

(Table 3k-1) Strainer Component	Interaction Ratio[1]
Perforated Plate (Inner Gap)	0.70 / 0.79
Wire Stiffener[2]	0.39
Perforated Plate (Core Tube End Cover DP Case)	0.52 / 0.44
Perforated Plate (Core Tube End Cover Seismic Case)	0.17 / 0.14
Radial Stiffening Spokes of the End Cover Stiffener	0.09 / 0.08
Core Tube End Cover Sleeve	0.06 / 0.04
Weld of Radial Stiffener to Core Tube	0.03 / 0.06
Weld of mounting tabs to End Cover Stiffener	0.02 / 0.01
Weld of End Cover Stiffener to End Cover Sleeve	0.03 / 0.02
Edge Channel Rivets	0.05 / 0.52
Inner Gap Hoop Rivets	0.09 / 0.11
End Cover Rivets	0.06 / 0.04
Connecting Bolts and Pins	0.37 / 0.67
Mounting Pin Weld	0.33 / 0.91
Substructure Sealing Plates	0.88
Substructure Bolted Connections	0.40 / 0.56
Substructure Welded Connections	0.30 / 0.61
Substructure Post Jack Bolt and Baseplate	0.57 / 0.92
Substructure Wall Jack Bolts	0.21 / 0.29
Superstructure Bolted Connections	0.10 / 0.41
Superstructure Welded Connections	0.27 / 0.80
Superstructure Expansion Anchors[3]	0.16 / 0.89 / 0.62
Superstructure Anchor Base Plate[3]	0.20 / 0.70 / 0.82
Superstructure Anchor Base Plate Stiffener Welds[3]	0.22 / 0.86 / 0.43

1. Interaction Ratio, i.e., the calculated stress divided by the allowable stress. Listed as OBE / SSE cases unless noted otherwise.
2. DP loads only
3. Worst case OBE /SSE for ShearX, ShearY, and Tension

3k.3 High-energy Break Dynamic Effects

The location of the recirculation sump strainers outside the secondary shield walls eliminates the need to consider a LOCA jet impingement load on the strainers.

3k.4 Backflushing Reverse Flow Considerations

WCNOC is not crediting a backflushing strategy for mitigating an excessive strainer head loss condition.

3l. Upstream Effects

The objective of the upstream effects assessment is to evaluate the flowpaths upstream of the containment sump for holdup of inventory which could reduce flow to and possibly starve the sump.
Provide a summary of the upstream effects evaluation including the information requested in GL 2004-02 Requested Information Item 2(d)(iv).
1. *Summarize the evaluation of the flow paths from the postulated break locations and containment spray washdown to identify potential choke points in the flow field upstream of the sump.*
2. *Summarize measures taken to mitigate potential choke points.*
3. *Summarize the evaluation of water holdup at installed curbs and/or debris interceptors.*
4. *Describe how potential blockage of reactor cavity and refueling cavity drains has been evaluated, including likelihood of blockage and amount of expected holdup.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(c).9. (Reference 2) The upstream effects evaluation process described below conforms to Section 7.2 of NEI 04-07, Vol. 2 (SE). (Reference 1)

3l.1 Evaluation to Identify Potential Choke Points in Flow Paths Upstream of the Recirculation Sumps

The WCGS upstream effects evaluation includes an assessment of the WCGS containment geometry and transport pathways that containment spray flow and ECCS leakage from the RCS will follow to the lower elevations of the containment building. The evaluation is based upon a review of WCGS design drawings and photographs of inside the containment building.

Each elevation of the containment building was reviewed to identify the physical and structural features that affect the flow of debris and water to the lower elevations of the containment building. The containment building was divided into seven general compartments for individual evaluation, separated by grating, concrete walls, and concrete floors.

1. Upper containment including lay down area (elevation 2068'-8" to dome elevation 2205'-0")

   Overall the area at this elevation is open with numerous areas of floor grating, which would allow water to pass through to the lower elevations unencumbered. There is one small area of concrete flooring near the pressurizer valve rooms, but water in this location will flow to the grated or open areas surrounding it. This area is open inside and outside the secondary shield walls down to the operating floor at elevation 2047'-6". No potential choke points or hold-up points were identified in this area.

2. Operating Floor (elevation 2047'-6")

   Overall the area is open inside and outside the steam generator secondary shield walls to allow water flow to the lower elevations. The area is open inside these secondary shield walls down to elevation 2001'-4" and outside the secondary shield walls down to the operating floor at elevation 2047'-6". No potential choke points were identified at this elevation.

   A hold-up point in this area of the containment building is the reactor head storage and decontamination area. Water collected on the head stand will drain to the surrounding floor but water will be retained by a curb surrounding the head stand. The area inside the curb has a 4" floor drain that directs water to a common drain header and then to the drain trenches at the ground floor elevation. However, the drain could become plugged with debris and is not considered functional for this evaluation.

3. Annulus and Inside Secondary Shield (elevation 2026'-0"):

   Major equipment and features in this area include the main steam and feedwater lines, the tops of the A and D safety injection accumulators, several HVAC openings, and a compartment for the letdown orifices, the top of which is located at elevation 2036'-0". The northern, northwestern, and southwestern sides of the annulus have mostly concrete floors while the rest of the elevation outside the secondary shield is grated. There are no curbs associated with the concrete floors, so water inventory will flow to the lower elevations without holdup. No potential choke points or hold-up points were identified at this elevation.

4. Refueling pool (elevation 2009'-9" and elevation 2007'-2"):

   The refueling pool floor (elevation 2009'-9") contains two 10" drains that are sealed with flanges during refueling operations and are completely open during power operations. There are debris exclusion devices (trash rack cages) installed during power operations to prevent the drain from becoming a choke point. Each debris exclusion device, described in more detail below, is welded to a flange which is bolted to the drain. The flange is approximately two inches thick which creates a two inch hold-up volume below the flange elevation.

   There is an upending pit below the refueling pool floor elevation at elevation 2007'-2". The drain in the upending pit is a 4" line which is normally isolated with a normally closed valve. This area is a hold-up point that would retain water inventory following a postulated design basis accident (post-DBA). No potential choke points were identified at either of these elevations.

5. Ground Floor Inside Secondary Shield (elevation 2001'-4"):

   There are only four significant openings through which post-DBA recirculation water may pass through the secondary shield wall. These passageways provide personnel and equipment access through the secondary shield wall in an area near each of the four reactor coolant pumps (RCPs), and include steps to transition from the 2001'-4" floor elevation inside the secondary shield wall to the 2000'-0" floor elevation outside the wall. Three of the four openings are approximately six feet wide. The fourth opening, entering under the pressurizer near the "D" loop RCP, is approximately three feet wide. In Figure 3I-1, the opening near the "A" loop RCP is shown to the left of the sump pits and the opening near the "D" loop RCP is shown to the right of the sump pits.

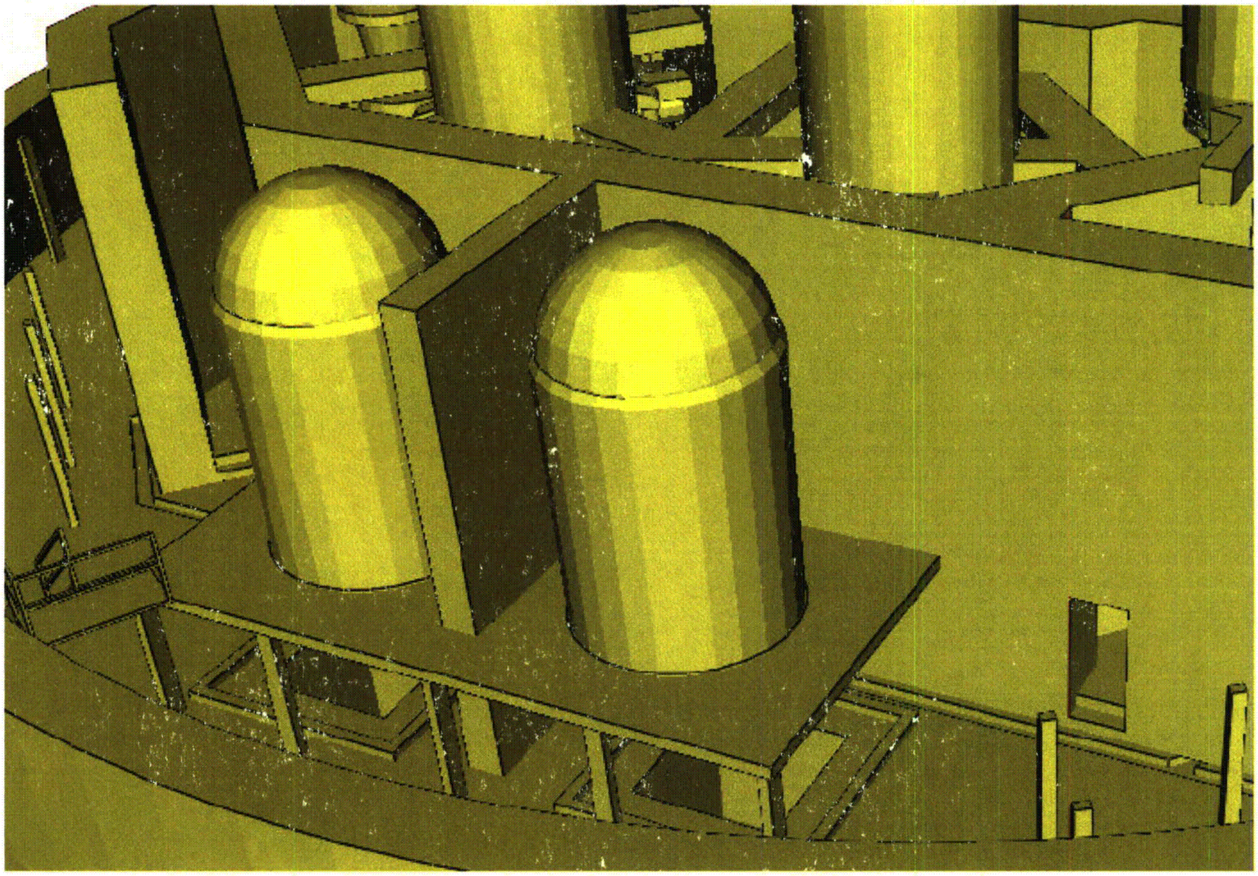


Figure 3I-1. Isometric View of Sump Pit Area

Additionally, there is a system of small drain trenches, approximately one foot wide by one foot deep, that surround the primary shield wall and transfer drain water to outside the secondary shield wall. Trenches and drain piping outside the secondary shield walls direct drainage to the normal containment sumps (which are not part of the ECCS system) located in the containment ground floor annulus at elevation 2000'-0". Since the containment flood level will exceed the floor elevation inside the secondary shield wall, the trench system is expected to transport water to the containment annulus.

Debris barriers have been installed in the loop "A" and loop "D" passageway entrances through the secondary shield wall, which are near the containment recirculation sumps. Debris barriers have also been installed in the portions of the drain trenches and other openings in the secondary shield wall that are near the recirculation sumps. A portion of the drain trench in the containment annulus region can be seen in Figure 3I-1, with a trench opening through the secondary shield wall just to the right of the loop "D" passageway. Using perforated plates with the same hole size as the sump strainers, the debris barriers are designed to restrict passage of debris while allowing water to pass through the barrier. The barriers prevent the flow of debris laden fluid directly to the sumps and force the fluid to take a "long path" through shield wall openings farther away from the sumps.

The remaining two open six foot wide passageways through the secondary shield wall will transport the ECCS break flow and CSS flow from inside the secondary shield to the containment annulus without restriction. In addition, the remaining trenches penetrating the secondary shield wall will also pass a significant quantity of water from inside the secondary shield wall to the containment annulus. Given these large passageways and large total trench length, large debris or mounds of debris would not create a choke point or hold-up point preventing the recirculation fluid from transporting to the sump. For added conservatism and ease of analysis, no water flow through the drain trenches is accounted for in the CDF model.

6. Ground Floor, Annulus (elevation 2000'-0"):

   The containment building emergency recirculation sumps are also located in this annular region between the secondary shield wall and the containment wall, as shown in Figure 3I-1. A six inch curb surrounds each sump pit, creating a six inch deep hold-up volume above the 2000'-0" floor elevation. As discussed above, the normal containment sumps receive water flow from the drain trenches and piping in this area. This represents an additional hold-up volume below the 2000'-0" floor elevation. Given the large flow passages in the annulus region, significant mounds of debris would not create a choke point preventing the recirculation fluid from transporting to the sump.

7. Reactor cavity basement and Incore Instrumentation Tunnel and Sump (elevation 1970'-6"):

   This area of evaluation encompasses the area under the reactor vessel in the reactor cavity as well as the incore instrumentation tunnel. Post-DBA water inventory flow to this area will come from the elevation 2001'-4" hatch north of the primary shield wall when the flood height exceeds 2001'-10" due to the protective 6" curb. In addition, flow to this area will also come from the permanent cavity seal ring access covers. This tunnel and area under the reactor cavity will retain water inventory during post-DBA recirculation mode operations. No potential choke points were identified in this area.

3I.2 Measures Taken to Mitigate Potential Choke Points

Administrative controls ensure the drains from the refueling cavity to lower containment are not obstructed during power operations.

3I.3 Water Holdup at Curbs or Debris Interceptors

As discussed above, a six inch curb surrounds each containment recirculation sump pit, creating a six inch deep hold-up volume above the 2000'-0" floor elevation. Debris barriers installed in the loop "A" and loop "D" passageway entrances through the secondary shield wall do not impact water hold-up since loop "B" and loop "C" passageways allow debris laden fluid to flow into the containment building annulus area and to the recirculation sumps.

3I.4 Potential Blockage of Reactor Cavity and Refueling Cavity Drains

The refueling pool floor (elevation 2009'-9") contains two 10 inch diameter drains that are open during power operations. There are debris exclusion devices (trash rack cages) installed during power operations over each of the 10 inch drains to prevent large pieces of debris from plugging the drains. The trash rack cages measure 33.5" x 33.5" x 15" with 5" openings. Figure 3I-1 shows a trash rack cage sitting on the refueling pool floor near the 10" drain, which has its blind flange installed for non-power operations (refueling preparations). The 10 inch drains go straight through the refueling cavity floor slab and discharge into the open area below; thus, the drain pipes themselves would not become plugged with debris. Administrative controls ensure the drains from the refueling cavity to lower containment are not obstructed during power operations.

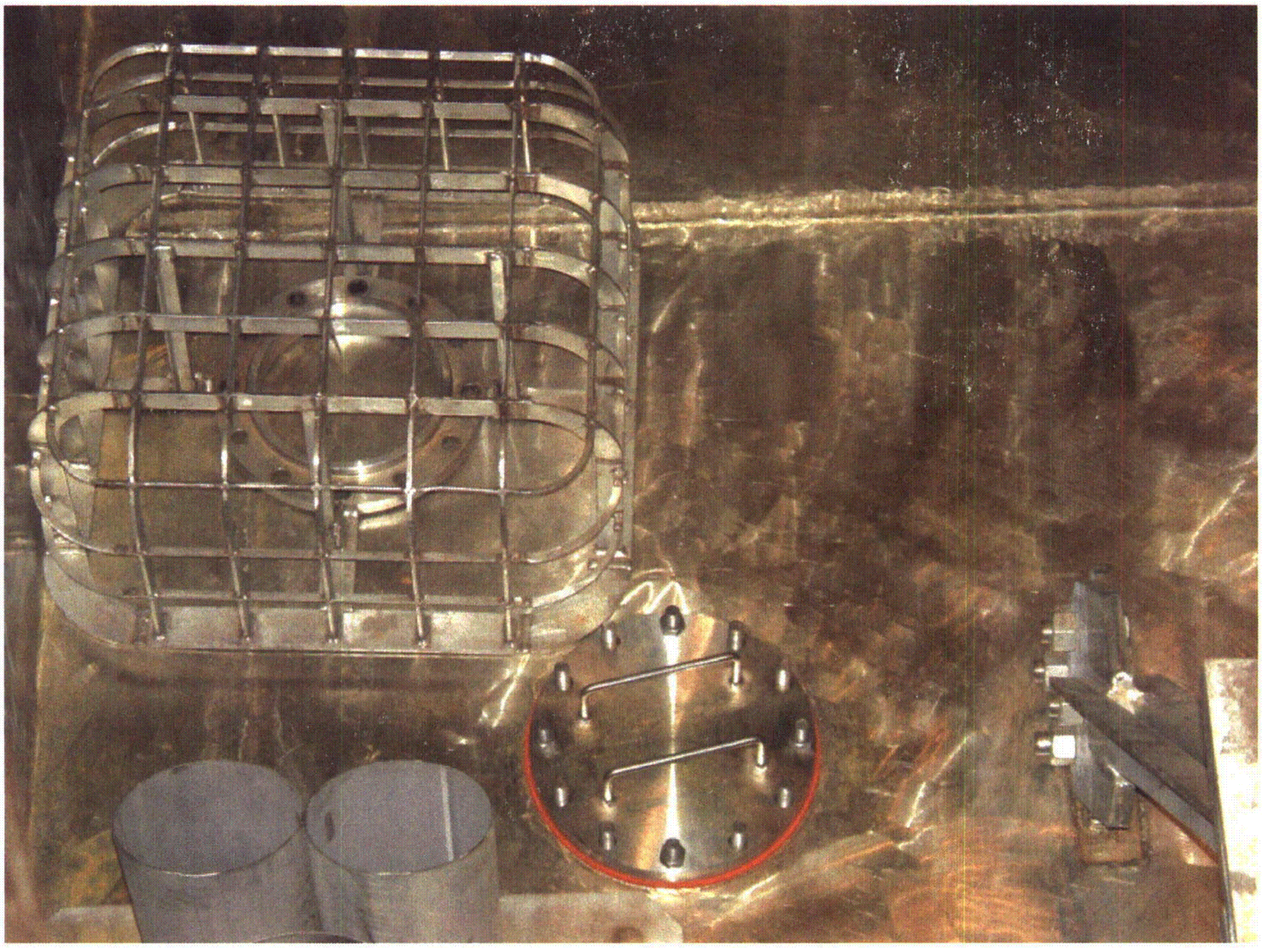


Figure 3l-1. Refueling Pool Trash Rack Cage

3m. <u>Downstream effects - Components and Systems</u>

The objective of the downstream effects, components and systems section is to evaluate the effects of debris carried downstream of the containment sump screen on the function of the ECCS and CSS in terms of potential wear of components and blockage of flow streams. Provide the information requested in GL 04-02 <u>Requested Information</u> Item 2(d)(v) and 2(d)(vi) regarding blockage, plugging, and wear at restrictions and close tolerance locations in the ECCS and CSS downstream of the sump.

1. *GL 2004-02 Requested Information Item 2(d)(v):*
   The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen, (e.g., a HPSI throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

2. *GL 2004-02 Requested Information Item 2(d)(vi):*
   Verification that the close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptable to plugging or excessive wear due to extended post-accident operation with debris-laden fluids.
3. *If NRC-approved methods were used (e.g., WCAP-16406-P with accompanying NRC SE), briefly summarize the application of the methods. Indicate where the approved methods were not used or exceptions were taken, and summarize the evaluation of those areas.*
4. *Provide a summary and conclusions of downstream evaluations.*
5. *Provide a summary of design or operational changes made as a result of downstream evaluations.*

3m.1, 3m.2, 3m.3, 3m.4, and 3m.5

As described in Section 2, WCNOC will update these responses following completion of the actions associated with the vendor testing.

3n. Downstream Effects - Fuel and Vessel

The objective of the downstream effects, fuel and vessel section is to evaluate the effects that debris carried downstream of the containment sump screen and into the reactor vessel has on core cooling.

1. *Show that the in-vessel effects evaluation is consistent with, or bounded by, the industry generic guidance (WCAP-16793), as modified by NRC staff comments on that document. Briefly summarize the application of the methods. Indicate where the WCAP methods were not used or exceptions were taken, and summarize the evaluation of those areas.*

As described in Section 2, WCNOC will update this response following completion of the actions associated with the vendor testing.

3o. Chemical Effects

The objective of the chemical effects section is to evaluate the effect that chemical precipitates have on head loss and core cooling.

1. *Provide a summary of evaluation results that show that chemical precipitates formed in the post-LOCA containment environment, either by themselves or combined with debris, do not deposit at the sump screen to the extent that an unacceptable head loss results, or deposit downstream of the sump screen to the extent that long-term core cooling is unacceptably impeded.*
2. *Content guidance for chemical effects is provided in Enclosure 3 to a letter from the NRC to NEI dated September 27, 2007 (ADAMS Accession No. ML0726007425).*

This section provides a portion of the information requested above. As described in Section 2, WCNOC will update this section following completion of the actions associated with the vendor testing.

3o.1 Summary of Chemical Effects Evaluation Results

Industry document WCAP-16530-NP (Reference 7) was issued to provide a consistent approach for plants to evaluate chemical effects in the post-LOCA containment environment. The results of this evaluation were intended to provide the type and amounts of chemical precipitates to be used as supporting information for testing of the replacement sump strainers. WCAP-16530-NP provided a spreadsheet using Microsoft Excel® spreadsheet software, which allowed the individual plants to enter their plant specific data. A plant chemical model was generated utilizing that data. In addition, industry document WCAP-16785-NP was issued which provided supplemental information to allow utilities to reduce conservatisms through the incorporation of plant specific refinements to the chemical model spreadsheet, originally provided in WCAP-16530-NP.

The WCAP-16530-NP Spreadsheet was utilized in determining the amount of chemical precipitates formed in a post-LOCA containment environment. The NRC safety evaluation (SE) for WCAP-16530-NP was used in the use of the spreadsheet. When the SE is incorporated into the WCAP and it is released as WCAP-16530-P-A, the spreadsheet use will be reviewed and changes made if necessary. None of the refinements suggested in WCAP-16785-NP were used in the WCGS evaluation. Four separate runs of the spreadsheet were performed, for four different LOCA scenarios. The four scenarios are:

- Break at loop "D" cross-over leg at the base of the steam generator (Case 1a), with two containment spray pumps running.
- Break at loop "D" cross-over leg at the base of the steam generator (Case 1a), with one containment spray pump running.
- Break at Reactor vessel cold leg loop "A" nozzle (Case 2).
- Small break LOCA (Case 3) in the Loop "D" alternate charging line.

The loop crossover LOCA was selected because it generated the largest amount of fiber (from NUKON™ insulation). The highest fiber generation was at the SG outlet. This LOCA was evaluated for both maximum spray and minimum spray. The reason for evaluating both scenarios was that maximum spray reduced containment temperature more rapidly. As shown in the results below, the hotter temperatures generated more chemical precipitates. The reactor vessel cold leg break was selected because that area contained Cerablanket® insulation, which is denser than NUKON™ insulation and it is made of aluminum silicate. Another insulation type (Min-K) is located in that LOCA area, but it was determined by testing to remain intact. Therefore, it was not included as an input to this chemical effects model. Only a small amount of NUKON™ insulation is affected by this break. The SBLOCA was selected because of the smaller water volume available for recirculation.

The WCGS containment spray system uses sodium hydroxide (NAOH) as a buffer solution. The NAOH is mixed with the spray water prior to being sprayed into the containment building. Therefore, the pH of the spray solution for the time when NAOH is being added is approximately 9.5. After all of the NAOH has been added and the spray pumps have been switched over to containment recirculation, the spray pH is the same as the sump pH. The spray pumps were conservatively assumed to run for four hours. This was based on the basis document for the safety injection termination procedure, which states the containment pressure requirement for stopping the pumps is reached in approximately two hours.

The only chemical precipitate generated from the four scenarios was sodium aluminum silicate ($NaAlSi_3O_8$). The following table shows the amount of precipitate generated and the containment material that contributed to the generated amount:

Table 3o-1. Contributions to Precipitates ($NaAlSi_3O_8$) by Material

Material	Loop 4 Crossover Max Spray Precipitates formed (kg)	Loop 4 Crossover Min Spray Precipitates formed (kg)	Reactor Cavity Precipitates formed (kg)	Small Break LOCA Precipitates formed (kg)
Metallic Aluminum submerged	3.06	2.69	5.49	0.69
Metallic Aluminum not-submerged	4.61	15.73	4.28	0.81
NUKON™	31.02	40.72	13.73	1.77
Aluminum Silicate (Cerablanket®)	0	0	11.20	0
Concrete	0	0	0	0.01
Total	38.69	59.14	34.7	3.28

The larger amount of precipitate generated from the minimum spray large break LOCA scenario is due to containment temperature remaining higher than the maximum spray scenario. The higher chemical precipitates amount generated in the minimum spray scenario was used during for the head loss testing.

3o.2 Content Guide for Chemical Effects Evaluation

The chemical effects evaluation process flow chart provided in the NRC guidance document has been modified, as shown in Figure 3o-1, to highlight the process approach taken for testing and evaluation. The remainder of the information provided in this section provides a portion of the information requested above. As described in Section 2, WCNOC will update this section following completion of the actions associated with the vendor testing.

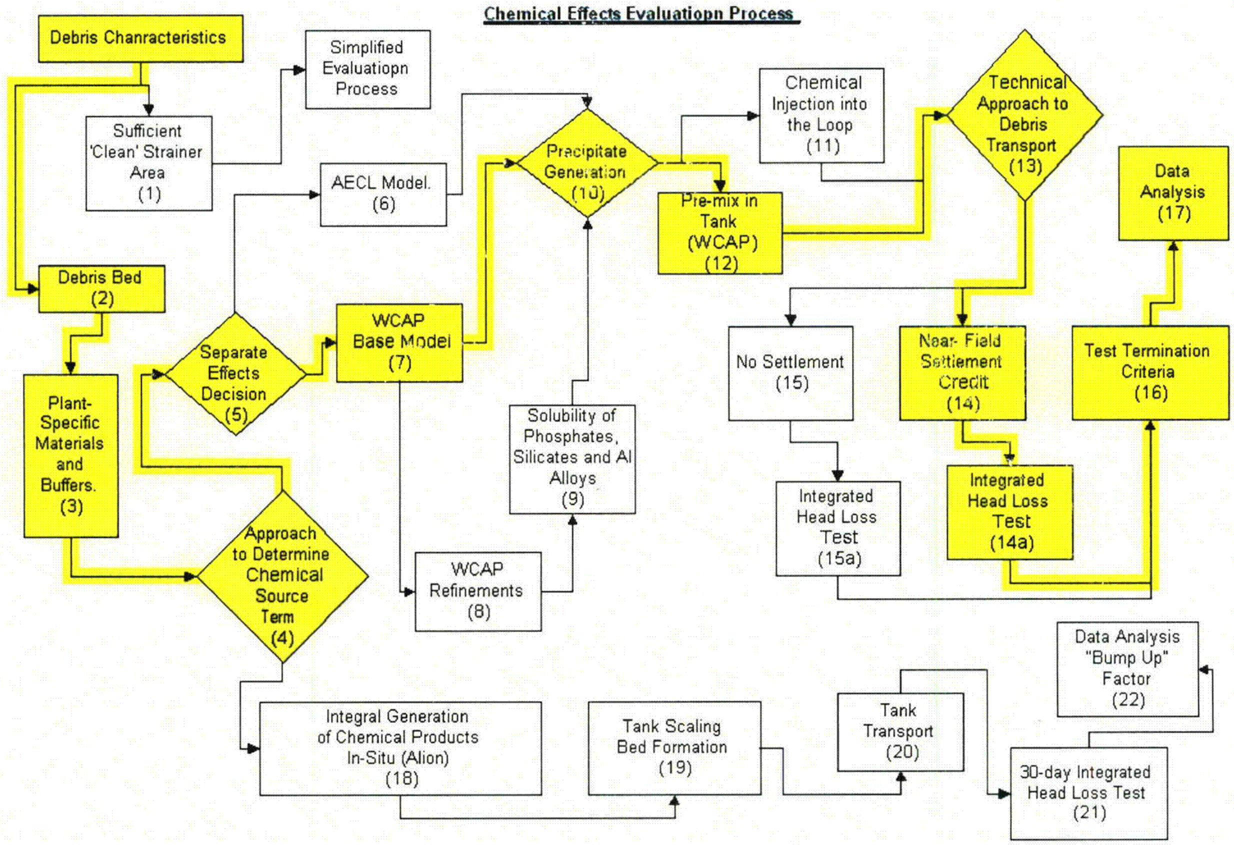


Figure 3o-1. Chemical Effects Evaluation Process Flow Chart

3o.2 Block 3 Plant-Specific Materials and Buffers

The WCGS containment spray system uses sodium hydroxide (NAOH) as a buffer solution. The NAOH is mixed with the spray water prior to being sprayed into Containment. Therefore, the pH of the spray solution for the time when NAOH is being added is approximately 9.5. After all of the NAOH has been added and the spray pumps have been switched over to containment recirculation, the spray pH is the same as the sump pH of 8.655. The spray pumps were conservatively assumed to run for four hours. This was based on the Basis Document for Operations SI Termination Emergency Procedure, which states the containment pressure requirement for stopping the pumps is reached in approximately two hours.

The containment atmosphere and sump temperature profiles following the LOCA events were derived from WCNOC safety analysis calculations. Since there was a substantial difference in how long it took the containment temperature to decrease between maximum safeguards actuation (i.e., two trains assumed to be operating at maximum acceptable operation) and minimum safeguards (i.e., one train of equipment assuming to be operating at minimum acceptable operation), the large break LOCA scenario for chemical precipitates generated was run for both the minimum and the maximum cases. The only chemical precipitate generated from the four scenarios was sodium aluminum silicate ($NaAlSi_3O_8$). Table 3o-1 shows the results.

As Table 3o-1 shows, metallic aluminum and NUKON™ are the major contributors in the Loop 4 crossover large break LOCA and the small break LOCA. Since in the reactor cavity break, there is only a small amount of NUKON™ insulation but some denser Cerablanket® insulation, aluminum and Cerablanket® are the contributors. Also, when comparing the minimum and maximum safeguards, higher temperature inside containment is a substantial contributor.

3o.2 Block 4 Approach to Determine Chemical Source Term

The strainer performance testing was conducted at the Alden Research Laboratory, Inc. (ALDEN) facility in Holden, Massachusetts and performed by ALDEN personnel.

3o.2 Block 7 WCAP Base Model

The WCAP-16530-NP (Reference 7) base model spreadsheet was used. None of the refinements suggested in WCAP-16785-NP were utilized. As discussed above, four different scenarios were evaluated. The only chemical precipitate generated from the four scenarios was sodium aluminum silicate ($NaAlSi_3O_8$). Table 3o-1 provides the spreadsheet results for the four scenarios:

3o.2 Block 10 Precipitate Generation Decision

Although sodium aluminum silicate ($NaAlSi_3O_8$) was the only major chemical precipitate shown to be formed using the chemical effects spreadsheet, aluminum oxyhydroxide (AlOOH) was used in place of it as a surrogate. The reason for this was because the production of $NaAlSi_3O_8$ is considered hazardous. The justification is from Section 7.3.2 of WCAP-16530-NP, which stated that the characteristics of $NaAlSi_3O_8$ are sufficiently similar to AlOOH, thus AlOOH may be used in lieu of $NaAlSi_3O_8$.

Chemical precipitate generation was performed using untreated city / potable water.

3o.2 Block 12 through Block 17

As described in Section 2, WCNOC will update these responses following completion of the actions associated with the vendor testing.

3p. Licensing Basis

The objective of the licensing basis section is to provide information regarding any changes to the plant licensing basis due to the sump evaluation or plant modifications. *Provide the information requested in GL 04-02 Requested Information Item 2(e) regarding changes to the plant licensing basis. The effective date for changes to the licensing basis should be specified. This date should correspond to that specified in the 10 CFR 50.59 evaluation for the change to the licensing basis.* *1. GL 2004-02 Requested Information Item 2(e):* *A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications made to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this GL. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included.*

The information in this section revises information previously provided by Wolf Creek Nuclear Operating Corporation (WCNOC) to Generic Letter 2004-02, Section 2(e) (Reference 2).

3p.1 Changes to the WCNOC Licensing Basis

WCNOC is in the process of implementing licensing basis changes to support overall resolution of GSI-191. These licensing basis changes include those associated with procedures and physical plant modifications, those associated with changes to Technical Specifications, and those associated with analyses and evaluations.

The plant licensing basis changes due to procedures and plant modifications were evaluated against the current licensing basis, as well as the future licensing basis that will be in effect following implementation of the actions in the Generic Letter 2004-02 corrective action package. Procedure and plant modifications that have been implemented are described in Section 2.

In Reference 18, WCNOC requested a license amendment to revise Technical Specification 3.5.2, "ECCS - Operating," to support replacement of the containment recirculation sumps inlet trash racks and screens with the Performance Contracting, Inc. (PCI) Sure-Flow® replacement strainers described in Section 3j above. The approved amendment (Reference 19) revises Surveillance Requirement 3.5.2.8 by replacing the phrase "trash racks and screens" with the word "strainers."

The analyses and evaluations supporting the Generic Letter 2004-02 corrective actions will become part of the plant licensing basis upon complete implementation of the Generic Letter 2004-02 corrective action package. As described in Section 2, vendor testing results are being assembled and when received, will be assessed for potential impact on analyses and evaluation performed to support Generic Letter 2004-02 corrective actions.

All changes to the current licensing basis will be described in the WCGS Updated Safety Analysis Report in accordance with the requirements of 10 CFR 50.71(e).

References:

1. NEI 04-07, Pressurized Water Reactor Sump Performance Evaluation Methodology, Revision 0, Nuclear Energy Institute, 1776 I Street N. W., Suite 400, Washington D.C., December 2004; Volume 1 – Pressurized Water Reactor Sump Performance Evaluation Methodology; Volume 2 – Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02.

2. Letter ET 05-0018, dated August 31, 2005, from T. J. Garrett, WCNOC, to USNRC

3. Letter WO 06-0028, dated May 31, 2006, from S. E. Hedges, WCNOC, to USNRC

4. Letter ET 07-0056, dated December 5, 2007, from T. J. Garrett, WCNOC, to USNRC

5. Letter dated December 27, 2007, from USNRC to R. A. Muench, WCNOC

6. WCAP-16406-P, Rev. 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191", Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, PA 15230-0355, August 2007, plus the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to WCAP-16406-P, Rev. 1.

7. WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191". Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, PA 15230-0355, February 2006, plus the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to WCAP-16530-NP.

8. WCAP-16568-P, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA-Qualified/Acceptable Coatings", Revision 0 dated June 2006.

9. Letter from Jon Cavallo, Corrosion Control Consultants and Labs Inc., to C. Feist, Luminant Power, dated September 20, 2007.

10. NUREG/CR-2791, Methodology for Evaluation of Insulation Debris Effects, dated September 1982.

11. Keeler and Long Report No. 06-0413, "Design Basis Accident Testing of Coating Samples from Unit 1 Containment, TXU Comanche Peak SES," April 13, 2006.

12. EPRI 1011753, Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings, September 2005.

13. NUREG/CR-6916, Hydraulic Transport of Coating Debris , December 2006.

14. EPRI Report No. 1014883, "Plant Support Engineering: Adhesion Testing of Nuclear Coating Service Level 1 Coatings," August 2007.

15. ANSI/ANS 58.2-1988, "American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," 1988.

16. Letter WO 03-0049, dated August 8, 2003, from B. T. McKinney, WCNOC, to USNRC

17. Letter WM 04-0050, dated November 5, 2004, from Richard A. Muench, WCNOC to USNRC

18. Letter WO 06-0023, dated June 2, 2006, from S. E. Hedges, WCNOC, to USNRC

19. Letter dated October 5, 2006, from USNRC to R. A. Muench, WCNOC

GL 2004-02 RAI Cross-Reference Table

The following table identifies each RAI request in Reference 1 and provides a reference to the applicable portion(s) of Attachment I or other appropriate information to address the request.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
	Plant Materials	
1	Identify the name and bounding quantity of each insulation material generated by a large-break loss-of-coolant accident (LBLOCA). Include the amount of these materials transported to the containment pool. State any assumptions used to provide this response.	Information provided in Sections 3b.4, 3e.2 and 3e.6 of Attachment I.
2	Identify the amounts (i.e., surface area) of the following materials that are;	Note: Zinc, copper and carbon steel were not used in the chemical effects evaluation.
2a	(a) submerged in the containment pool following a LOCA,	
2a1	- aluminum	25 ft^2
2a2	- zinc (from galvanized steel and from inorganic zinc coatings)	4131 ft^2
2a3	- copper	0
2a4	- carbon steel not coated	0
2a5	- uncoated concrete	0 – Prior to LOCA, after LOCA total of 960 ft^2 used as amount inside ZOI factored into chemical effects evaluation.
2b	(b) in the containment spray zone following a LOCA:	
2b1	- aluminum	890 ft^2
2b2	- zinc (from galvanized steel and from inorganic zinc coatings)	199,890 ft^2
2b3	- copper	162,051 ft^2
2b4	- carbon steel not coated	0
2b5	- uncoated concrete	See response to 2a5 above
2c	Compare the amounts of these materials in the submerged and spray zones at your plant relative to the scaled amounts of these materials used in the Nuclear Regulatory Commission (NRC) nuclear industry jointly-sponsored Integrated Chemical Effects Tests (ICET) (e.g., 5x the amount of uncoated carbon steel assumed for the ICETs).	Chemical effects information provided in 3o.1

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
2c1	- aluminum	Aluminum data provided in Table 3o-1.
2c2	- zinc (from galvanized steel and from inorganic zinc coatings)	ICET#1 had a test ratio of 8.0 for galvanized steel-hot dipped (.055 at WCGS) and 5% submerged (3% at WCGS). ICET#1 had a test ratio of 4.6 for zinc coating (.056 at WCGS) and 4% submergence (0.5% at WCGS)
2c3	- copper	ICET#1 had a test ratio of 6.0 and a submergence of 25% (No copper submerged at WCGS)
2c4	- carbon steel not coated	ICET#1 had a test ratio of 0.15 and 34% submerged (No uncoated carbon steel at WCGS)
2c5	- uncoated concrete	See response to 2a5 above
3	Identify the amount (surface area) and material (e.g., aluminum) for any scaffolding stored in containment. Indicate the amount, if any, that would be submerged in the containment pool following a LOCA. Clarify if scaffolding material was included in the response to Question 2.	200 ft^2 of aluminum is allotted for scaffolding, none is assumed to be submerged. This 200 ft^2 was included in the Question 2 response.
4	Provide the type and amount of any metallic paints or non-stainless steel insulation jacketing (not included in the response to Question 2) that would be either submerged or subjected to containment spray.	Metallic paints – None Non-stainless steel jacketing - None
	Containment Pool Chemistry	
5	Provide the expected containment pool pH during the emergency core cooling system (ECCS) recirculation mission time following a LOCA at the beginning of the fuel cycle and at the end of the fuel cycle. Identify any key assumptions.	Pool chemistry discussed in Sections 3o.1 and 3o.2.
6	For the ICET environment that is the most similar to your plant conditions, compare the expected containment pool conditions to the ICET conditions for the following items: boron concentration, buffering agent concentration, and pH. Identify any other significant differences between the ICET environment and the expected plant-specific environment.	Pool chemistry discussed in Sections 3o.1 and 3o.2.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
7	For a LBLOCA, provide the time until ECCS external recirculation initiation and the associated pool temperature and pool volume. Provide estimated pool temperature and pool volume 24 hours after a LBLOCA. Identify the assumptions used for these estimates.	Information is provided in Section 3g.1. Assumptions are provided in Sections 3g.7 through 3g.12
	Plant- Specific Chemical Effects	
8	Discuss your overall strategy to evaluate potential chemical effects including demonstrating that, with chemical effects considered, there is sufficient net positive suction head (NPSH) margin available during the ECCS mission time. Provide an estimated date with milestones for the completion of all chemical effects evaluations.	Later – pending vendor head loss testing results, as described in Section 2.
9	Identify, if applicable, any plans to remove certain materials from the containment building and/or to make a change from the existing chemicals that buffer containment pool pH following a LOCA.	No changes planned
10	If bench-top testing is being used to inform plant specific head loss testing, indicate how the bench-top test parameters (e.g., buffering agent concentrations, pH, materials, etc.) compare to your plant conditions. Describe your plans for addressing uncertainties related to head loss from chemical effects including, but not limited to, use of chemical surrogates, scaling of sample size and test durations. Discuss how it will be determined that allowances made for chemical effects are conservative.	Later – pending vendor head loss testing results, as described in Section 2.
	Plant Environment Specific	
11	Provide a detailed description of any testing that has been or will be performed as part of a plant-specific chemical effects assessment. Identify the vendor, if applicable, that will be performing the testing. Identify the environment (e.g., borated water at pH 9, deionized water, tap water) and test temperature for any plant-specific head loss or transport tests. Discuss how any differences between these test environments and your plant containment pool conditions could affect the behavior of chemical surrogates. Discuss the criteria that will be used to demonstrate that chemical surrogates produced for testing (e.g., head loss, flume) behave in a similar manner physically and chemically as in the ICET environment and plant containment pool environment.	Surrogate information provided in Section 3h.3. Other information will be provided later, pending vendor head loss testing results, as described in Section 2.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
12	For your plant-specific environment, provide the maximum projected head loss resulting from chemical effects (a) within the first day following a LOCA, and (b) during the entire ECCS recirculation mission time. If the response to this question will be based on testing that is either planned or in progress, provide an estimated date for providing this information to the NRC.	Later – pending vendor head loss testing results, as described in Section 2.
	ICET 1 and ICET 5 Plants	
13	Results from the ICET #1 environment and the ICET #5 environment showed chemical products appeared to form as the test solution cooled from the constant 140 °F test temperature. Discuss how these results are being considered in your evaluation of chemical effects and downstream effects.	Utilizing the Chemical Model provided in WCAP-16530-NP, a sensitivity analysis for a long term sump temperature of 100°F vice the 140 °F was performed. This evaluation showed a slight decrease in the amount of precipitates formed. Additional information will be provided later, pending vendor head loss testing results, as described in Section 2.
	Trisodium Phosphate (TSP Plants)	
	(Question 14 not applicable to WCNOC).	
15	Your Generic Letter (GL) 2004-02 response indicated that you were considering switching from the existing containment pool buffering agent to trisodium phosphate (TSP). Discuss whether these plans have changed given recent test results (IN 2005-26 and Supplement 1) that indicate formation of calcium phosphate can result in significant head loss across a debris bed. If you intend to switch to TSP, estimate the concentration of dissolved calcium that would exist in your containment pool from all containment sources (e.g., concrete and materials such as calcium silicate, Marinite™, mineral wool, kaylo) following a LBLOCA and discuss any ramifications related to the evaluation of chemical effects and downstream effects.	There are no plans to change from sodium hydroxide to TSP. Utilizing the Chemical Model provided in WCAP-16530-NP, a sensitivity analysis with TSP input into the model showed an increase in precipitates being formed. Additional information will be provided later, pending vendor head loss testing results, as described in Section 2.
	(Questions 16 through 24 not applicable to WCNOC).	

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
	Coatings	
	Generic - All Plants	
25	Describe how your coatings assessment was used to identify degraded qualified/acceptable coatings and determine the amount of debris that will result from these coatings. This should include how the assessment technique(s) demonstrates that qualified/acceptable coatings remain in compliance with plant licensing requirements for design basis accident (DBA) performance. If current examination techniques cannot demonstrate the coatings' ability to meet plant licensing requirements for DBA performance, licensees should describe an augmented testing and inspection program that provides assurance that the qualified/acceptable coatings continue to meet DBA performance requirements. Alternately, assume all containment coatings fail and describe the potential for this debris to transport to the sump.	Information presented in Section 3h.
	Plant Specific	
26	Provide test methodology and data used to support a zone of influence (ZOI) of 5.0 L/D. Provide justification regarding how the test conditions simulate or correlate to actual plant conditions and will ensure representative or conservative treatment in the amounts of coatings debris generated by the interaction of coatings and a two-phase jet. Identify all instances where the testing or specimens used deviate from actual plant conditions (i.e., irradiation of actual coatings vice samples, aging differences, etc.). Provide justification regarding how these deviations are accounted for with the test demonstrating the proposed ZOI.	Information presented in Section 3h.5
	(Questions 27 through 29 not applicable to WCNOC).	

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
30	The NRC staff's safety evaluation (SE) addresses two distinct scenarios for formation of a fiber bed on the sump screen surface. For a thin bed case, the SE states that all coatings debris should be treated as particulate and assumes 100% transport to the sump screen. For the case in which no thin bed is formed, the staff's SE states that the coatings debris should be sized based on plant-specific analyses for debris generated from within the ZOI and from outside the ZOI, or that a default chip size equivalent to the area of the sump screen openings should be used (Section 3.4.3.6). Describe how your coatings debris characteristics are modeled to account for your plant-specific fiber bed (i.e. thin bed or no thin bed). If your analysis considers both a thin bed and a non-thin bed case, discuss the coatings' debris characteristics assumed for each case. If your analysis deviates from the coatings' debris characteristics described in the staff-approved methodology, provide justification to support your assumptions.	Later – pending vendor head loss testing results, as described in Section 2.
31	How will your containment cleanliness and foreign material exclusion (FME) programs assure that latent debris in containment will be controlled and monitored to be maintained below the amounts and characterization assumed in the ECCS strainer design? In particular, what is planned for areas/components that are normally inaccessible or not normally cleaned (containment crane rails, cable trays, main steam/feedwater piping, tops of steam generators, etc.)?	A Containment Latent Debris Assessment Program has been established, which will survey Containment per a Latent Debris Sample Plan. This sampling plan includes; 1) Floors and walls, 2) Tops of cable trays, 3) Tops of ductwork, 4) Tops of major equipment, 5) Tops of valve operators, and Top surfaces of major piping. The Containment Entry and Material Control Procedure specifies both general cleaning and targeted cleaning. Targeted cleaning involves areas that are not easily accessible and will be planned and scheduled via the work controls process. The areas selected for cleaning will consider the results of the latent debris survey. Also refer to Section 3d.1.
32	Will latent debris sampling become an ongoing program?	Information provided in Section 3d.1

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
33	Was/will "leak before break" be used to analyze the potential jet impingement loads on the new ECCS sump screen?	No, "leak before break" was not used in the evaluation of potential jet impingement loads on the new ECCS sump screen.
34	You indicated that you would be evaluating downstream effects in accordance with WCAP 16406-P. The NRC is currently involved in discussions with the Westinghouse Owner's Group (WOG) to address questions/concerns regarding this WCAP on a generic basis, and some of these discussions may resolve issues related to your particular station. The following issues have the potential for generic resolution; however, if a generic resolution cannot be obtained, plant specific resolution will be required. As such, formal RAIs will not be issued on these topics at this time, but may be needed in the future. It is expected that your final evaluation response will specifically address those portions of the WCAP used, their applicability, and exceptions taken to the WCAP. For your information, topics under ongoing discussion include: a. Wear rates of pump-wetted materials and the effect of wear on component operation b. Settling of debris in low flow areas downstream of the strainer or credit for filtering leading to a change in fluid composition c. Volume of debris injected into the reactor vessel and core region d. Debris types and properties e. Contribution of in-vessel velocity profile to the formation of a debris bed or clog f. Fluid and metal component temperature impact g. Gravitational and temperature gradients h. Debris and boron precipitation effects i. ECCS injection paths j. Core bypass design features k. Radiation and chemical considerations l. Debris adhesion to solid surfaces m. Thermodynamic properties of coolant	Later – pending vendor head loss testing results, as described in Section 2.
35	Your response to GL 2004-02 question (d) (viii) indicated that an active strainer design will not be used, but does not mention any consideration of any other active approaches (i.e., backflushing). Was an active approach considered as a potential strategy or backup for addressing any issues?	No, passive only. No other active approaches are being considered.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
36	You stated that a containment walkdown consistent with draft Nuclear Energy Institute (NEI) 02-XX was completed in April, 2002. Please discuss any recommendations in NEI 02-01 which may not have been included in the draft document used by Wolf Creek. Does the licensee intend to perform a confirmatory walkdown consistent with NEI 02-01 as part of their screen design process?	NEI 02-01 provided clarifications in most of the areas from NEI-02-XX. An Appendix A, "Application Experience" was added based on having the walkdown completed at several plants. The recommendations added from previous drafts are; 1) 5.2.2.5 documentation was revised to document locations of "DBA qualified" or "Acceptable" coatings and unqualified or non-qualified coatings. (A separate Coatings Assessment was performed under Refuel XIV to confirm the location and amounts of said coatings.), and 2) 5.2.4 Additional consideration was added to provide a better understanding of other considerations that may be needed to review. The walkdown conducted under NEI-02-XX considered these areas. Since WCGS was also one of the plants that provided input to revise the document based on their walkdown and new recommendations were either performed under a separate assessment or included in the original assessment, no plans exist to perform a new walkdown using NEI-02-01.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
37	You stated that for materials which have no experimentally-determined ZOI, a conservative assumption was made and the lowest available destruction pressure and ZOI were adopted (28.6 D). Please provide a listing of the materials for which this ZOI was applied and the technical reasoning for concluding this is conservative.	This information is presented in Sections 3b.4 and 3c.1. In addition, the four category size distribution has been reduced to a three category size distribution. The new categories are for NUKON™: small fines, large pieces and intact blankets. Section 3e.6 provides the data for fiber erosion.
38	The September 2005 response to GL 2004-02 stated that a four-category size distribution was used to characterize Nukon and thermal wrap insulation debris. However, numerical values were not provided to specify what fraction of debris was placed in each category. Furthermore, as the four-category distribution is a refinement to the baseline methodology, the NRC staff expects that the effects of fibrous debris erosion be considered explicitly. Numerical values were similarly not provided to identify what fraction of debris is considered to erode during the accident. The staff requests that the licensee provide numerical values to specify what fraction of debris is grouped into each of the four categories of fibrous debris, and specify what fraction of debris is assumed to erode.	This information is presented in Sections 3b.4 and 3c.1. In addition, the four category size distribution has been reduced to a three category size distribution. The new categories are for NUKON™: small fines, large pieces and intact blankets.
39	Has debris settling upstream of the sump strainer (i.e., the near-field effect) been credited or will it be credited in testing used to support the sizing or analytical design basis of the proposed replacement strainers? In the case that settling was credited for either of these purposes, estimate the fraction of debris that settled and describe the analyses that were performed to correlate the scaled flow conditions and any surrogate debris in the test flume with the actual flow conditions and debris types in the plant's containment pool.	This information is presented in Section 3f.12.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
40	Are there any vents or other penetrations through the strainer control surfaces which connect the volume internal to the strainer to the containment atmosphere above the containment minimum water level? In this case, dependent upon the containment pool height and strainer and sump geometries, the presence of the vent line or penetration could prevent a water seal over the entire strainer surface from ever forming; or else this seal could be lost once the head loss across the debris bed exceeds a certain criterion, such as the submergence depth of the vent line or penetration. According to Appendix A to Regulatory Guide 1.82, Revision 3, without a water seal across the entire strainer surface, the strainer should not be considered to be "fully submerged." Therefore, the NRC staff requests that, if applicable, the licensee explain what sump strainer failure criteria are being applied for the "vented sump" scenario described above.	NA – The strainers are completely submerged, except for a small break LOCA scenario. There are no vents.
41	What is the basis for concluding that the refueling cavity drain(s) would not become blocked with debris? What are the potential types and characteristics of debris that could reach these drains? In particular, could large pieces of debris be blown into the upper containment by pipe breaks occurring in the lower containment, and subsequently drop into the cavity? In the case that large pieces of debris could reach the cavity, are trash racks or interceptors present to prevent drain blockage? In the case that partial/total blockage of the drains might occur, do water hold-up calculations used in the computation of NPSH margin account for the lost or held-up water resulting from debris blockage?	The installed trash racks are discussed in Sections 3l.1 and 3l.4. Also, from the debris transport calculation,: NUKON™; Small Fines 25%, unjacketed large pieces 0%, jacketed large pieces 0%, RMI: small pieces 25%, unjacketed large pieces 0%, Thermo-Lag™; fines 25%, coatings inside ZOI; 25%, coatings outside ZOI; 0%. Drain blockage is not expected to occur. However, 155 ft^3 of hold-up volume is calculated for the refueling pool, due to the drain flanges setting above the floor surface.

No.	GL 2004-02 2/9/06 RAI Questions	GL 2004-02 2/9/06 RAI Responses
42	What is the minimum strainer submergence during the postulated LOCA? At the time that the re-circulation starts, most of the strainer surface is expected to be clean, and the strainer surface close to the pump suction line may experience higher fluid flow than the rest of the strainer. Has any analysis been done to evaluate the possibility of vortex formation close to the pump suction line and possible air ingestion into the ECCS pumps? In addition, has any analysis or test been performed to evaluate the possible accumulation of buoyant debris on top of the strainer, which may cause the formation of an air flow path directly through the strainer surface and reduce the effectiveness of the strainer?	Information on minimum strainer submergence is provided in Section 3f.2. Information on uniform flow rate associated with PCI strainers is provided in Section 3j.1. Information on vortexing is provided in Section 3f.3.
43	The September 2005 GL response noted that the licensee determined the fraction of debris blown into upper containment based on the volumes of upper and lower containments. Please explain how you determined this fraction and the basis.	As discussed in Section 3e.1, using computer aided drafting (CAD) software, the volumes of Containment were estimated to be 2,064,000 ft^3 for upper Containment (including the refueling canal and areas above the operating deck); and 512,000 ft^3 for lower Containment (including the area inside the primary shield wall and the area outside the primary shield wall).
44	The September 2005 GL response noted that the licensee determined the quantity of debris that could experience erosion due to the break flow or spray flow. Please explain how you determined this quantity and the basis.	Information is provided in Section 3e.2.

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation in this document. Any other statements in this letter are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Richard Flannigan, Manager Regulatory Affairs at Wolf Creek Generating Station, (620) 364-4117.

Regulatory Commitment	Due Date
Descriptions of the results of the testing and the implementation of calculations, evaluations and other changes described in Letter ET 07-0056, dated December 5, 2007, will be submitted to the NRC as an update to this supplemental response to GL 2004-02.	June 30, 2008
When the WCAP-16530-NP NRC safety evaluation is incorporated into the WCAP and it is released as WCAP-16530-P-A, the use of the WCAP-16530-NP spreadsheet will be reviewed and changes made if necessary.	Following issuance of WCAP-16530-P-A
Note: Even though the responses in Attachments I and II of this letter supersede the information provided in previous WCNOC responses to Generic Letter 2004-02, this letter does not change or otherwise impact regulatory commitments contained in WCNOC's previous letters responding to GL 2004-02, listed below. 1. Letter ET 05-0018, dated August 31, 2005, from T. J. Garrett, WCNOC, to USNRC, "Wolf Creek Nuclear Operating Corporation Response to Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" 2. Letter WO 06-0028, dated May 31, 2006, from S. E. Hedges, WCNOC, to USNRC, "Wolf Creek Nuclear Operating Corporation Update Response to Requested Information Part 2 of Generic Letter 2004-02: 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors'" 3. Letter ET 07-0056, dated December 5, 2007, from T. J. Garrett, WCNOC, to USNRC, "Request for Extension of Completion Date for Corrective Actions Associated with NRC Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors'"	