Contents of Attachment 1 should be withheld from public disclosure in accordance with the requirements of 10CFR2.390.

Nuclear Operating Company

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

May 11, 2016
NOC-AE-16003366
10 CFR 50.12
10 CFR 50.90
10 CFR 2.390

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
First Set of Responses to April 11, 2016 Requests for Additional Information
Regarding STP Risk-Informed GSI-191 Licensing Application
(TAC NOs MF2400 and MF2401)

References:

1. Letter, G. T. Powell, STPNOC, to NRC Document Control Desk, "Supplement 2 to STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02", August 20, 2015, NOC-AE-15003241, ML15246A126
2. Letter, Lisa Regner, NRC, to Dennis Koehl, STPNOC, "South Texas Project, Units 1 and 2- Request for Additional Information Related to Request for Exemptions and License Amendment for Use of a Risk-Informed Approach to Resolve the Issue of Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors", April 11, 2016, ML16082A507

Reference 2 transmitted RAIs on STPNOC's application in Reference 1 and divided the RAIs into 3 sets to be responded to in 30-day intervals (end of April, May, and June 2016). This submittal responds to the first set of RAIs with the exception of Follow-up RAI 34, which was deferred to the second set.

The response to RAI 18 in Attachment 1 contains information proprietary to Alion Science and Technology. A non-proprietary version is also provided and the affidavit for withholding from public disclosure is included as Attachment 7.

There are no commitments in this submittal.

If there are any questions, please contact Mr. Wayne Harrison at 361-972-8774.

A001
NRR

STI 34306883

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

Executed on: May 11, 2016

James Connolly

James Connolly
Site Vice President

awh

Attachments:

1. Response to Follow-up RAIs 18, 38, and 44
2. Response to SSIB-3-1 through 3
3. Response to DORL-3-1,
4. Response to ENPB-3-1 through 3,
5. Response to ESGB-3-1,
6. Response to SNPB-3-1, -3, -4, -5, -8, -11, -12, -13, -14, -16, -19
7. Affidavit of Withholding for the Response to RAI 18
8. Definitions and Acronyms

cc:

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Follow-up Questions

Debris Transport

Follow-up RAI 18 – NON-PROPRIETARY VERSION

In the December 23, 2009, RAIs, the NRC staff asked for a justification for an erosion value of 10%. Since the RAI was written, the NRC accepted 10%, but STP uses 7% in their RoverD evaluation. Please justify the use of 7% as an erosion fraction for low density fiber glass (LDFG) in the pool for the deterministic portion of the evaluation. The Alion test report concluded that 10% is a conservative erosion value for plants that can show that the test is applicable to their plant conditions. NRC acceptance of the Alion erosion report states that values less than 10% should not be used. There is no basis for acceptance of the 7% erosion value.

STP Response

The staff comments highlighting the use of the 7% erosion fraction for LDFG are understood to have come from documentation asserting that the use of a 10% erosion fraction was appropriate [1]. However, Reference 1 was intended to demonstrate the applicability of the embedded test data to plant conditions at STP although a lower bounding erosion limit resulted from testing. Furthermore, the fiber erosion testing that was performed [2] drew the conclusion that a value lower than 10% could in fact be used to describe fiber erosion during recirculation. A 7% erosion fraction bounds the average erosion of small pieces from testing [2] of [ ] in high and low velocity conditions, which are bounding of postulated LOCA conditions at STP. A summary of Alion fiber erosion testing and STP transport analysis conditions are provided in the bulleted list below to support this discussion.

- [ ]
- Alion transport calculation [3] utilized tumbling velocities of 0.12 and 0.37 ft/s for small and large pieces of fibrous debris respectively taken from NUREG/CR-6772 [4] and NUREG/CR-6808 [5] respectively.

From the parameters provided in the bulleted list above, the maximum high velocity erosion testing target velocity of [ ] bounds the maximum tumbling velocity of 0.37 ft/s used in STP transport analysis for large fibrous pieces and provided in NUREG/CR-6808 [5] for 6 by 6 by 2-inch tested samples of Nukon. Pieces of small and large fiber in the STP analysis are only assumed to settle out in regions with lower velocity than the large fibrous piece settling velocity of 0.37 ft/s.

References:

[1] ALION-REP-STP-8511-01 Rev. 0, "South Texas Fiberglass Debris Erosion Testing Report," October, 2012.
[2] ALION-REP-ALION-I006-04, Rev. 1, "Erosion Testing of Small Pieces of Low Density Fiberglass – Test Report," November, 2011.
[3] ALION-CAL-STP-8511-08 Rev. 3, "Risk-Informed GSI-191 Debris Transport Calculation," June, 2014.
[4] NUREG/CR-6772, "GSI-191: Separate-Effects Characterization of Debris Transport in Water," August, 2002.
[5] NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core cooling Sump Performance," February 2003.

Attachment 2

Response to SSIB-3-1 through 3

SSIB-3-1

Nuclear Energy Institute (NEI) 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004 (ADAMS Accession No. ML050550138), recommends treating labels and tags either as intact and transportable to the strainer using 75 percent of the total area, or as 100 percent fine fiber if they may not remain intact. Alternately, it can be shown that they will remain intact under accident conditions and not transport to the strainer. Section 3b of the licensee's December 11, 2008, submittal (ADAMS Accession No. ML083520326), states that the miscellaneous debris term is bounded by 100 square feet. Section 3d states that the 100 square feet was implemented in the debris generation and transport analysis. Section 3f states that transport testing was completed that determined that the miscellaneous debris objects would not transport to the strainer so they were not included in the test. Please explain the effect that miscellaneous debris would have on the RoverD evaluation.

STP Response

Walkdowns of the plant were performed to identify miscellaneous debris sources. Removable sources that were identified, including all unqualified tags and labels, were removed from the plant leaving only materials which are impervious to erosive forces. These materials include qualified plastic danger tags, and plastic ty-wraps. Therefore it is reasonable to assume that all miscellaneous plastic debris remains intact throughout recirculation after a LOCA.

Because the tag debris was shown not to transport in the February 2008 [1] head loss test, and because the makeup of the tags is impervious to erosion, this debris does not impact the RoverD evaluation as it never reaches the strainer. Tag debris could also act as a barrier to transport for other debris types during recirculation, though credit for this is not taken.

Reference:

1. AREVA. "South Texas Project Test Report for ECCS Strainer Performance Testing - February 2008." 66-9074541 R0, 2008.

SSIB-3-2

Based on NRC staff's review of the RoverD submittal, it appears that large pieces of debris do not reach the sump pool since they are eroded at 1 percent, as would be expected for debris held up on gratings. Please confirm that large pieces of debris do not reach the pool, and provide details on what prevents any large pieces from falling into the pool.

STP Response:

The evaluation described below confirms that a large piece LDFG in the limiting debris transport break location used in the RoverD submittal will be 100% held up (aside from erosion) in the steam generator compartment and 0% percent transport to the pool. A LBLOCA situated within the steam generator compartment produced the highest debris total transport fractions while still representing all but a small number of low debris generation breaks, (that is, those in the compartment below the generator compartment). These transport fractions, representing transport from the steam generator compartment, were applied to all breaks as a reasonably conservative assumption.

The steam generator compartment LBLOCA transport fraction for large pieces of LDFG was zero [1]. This is due to the tortuous path debris must traverse in order to reach the containment pool. The floor of the steam generator compartment is shown in Figure 1 and Figure 2. The only potential downward exits for fluid are through the 8 grated floor panels indexed in Figure 1.

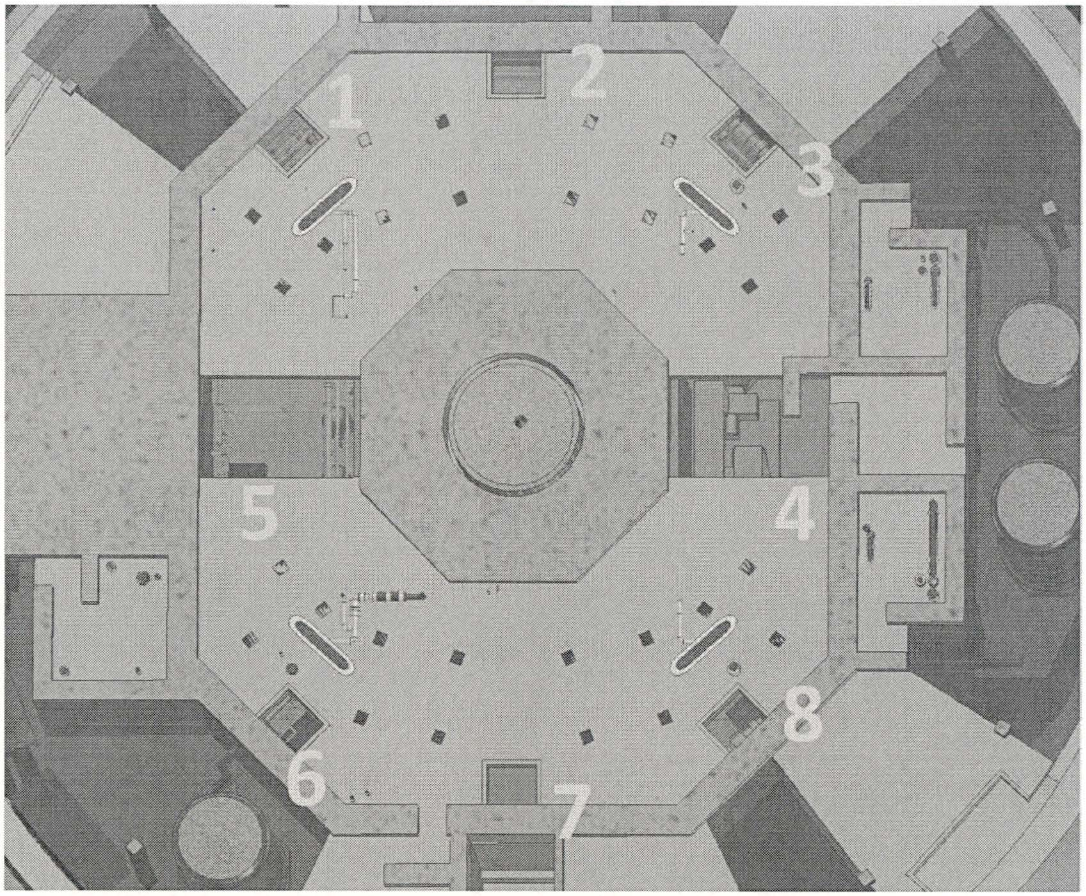

Figure 1: Upper Steam Generator Compartment Fluid Exit Paths

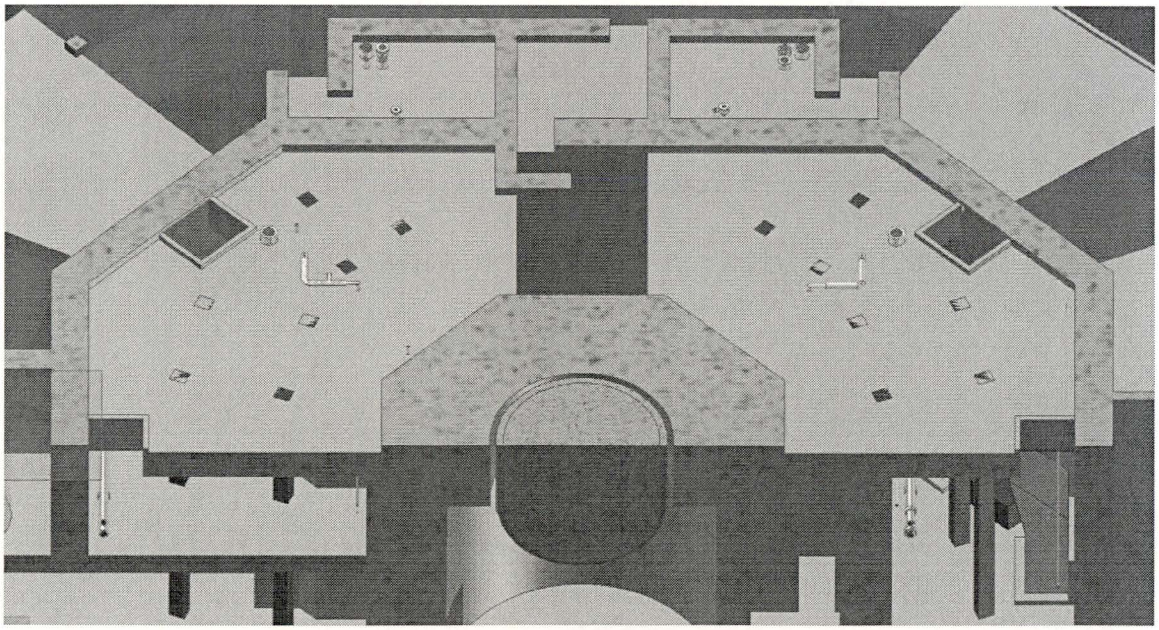

Figure 2: Upper Steam Generator Compartment Exit Flow Paths - Cross-sectional View

Additionally Figure 3 has been provided to show the vertical constraints on debris caused by the SG compartment walls.

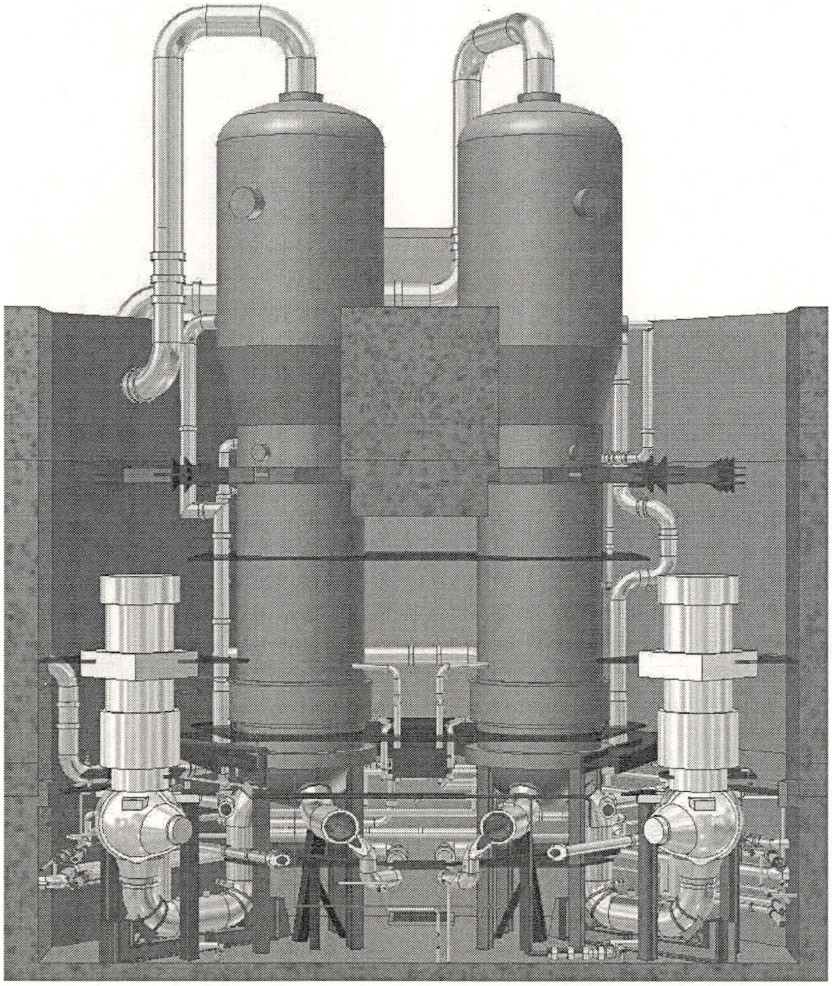

Figure 3: Upper Steam Generator Compartment - Vertical Cross-Sectional View

Large piece LDFG was conservatively assumed to be the smallest possible size of the category: 6" pieces. These pieces are of a size which is incapable of passing through a grating like those which cover the downward fluid exit paths of the compartment. The Drywell Debris Transport study [2] classified these fiber debris sizes in Table 1 where large piece LDFG present at STP is comparable to the DDTS classification of the same name. Due to this nature of the plant construction and large piece debris characteristics it is reasonable and appropriate to conclude that all large piece LDFG are trapped within the SG compartment on concrete and gratings, and fail to transport to the containment pool.

Table 1: Drywell Debris Transport Study - Table 3-7 - LDFG Size Classification

Debris classification	Relative size	Description
Large	> hand size	Large pieces were too large to pass through a grating and therefore were all located either upstream of the continuous grating or behind the target mount.
Medium	> grating cell but < hand size	Medium pieces sometimes were forced through a grating, although they were generally larger than a grating cell.
Small	< grating cell	Small pieces would generally pass through a grating cell unless the piece was to inertially impact on a grating bar. Small debris included fine particles such as individual fibers that could also pass through the catch screen of the exhaust flow.

References:

[1] ALION-CAL-STP-8511-08, Revision 3 – Risk-Informed GSI-191 Debris Transport Calculation. June 2014.

[2] NUREG/CR-6369, Volume 2, Drywell Debris Transport Study: Experimental Work. September 1999.

SSIB-3-3

The NRC staff needs additional information to verify the RoverD computations. For all debris types and sizes, please provide a summary table of the fractions of:

a) all debris transported to the recirculation pool and the strainer,

b) the debris retained in structures, and

c) the debris settled in the pool.

<u>STP Response</u>:

The limiting case applied to the RoverD analysis was that of the Steam Generator Compartment Case. Debris transport fractions were extracted from the STP transport calculation for that case [1] and are presented in Tables 1 and 2.

Table 1: Transport Fraction Totals: Retained on Structures and To Recirculation Pool

Debris Type	Debris Size	Debris Transport Fraction (Retained in Structures)	Debris Transport Fraction (To Recirculation Pool)
LDFG (NUKON & TempMat)	Fines	0.00%	100.00%
	Small Pieces (<6")	36.48%/36.12% *	0.36% * /63.52% ** /63.88% ***
	Large Pieces (>6")	100%/99% *	1% * /1% ***
	Intact Pieces	100.00%	0.00%
Microtherm	Fines	0.00%	100.00%
Qualified Coatings Fines	Fines	0.00%	100.00%
Unqualified Coatings Fines	Fines	0.00%	100.00%
Crud	Fines	0.00%	100.00%
Dirt/Dust	Fines	0.00%	100.00%
Latent Fiber	Fines	0.00%	100.00%

*Pre-Erosion/Post-Erosion

* As Fines
** As Smalls
*** Total Fraction

The two transport fraction values in each row of Table 1 add to unity. Where present the Post-Erosion and Total Fraction values must be taken. This reflects that all of the debris either is held up on structures above the pool or eventually transports to the pool. The small and large piece debris sizes of LDFG that were held up on structures were subject to erosion by containment sprays and produced fines in the process. This percentage is shown as a mass percentage of the total respective source debris size, i.e. 0.36% of the total small piece LDFG debris erodes to fines after being held up on structures, however this debris still reaches the pool and the table is constructed to indicate that.

It is important to note that RoverD computations did not transport any of the small size LDFG despite the “To Strainer” transport fraction based on observations from the July 2008 STP strainer flume test [2]. The sample of small piece LDFG which arrived at the recirculation pool in regions where transport was initially deemed appropriate, valued at 39.69%, was instead treated as settled out of recirculation pool. In this scenario the debris may be subject to erosion however the computations did not account for these additionally generated fines.

Table 2: Transport Fraction Totals: To Inactive Cavities, Settled Out of Pool, and To Strainers

Debris Type	Debris Size	Debris Transport Fraction (To Inactive Pool Cavities)	Debris Transport Fraction (Settled Out of Recirculation Pool)	Debris Transport Fraction (To Strainers)
LDFG (NUKON & TempMat)	Fines	1.50%	0.00%	98.50%
	Small Pieces (<6")	0.0%	23.83%/22.16% *	2.03% * /39.69% ** /41.72% ***
	Large Pieces (>6")	0.00%	0.00%	1% *
	Intact Pieces	0.00%	0.00%	0.00%
Microtherm	Fines	1.50%	0.00%	98.50%
Qualified Coatings Fines	Fines	1.50%	0.00%	98.50%
Unqualified Coatings Fines	Fines	0.00%	0.00%	100.00%
Crud	Fines	1.50%	0.00%	98.50%
Dirt/Dust	Fines	5.00%	0.00%	95.00%
Latent Fiber	Fines	5.00%	0.00%	95.00%

*Pre-Erosion/Post-Erosion

* As Fines
** As Smalls
*** Total Fraction

Marinite debris was also treated as 100% transported in RoverD calculations. This debris type was removed from the plant but since it was used in the 2008 head loss testing it is included as part of this analysis for supportive particulate margin calculations to account for deficiencies in other debris amounts.

The qualified coatings fines consists of all debris types associated with qualified coatings, i.e. epoxy, inorganic zinc, polyamide primer. While the unqualified coatings fines consisted of those of unqualified systems, i.e. inorganic zinc, alkyds, baked enamel, epoxy. Transport fractions for miscellaneous debris such as equipment labels, tags, plastic signs and ty-wraps are not provided as this debris was assumed to arrive at containment floor but not transport to the strainer. The non-transportability of miscellaneous debris at STP was based on the February 2008 STP strainer testing [3] where miscellaneous debris was observed not to transport when exposed to STP bounding fluid flow conditions.

While not specifically requested, Table 2 includes the transport fraction due to inactive pool cavities in addition to fractions that settle out of recirculation and reach the strainers. The three transport fraction values in each row of Table 2 add to the “To Recirculation Pool” transport fraction. Where present the Post-Erosion and Total Fraction values must be taken. This reflects that all debris that reaches the recirculation pool either ends up in an inactive cavity, settling out of recirculation or at the strainers.

References:

[1] ALION-CAL-STP-8511-08, Revision 3 – Risk-Informed GSI-191 Debris Transport Calculation. June 2014.
[2] AREVA NP Document No. 66-9074541-000 – South Texas Project Test Report for ECCS Strainer Performance Testing, February 2008.
[3] AREVA NP Document No. 66-9088089-000 – South Texas Project Test Report for ECCS Strainer Testing, August 2008.

Attachment 3

Response to DORL-3-1

DORL-3-1

Please provide a specific list of all licensing basis changes, in the application, for which you are requesting NRC review and approval via Section 50. 90 of Title 10 of the *Code of Federal Regulations* (10 CFR).

STP Response:

Attachment 3 to the August 20, 2015 supplement (ML15246A126 – cover letter, A129 - attachment) identifies the changes that require NRC approval per 10CFR50.90. The list below includes those items. The list provides additional detail on the basis for the need for NRC approval.

1. A change in methodology is proposed to allow risk-informed methods to demonstrate satisfactory ECCS sump/strainer performance for certain large breaks instead of the deterministic methods and assumptions described in the current licensing basis. Application of the risk-informed methodology also requires exemption to 10CFR50.46(a)(1) "other properties" and GDC 35, 38 & 41 as described in Attachment 2 to the August 20, 2015 supplement (ML15246A129) and STPNOC letter dated April 13, 2016 (ML16111B204) revising the exemption to 10CFR50.46. NRC approval of this methodology change is required by 10CFR50.59(c)(1)(vii) and (viii). Paragraph (vii) applies because the risk-informed method assumes that breaks with debris not bounded by testing result in core damage; i.e., fission product barrier exceeds a design basis limit, and Paragraph (viii) applies because the risk-informed methodology used to show the risk of core damage and large early release is acceptably small replaces the deterministic method used in the current licensing basis.

   This methodology change includes the use of RELAP5-3D as a screening evaluation for in-core effects of debris from the deterministic scope of breaks. The RoverD risk evaluation is based on all failures being at the strainer (i.e., no in-core risk contribution from smaller breaks). RELAP5-3D is applied to the deterministic scope of breaks that are bounded by the plant-specific testing to confirm there are no smaller (i.e., higher frequency) breaks that cause failure due to in-core effects. The analysis uses the 800°F LTCC PCT based on the WCAP 16793 methodology as its success criteria.

2. The proposed change in methodology for the use of RELAP5-3D is also credited to show satisfactory long-term cooling for the deterministic scope of large HLBs and small CLBs. RELAP5-3D is not required for deterministic scope of large CLBs because the RoverD evaluation shows that there are no debris effects for these breaks. The thermal hydraulic analysis is described in the RoverD discussion of Attachment 1-3 to the August 20, 2015 LAR supplement. The current licensing basis is described in UFSAR Chapter 15.6. The STP current licensing basis thermal-hydraulic analysis is applied only through the reflood

stage to show that the core cooling meets the 10CFR50.46 acceptance criteria such that there are no conditions that would prevent long-term cooling; it does not assess potential effect of in-core debris. The RELAP5-3D method conservatively assumes blockage of certain cooling flow paths by debris during recirculation and long term cooling phases in HLB and small CLB to demonstrate that PCT is acceptable. In addition to flow path blockages, STP conservatively assumes 800°F is the PCT success criterion. NRC approval of this methodology change is required by 10CFR50.59(c)(1)(viii) because the use of RELAP5-3D in this application has not been previously reviewed and approved. However, application of the proposed 800°F PCT has been found acceptable in the NRC review of WCAP 16793. The calculations performed in this methodology change complement deterministic testing by showing that the core damage risk quantification can be represented by failures from debris effects at the strainer. There is no contribution from in-core debris effects so only debris quantities that accumulate at the strainer need to be considered.

3. A change to the STP TS for ECCS and CSS incorporating a debris-specific action and required completion time is proposed. All changes to the TS require NRC review and approval per 10CFR50.90.

Attachment 4

Response to ENPB-3-1 through 3

EPNB-3-1

In Attachment 1-2, page 4, of the August 20, 2015, submittal, the licensee stated that the large main steam and feedwater line breaks were not evaluated because recirculation is not required under the plant licensing basis for STP.

a) Please explain why sump recirculation is not required for main steam and feedwater line breaks inside the containment.
b) Please discuss whether any other American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 2 piping inside the containment, besides the main steam and feedwater lines, are evaluated for debris generation and sump recirculation. If none, explain why ASME Code Class 2 piping inside the containment are not evaluated.

STP Response

a) The following will discuss why sump recirculation is not required for main steam line breaks (SLBs) and feedwater line breaks (FLBs) inside the containment. The purpose of sump recirculation is to supply water to the SIS and CSS from the containment sump when the water supply in the Refueling Water Storage Tank (RWST) is depleted. The following discusses the impact of sump recirculation on the SIS and CSS.

Safety Injection System

A large main steam or feedwater line break results in a decrease in RCS temperature and subsequent reduction in RCS pressure and water level. The SIS restores the RCS water level and pressurizes the RCS to a pressure no less than the shut-off head of the HHSI, which is approximately 1700 psig. In addition, the SIS provides boron to the RCS which terminates a potential return to power. The restoration of RCS water level, pressurization of the RCS and termination of a potential return to power occurs prior to the depletion of the water in the RWST. In the event safety injection flow were secured, the RCS pressure would be maintained by the pressurizer heaters or at the hot leg saturation pressure if the pressurizer heaters are not available. Since a SLB or FLB does not result in a loss of RCS water inventory, inventory addition from the safety injection system is not required using sump recirculation. Therefore, sump recirculation is not required to support the design function of the safety injection system for a SLB or FLB.

Containment Spray System

The purpose of the CSS is to reduce the containment pressure, temperature and minimize the release of radioisotopes (primarily iodines) to the environment. Containment peak pressure is not a concern for secondary side breaks because they are bounded by the large break LOCA event. Containment peak temperature is bounded by the SLB event. During a SLB event, superheated steam is released into containment which rapidly increases the containment atmospheric temperature. The containment atmospheric temperature quickly decreases to saturation conditions upon the initiation of the CSS. The potential for containment temperature returning to superheated conditions ends when the water inventory in the faulted steam generator is depleted. For SLBs that challenge containment pressure and temperature limits, such as the DEGB SLB, the

water inventory in the faulted steam generator is depleted prior to sump recirculation Therefore, sump recirculation is not required to support the design function of the CSS for the SLB with regard to containment temperature.

During a FLB, two-phase flow from the break occurs and is therefore bounded by the SLB for the purposes of containment pressure and temperature response.

The containment spray system is not required for radiological effects of secondary side breaks inside containment because no concurrent RCS break is postulated.

For a SLB or FLB, energy is removed from containment by the RCFCs, which is independent of sump recirculation. The CSS does not have a heat exchanger, and does not remove energy from containment. Therefore, the CSS is not required using sump recirculation to ensure the containment pressure and temperature limits are not exceeded.

Conclusion

To summarize, sump recirculation is not required to support the design functions of the SIS or CSS for a SLB or FLB.

For completeness, STPNOC evaluated the risk contribution for SLBs and FLBs for beyond-design-basis failures that would require ECCS recirculation in a "feed and bleed" function. STPNOC assumed these conditions result in core damage. The risk contribution does not affect the STPNOC conclusions.

b) All Class 1 welds were evaluated for breaks causing debris generation in the RoverD analysis with the exception of welds beyond a check valve or other isolation valve in the Class 1 piping. In Table 17 starting on Page 57 of Attachment 1-3 of the August 20, 2015 submittal is a list of these welds which are over 600 in number.

The ASME Code Class 2 piping inside containment is not evaluated for breaks that cause debris generation. The Main Steam and Main Feedwater lines are excluded as discussed above. Other Class 2 piping such as Containment Spray and Safety Injection are excluded since the lines are pressurized only for accident mitigation and are not considered as the initiating event of an accident which causes debris that might challenge the use of the emergency sump in the recirculation mode. Class 2 pipe lines such as charging and letdown of the Chemical and Volume Control System and also Steam Generator Blowdown could be pressurized during normal operation. However, mitigation of breaks in these lines does not require the use of the emergency sump in the recirculation mode. The Residual Heat Removal System has Class 2 lines which are pressurized during the cooldown and startup of the reactor. Again, mitigation of a break in the Residual Heat Removal System piping does not require the use of the emergency sump in the recirculation mode. Thus for the reasons discussed above, ASME Class 2 lines are not evaluated for breaks causing debris generation.

EPNB-3-2

In Attachment 1-4, page 22, of the August 20, 2015, submittal, under the heading, *Reactor Coolant System Weld Mitigation,* the licensee stated that " ... All STP large bore RCS [reactor coolant system] welds susceptible to pressurized water stress corrosion cracking (PWSCC) have been replaced with Alloy 690 material which is not susceptible to PWSCC (SG [steam generator] nozzles) or overlaid with non-susceptible Alloy 52/52M/152 material (pressurizer piping safe ends) with the exception of the reactor vessel nozzle welds ... "

a) Please clarify whether "the reactor vessel nozzle welds" discussed in the above statement are the J-groove welds associated with the reactor vessel closure head penetration nozzles to house the control rod drive mechanisms (i.e., control rod drive mechanism (CROM) nozzles), or the full-penetration butt welds associated with the hot-leg nozzles that are attached to the reactor vessel shell.
b) Please discuss of what material the CROM nozzles and the associated J-groove welds in both units are made. If the nozzles are composed of Alloy 600 material and the welds are Alloy 82/182, then discuss why these were not selected as break locations since these materials are susceptible to PWSCC.
c) Please identify: 1) the large bore RCS piping (e.g., hot leg, cold leg, or crossover piping) and other ASME Code Class 1 pipes (e.g., pressurizer surge line, pressurizer spray line, or safety injection piping) that contain either Alloy 690 weld material or are mitigated with Alloy 52/52M/152 material, and that are considered in the GSI-191 evaluation; and (2) all ASME Class 1 piping that is larger than 2 inches that contain Alloy 82/182 weld material, has not been mitigated with Alloy 52/52M/152 material, and are considered in the GSI-191 evaluation.

<u>STP Response:</u>

a. The reactor vessel nozzle welds excepted in the referenced statement are the full penetration butt welds on the hot and cold leg RPV shell. These dissimilar metal welds have not yet been mitigated but have been approved for mitigation by non-welded stress improvement process known as Mechanical Stress Improvement Process (MSIP) – STP Unit 1 spring 2017 (1RE20) and STP Unit 2 fall 2019 (2RE20). MSIP compresses the pipe adjacent to the weld and creates compressive stresses through ~50% of the pipe wall thickness. Eliminating the tensile stresses at the inside surface of the PWSCC susceptible weld removes one of the three conditions required to be present for PWSCC to initiate.

The CRDM J-groove weld are addressed as described in the response to part (b) of this RAI.

b. STP replacement RV Closure Head CRDM nozzles and associated butt welds (not J-groove welds) are made of Alloy 690/52/152 material and are not considered susceptible to PWSCC. STP RV Closure Heads were replaced in both units (2009 for Unit 1 and 2010 for Unit 2). Due to the relatively small size and location of the CRDM nozzles, breaks do not need to be considered since they are bounded by larger breaks that produce much more debris (Ref. NEI 04-07).

c. STP primary loop piping and branch connection pipes are constructed of stainless steel (e.g. SA-351 Gr CF8A for loop piping and SA-312 for branch piping) and is not susceptible to PWSCC.

The RCS piping welds considered for GSI-191 which have been mitigated with material not susceptible to PWSCC are:

- SG nozzles
- Pressurizer nozzles (overlays)
- RPV head CRDM penetrations (as discussed in the response to part (b) of this RAI)

The RCS piping welds considered for GSI-191 which are susceptible to PWSCC which have not been mitigated are:

- RPV hot and cold leg nozzle weld as discussed in the response to part (a) of this RAI.

EPNB-3-3

Please explain why breaks from the pressurizer heater sleeves and reactor vessel bottom-mounted instrumentation nozzles were not considered as a source of debris generation.

STP Response:

Due to their relatively small size and location, pressurizer heater sleeves and reactor vessel bottom mounted instrumentation nozzles were not considered as a source of debris generation. Any breaks for these items are bounded by larger breaks in the nearest piping which would produce much more debris (Ref. NEI 04-07). Also the only type of insulation debris from a break of one of these items would be from reflective metal insulation which is not considered to transport onto the emergency sump strainers and thus would not be an impact.

Attachment 5

Response to ESGB-3-1

ESGB-3-1

Provide additional details and clarification with respect to the manner in which the total mass of unqualified coatings was calculated. Attachment 1-2, page 67 of 95 of the August 20, 2015, submittal states, in part, that "the weight of applied coatings are determined based on a theoretical coating spread rates (sq. ft per gallon @ 1 mil thick) instead of specific vendor coating spread rates." If a 1 mil (thousandths of an inch) thick coating was assumed for IOZ or epoxy coatings, the analysis may be significantly underestimating the amount of coating debris. Please describe the thickness used in the analysis for both epoxy and IOZ coatings since the mass of epoxy within the ZOI may be impacted and the mass of unqualified IOZ throughout containment may be impacted.

<u>STP Response</u>:

The theoretical coating spread rate referenced is simply a conversion factor for calculating a coating's area of coverage per gallon (ft^2/gal) for a given dry film thickness (DFT). The relationship is defined by the following equation:

$$C = \frac{R}{DFT} \qquad \text{Eqn 1}$$

where:

C	=	Coverage (ft^2/gal)
R	=	Spread Rate ($ft^2/gal/mil$)
DFT	=	Dry film thickness (mils)

The use of this parameter does not imply that a coatings thickness of 1 mil is assumed. As an example, consider the following coatings system (these values do not represent a specific STP system, but are typical of what might be used in an unqualified coatings calculation):

Area:	10,000 ft^2
DFT:	5 mils
Weight/Gallon:	12 lbs/gal
% Solids (by Volume):	50%
% Solids (by Weight):	75%
Spread Rate:	1600 $ft^2/gal/mil$

To calculate the dry weight from the given information, the area of coverage per gallon is first calculated using the theoretical spread rate:

$$Coverage = \frac{(Spread\ Rate)\ (\ \%\ Solids\ by\ Volume)}{DFT} = \frac{\left(\frac{1600\frac{ft^2}{gal}}{mil}\right)(0.50)}{5\ mils} = 160\frac{ft^2}{gal} \qquad \text{Eqn 2}$$

The total volume (in gallons) required to cover the area can then be calculated:

$$Gallons\ Required = \frac{Area}{Coverage} = \frac{10000\ ft^2}{160\ \frac{ft^2}{gal}} = 62.5\ gal \qquad \text{Eqn 3}$$

$$\begin{aligned} Dry\ Weight &= (Gallons\ Required)(Weight\ per\ Gallon)(\%\ Soilds\ by\ Weight) \\ &= (62.5\ gal)\left(12\frac{lbs}{gal}\right)(0.75) = 562.5\ lbs \end{aligned} \qquad \text{Eqn 4}$$

The thicknesses used for unqualified coatings in the STP assessment vary for different substrates and coatings systems. Unqualified epoxy thicknesses ranged from 2 to 22 mils, and the thickness of unqualified IOZ coatings ranged from 2.5 to 6 mils. The dry weight calculations, including the dry weight thicknesses, for every unqualified coatings system considered can all be found in STP's unqualified coatings quantifications (9AC5002#1 and 9AC5002#2).

Attachment 6

Response to SNPB-3-1, -3, -4, -5, -8, -11, -12, -13, -14, -16, -19

Thermal-Hydraulic Review Questions

Note 1: the draft SNPB questions sent to STPNOC by e-mail dated October 21, 2015 (ADAMS Accession No. ML 16022A 177), were subsumed by questions SNPB-3-3 and SNPB-3-20 below.

Note 2: The following SNPB questions are from the criteria set forth in the following two NRC staff guidance documents:

- *Safety Evaluation for the Westinghouse Topical Report WCAP-16793-NP "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2. This will be identified as "SE for WCAP."*
- *NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: L WR Edition," Section 15. 0. 2. The subsection referenced will be provided with the question. This will be identified as "SRP."*

SNPB-3-1

Cladding Oxide

Please demonstrate that the thickness of the cladding oxide and the deposits of material on the fuel do not exceed 0.050 inches in any fuel region.
Criterion 0.1 Reference: SE for WCAP

STP Response:

Starting on page 74 of 77 of ML083520326, STP RAI #31 and 36. The deposit layer thickness for STP fuel is 13.64 mils.

SNPB-3-3

Clarification on Core Bypass Blockage

During the audit, STPNOC considered performing the LTCC analysis with the core bypass open to allow flow in the axial direction. If STPNOC credits the use of the bypass, it should provide test data to demonstrate that the bypass will not block during the scenarios. This test data should bound the flow rates, flow areas, and debris loading expected in the RCS.
Criterion 1.3 Reference: SRP, III.3c

STP Response:

Flow path analysis basis/background

There are six flow paths that need to be taken into account to accurately model the Barrel-Baffle (BB) flow during blockage:

1. Thimble Tube (TT) flow
2. Core former-to-fuel gap flows
3. LOCA hole flows
4. BB flows
5. Cold leg to hot leg leakage flow (direct bypass)
6. Upper Head Spray Nozzle (UHSN) flow

Of these 6 flows, only the TT and UHSN have been well characterized at STP in specific studies for upper head cooling and removal of thimble plugs. The other flow paths, (2, 3, 4, and 5) are not characterized except to ensure that the total meets either design (non-conservative for GSI-191), or best estimate.

Assuming minimum flow through the open BB flow path is a conservative approach since it reduces the amount of flow available to the top of core with the only remaining path being the UHSNs. Since the UHSN path requires cooling water to first go the upper head and then flow down to the top of the core, the water available to the top of the core is delayed and minimized resulting in flow to the top of the core being conservatively estimated.

Phenomenology basis/background

STP will assume the largest HLB that is assumed to fail the strainer is 16". While this assumption increases the risk related to the concerns raised in GSI-191, it bounds locations where fine fiber amounts exceed the amounts tested based on an assumed 17D ZOI for unjacketed NUKON.

By adopting the smaller maximum break size, uncertainty associated with complications at larger (up to DEGB) break behavior are subsumed in the assumed failures above 16"; any thermal-hydraulic uncertainties associated with more violent behavior realized at larger break sizes is bounded.

SNPB-3-4

Describe Important Phenomena

Please provide a description of the important phenomena being modeled in the LTCC EM for each of the accident scenarios being simulated. These phenomena should include those important to obtaining the correct initial conditions for the long-term phase, and the important phenomena during the long-term phase.
Criterion 1.3 Reference: SRP, III.3c

STP Response:
HOT LEG BREAK LOCA
The important phenomena in a HLB are grouped based on the accident phases: Blowdown, refill, reflood, long-term core cooling, and core blockage. As break size decreases from large to small, the phases become less distinct, and some phenomena described below become unimportant. For example, in smaller break sizes, the ECCS flow rapidly overcomes the volumetric flow out the break; the RCS and core are repressurized and flooded.

Blowdown.
The blowdown period is the result of a hypothesized break in the coolant system through which the primary coolant is expelled. Blowdown physical processes include choked flow at the break, fluid flashing and depressurization, and heating of the fuel rods due to degraded heat transfer. During the blowdown phase, some components are affected more than others. In particular, the heat generated in the core may not be adequately removed due to decreased heat transfer resulting from the loss of fluid. The performance of the reactor coolant pumps will degrade as the coolant flashes. The steam generator heat transfer degrades because primary system temperatures are lower than secondary system temperatures during this period. In small break LOCA, heat removal from the steam generators is possible early due to rapid refilling of the RCS.

Important phenomena and processes expected to occur during the blowdown phase are:
- Choked flow at the break. Choked flow is expected to occur due to the pressure ratio between the RCS and RCB being sufficiently high. Both subcooled (initially) and saturated (later) choked flows are expected.
- Flashing and voiding. The volumetric flow through the break greatly exceeds the makeup flow, producing flashing and voiding in all volumes. The primary coolant, initially subcooled, reaches saturation temperature of the RCS water.
- Depressurization. As a result of the volumetric flow difference between break flow and makeup flow, the primary system pressure decreases.
- Release of stored energy and decay heat in the fuel. Unlike the large break in cold leg, the flow through the core during a large hot leg break is expected to accelerate instead of stagnating, removing the stored energy and decay heat efficiently in the initial phase of the accident.

- Boiling on the cladding surface. The heat transfer at the cladding surface is initially dominated by single phase forced convection and rapidly (depending on break size) transitions to boiling-dominated heat transfer as the system depressurizes.

Refill.

In a hot leg break scenario, refill and reflood are not realized as distinct processes. Instead, because the combined resistance to flow of the pump and steam generator to the break opposite the vessel side is much higher than the vessel side, safety injection water (from accumulators and SI pumps) preferentially flows through the core to the broken hot leg. Because of the overall pressure gradient developed by the hypothesized break at the core outlet, phenomena and processes such as counter current flow limiting (CCFL) and ECC bypass in the downcomer are not of interest in HLB scenarios.

Reflood

Because the pressure gradient favors ECCS flow through the core (in hot leg break), the core is continuously supplied with coolant and flooded almost immediately.

Long-term core cooling.

Decay heat removal is dominated by forced convection heat transfer. In the long-term phase, colder water from the RWST is injected into the core until the RWST water is depleted. When the RWST is depleted, the ECCS transfers to sump recirculation mode with a higher injection temperature; the vessel remains full of subcooled water up to the hot loop nozzles where the water exits back to the reactor containment pool.

Core blockage.

In recirculation phase, debris may accumulate at the bottom of the core and reduce the ECCS flow through the core. Under the limiting assumption of hypothetical full core blockage with ECCS flow to the core inhibited, the temperature of the water in the core would be expected to increase to saturation, and voiding is expected to occur until water from the UHSNs reaches the top of the core. The following phenomena are expected to occur during this phase:

- Forced to natural convection heat transfer. The heat transfer regime in the core suddenly changes from forced single-phase convection to two-phase natural circulation. As voiding increases, the heat transfer coefficient at the cladding surface continues to reduce. If enough liquid coolant arriving from alternative flow paths replaces the water boil off from the core, the heat transfer regime at the cladding wall is expected to be saturated boiling with the cladding temperature slightly above the saturation temperature at the RCS pressure.

- Counter current flow limiting (CCFL). Any injected ECCS water that reaches the core under this scenario will arrive from the top of the core. Due to boiling in the core, steam exiting the core may establish conditions favorable for CCFL.

The most relevant phenomena which are expected to occur during a large break are summarized in the table below. The table is based on a Westinghouse 4-loop plant of CSAU study [1].

Table 1. LB LOCA Important Phenomena

Phenomena	**Phenomena description**	**Phase[a]**
Asymmetries	A difference in T-H behavior that can be attributed to the geometrically asymmetric arrangement of hardware.	1, 2
Boiling – film	Boiling regime in which vapor blankets all or an appreciable portion of the heating surface.	1,2,3,5
Boiling – transition	A boiling regime that spans the boiling surface between critical heat flux and minimum film boiling	1,2,3,5
Condensation – interfacial	The process whereby steam is cooled due to contact with a colder liquid, resulting in a change of phase from vapor to liquid at the interface between the two phases.	2
Entrainment / deentrainment	The process whereby liquid is captured (entrained) by a high-velocity steam flow. The process whereby liquid departs (deentrained) from a steam flow.	2,3,5
Evaporation – interfacial	The process whereby a fluid changes from the liquid state to the vapor state by the addition of energy.	1,2,3,5
Flashing – interfacial	The process whereby fluid changes from the liquid state to the vapor state due to a reduction in the fluid pressure, which lowers the saturation temperature.	1
Flow – countercurrent	The process whereby liquid flows opposite (counter) to the gas flow direction.	2,5
Flow - choked	The maximum possible flow through a flow constricting item of hardware, usually a nozzle, orifice, or break in a pipe.	1,2
Flow – multidimensional	Flow that has two or more dominant velocity vectors. Examples are multidimensional flows in a PWR core during reflooding.	2,3,5
Heat conductance – fuel-clad gap	The overall thermal resistance to the flow of heat between the fuel pellets and cladding in a nuclear fuel rod.	1
Heat transfer – forced convection to vapor	Process of energy transport by the combined action of heat conduction, energy storage, and mixing motion.	2
Heat transfer – stored energy release	The process by which the energy within a solid structure is released to a lower energy state through one or more heat transfer processes, e.g., conduction and convection. Applies specifically to the transport of the energy residing in fuel rods operating at full power to the coolant following a reactor trip. The first peak is associated with the blowdown time period and is caused by the initial stored energy in the fuel rods and degraded fuel rod-to-coolant heat transfer.	1
Interfacial shear	The friction caused by the velocity difference between two phases at their interface.	2,3,5
Level	The vertical height of a column of single- or two-phase fluid.	3
Noncondensable effects	The impact of the presence of noncondensable gases upon heat transfer or any other phenomenon such as flow, condensation, flashing, and vapor volume expansion.	3,5
Oscillations	The periodic variation of any given hydraulic characteristic between two values.	3,5
Power-decay heat	Heat produced by the decay of radioactive nuclides.	2,3,4,5
RC Pump – performance, including degradation	The behavior of a pump under all normal and off-normal conditions.	1
a: Phase of the LB LOCA sequence: Blowdown = 1, Refill = 2, Reflood = 3, Long-Term = 4, Core Blockage = 5.		

SMALL COLD LEG BREAK

The important phenomena in a small CLB are grouped based on the accident phases: Blowdown, natural circulation, loop seal clearance, boil-off, core recovery, long-term core cooling, and core blockage.

Blowdown.

On initiation of the break, there is a rapid depressurization of the primary side of the RCS. Reactor trip is initiated on a low pressurizer pressure set point. RCPs are tripped by the operators when the Emergency Operation Procedures conditions are met (low RCS pressure and ECCS flow). A safety injection signal occurs when the primary pressure decreases to the SI injection set point. The RCS remains liquid solid for most of the blowdown period, with phase separation starting to occur in the upper head, upper plenum and hot legs near the end of this period. During the blowdown period, the break flow is single phase liquid only. Eventually, the rapid depressurization ends and the RCS reaches a pressure just above the steam generator secondary side pressure.

Natural circulation.

At the end of the blowdown period, the RCS reaches a quasi-equilibrium condition which can last for several hundred seconds depending on break size. During this period, the loop seals remain plugged and the system drains from the top down with voids beginning to form at the top of the steam generator tubes and continuing to form in the upper head and top of the upper plenum region. Decay heat is removed by the steam generators during this phase. Vapor generated in the core is trapped within the RCS by liquid plugs in the loop seals, and a low quality flow exits the break. This period is referred to as the natural circulation period.

Loop Seal Clearance.

The third period is the loop seal clearance period. When the liquid level in the downhill side (cold leg tube side) of the steam generator is depressed to the elevation of the loop seal, steam previously trapped in the RCS can be vented to the break. The break flow, previously a low quality mixture, transitions to primarily steam. Prior to loop seal venting, the inner vessel mixture level can drop rapidly, resulting in a deep but short core uncovery. Following loop seal venting, the core level recovers to about the cold leg elevation, as pressure imbalances throughout the RCS are relieved.

Boil-off.

Following loop seal venting, the vessel mixture level will decrease. In this period, the decrease is due to the gradual boil-off of the liquid inventory in the reactor vessel. The mixture level will reach a minimum, in some cases resulting in a deep core uncovery. The boil-off period ends when the collapsed liquid level in the core reaches a minimum. At this time, the RCS has depressurized to the accumulator set point, and the core boil-off rate matches the delivery of safety injection to the vessel.

Core Recovery and Long-term cooling.

The core recovery period extends from the time at which the inner vessel mixture level reaches a minimum in the boil-off period, until all parts of the core quench and are covered by a low quality mixture. In the long-term cooling period, the entire core is quenched and the safety injection flow equalizes the break flow.

Core blockage.

After the RWST is exhausted, ECCS transfers to recirculation mode. Under a hypothetical full core blockage, due to interruption of the ECCS flow, the temperature of the water in the core is expected to increase to saturation, and voiding is expected to occur until water from alternative flow paths reaches the top of the core. The following phenomena are expected to occur during this phase:

- Forced to natural convection heat transfer. The heat transfer regime in the core suddenly changes from forced single-phase convection to two-phase natural circulation. Liquid coolant from alternative flow paths will replace the water boil off from the core. Heat transfer regime at the cladding wall is expected to be saturated boiling with the cladding temperature slightly above the saturation temperature at the RCS pressure.
- Counter current flow limiting (CCFL). Any injected ECCS water that reaches the core under this scenario will arrive from the top of the core. Due to boiling in the core, steam exiting the core may establish conditions favorable for CCFL. Due to the relatively smaller amount to vapor produced in the core and the larger amount of water inventory in the vessel compared to a large break scenarios, the conditions for CCFL may not be met.

Due to the longer sump switchover time compared to the large break scenarios; a subsequently lower decay heat to be removed from the core; and a larger water inventory in the RCS (the RCS and vessel are expected to be full of water at this time); these phenomena may have less importance than they do in large break scenarios.

The most relevant phenomena which are expected to occur during a small cold leg break are summarized in the table below. The table is based on Westinghouse 4-loop plant PIRT panel [2].

Table 2. SB LOCA Important Phenomena

Phenomena	Phenomena description	Phase[a]
Condensation – fluid to surface	The process whereby steam is cooled due to contact with a colder surface, resulting in a change of phase from vapor to liquid at the surface.	1, 3
Condensation – interfacial	The process whereby steam is cooled due to contact with a colder liquid, resulting in a change of phase from vapor to liquid at the interface between the two phases.	4, 5
Entrainment / deentrainment	The process whereby liquid is captured (entrained) by a high-velocity steam flow. The process whereby liquid departs (deentrained) from a steam flow.	3
Flashing – interfacial	The process whereby fluid changes from the liquid state to the vapor state due to a reduction in the fluid pressure, which lowers the saturation temperature.	3, 4, 5
Flow regime – break inlet	The characteristics of the flow at the break entrance, e.g., subcooled liquid, saturated, two-phase, stratified, vapor, etc.	All
Flow – countercurrent	The process whereby liquid flows opposite (counter) to the gas flow direction.	2, 3, 6
Flow – choked	The maximum possible flow through a flow constricting item of hardware, usually a nozzle, orifice, or break in a pipe.	All
Flow – gap	Flow through the hot leg to downcomer gap.	3
Heat transfer – post-CHF	Heat transfer between the two-phase fluid and the heated surface in the liquid-deficient region downstream of the CHF point, i.e., the location at which the heat transfer condition of the two-phase flow substantially deteriorates.	4, 5
Interfacial shear	The friction caused by the velocity difference between two phases at their interface.	3, 6
Level	The vertical height of a column of single- or two-phase fluid.	3, 4, 5, 6
Oxidation	A chemical reaction that increases the oxidation content of a material. Of specific interest is cladding oxidation, which occurs at elevated temperatures, which can occur only under accident conditions.	4, 5
Power – 3D distribution	The axial, radial and azimuthal power variation in a core.	4, 5
Power – decay heat	Heat produced by the decay of radioactive nuclides.	All
Power – local peaking (fuel rod)	The ratio of power at a location (specific fuel rod) to the core average power.	4, 5
Pressure drop	The reduction in pressure with distance.	3
Rewet	The post-dryout process in which liquid once again resumes intimate contact with a heated surface.	4, 5
Stratification - horizontal	The variation of physical properties such as temperature or density across the vertical cross section of a fluid body having a primarily horizontal orientation, e.g., the cold leg of a nuclear steam supply system.	3
a:.Blowdown = 1, Natural Circulation = 2, Loop Seal Clearance = 3, Boil-off = 4, and Core Recovery and Long-Term= 5, Core Blockage = 6.		

References:

[1]. Technical Program Group, EG&G Idaho, Inc., Quantifying Reactor Safety Margins:A Application of CSAU to a LB LOCA, USNRC report NUREG/CR-5249, 1989

[2] S. M. Bajorek, A. Ginsberg, D. J. Shimeck, K. Ohkawa, M. Y. Young, L. E. Hochreiter, P. Griffith, Y. Hassan, T. Fernandez, and D. Speyer, "Small Break Loss of Coolant Accident Phenomena Identification and Ranking Table (PIRT) for Westinghouse Pressurized Water Reactors," Proceedings of the Ninth International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-9), San Francisco, California (October 3–8, 1999)

SNPB-3-5

Debris at Grid Spacers

Please describe how the LTCC EM accounts for potential blockages at the spacer grid in the core above the bottom grid.
Criterion 1.3 Reference: SRP, III.3c

STP Response:

STP has approximately 2 lbs of CRUD total (ML15901A440, Attachment 3 Pages 8 of 10 and 9 of 10) in the RCS that can be transported to the fuel grids. This small amount (CRUD is primarily oxidized metals and therefore dense) of fine particles is not capable of blocking up fuel channels. Fibrous debris collected at the spacer grids will not cause flow blockage or cause inadequate cooling (ML11292A021, Section 4).

SNPB-3-8

How are the Phenomena Modeled

Please summarize how the important phenomena are being modeled in the LTCC EM. This discussion should provide the phenomena and a summary of how it is being modeled (e.g., through the field equations, by an identified closure relationship).
Criterion 2.2 Reference: SRP, III.3a

STP Response:

The RELAP5-3D code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code contains models to predict the coupled behavior of the reactor coolant system and the core during loss-of-coolant accident scenarios.

The code is based on a non-homogeneous and non-equilibrium model for two-phase systems that is solved by a partially implicit numerical scheme, and includes important first-order effects necessary for accurate prediction of system transients.

RELAP5-3D includes many component models from which normal operation and transients in PWR systems can be simulated. The component models include pumps, valves, pipes, transient conduction-convection heat transfer systems, reactor kinetics, separators, annuli, pressurizers, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, fluid-wall friction, branching, and choked flow [1].

The RELAP5-3D code manual is organized in five volumes. The content of each volume is briefly described below:

Volume I: Code Structure, System Models and Solution Methods.
This volume presents modeling theory and associated numerical schemes adopted in RELAP5-3D. The volume is divided into eight sections:

- Section 1: Introduction
- Section 2: Code Architecture
- Section 3: Hydrodynamic Model (including field equations, state relationships, constitutive models, special process models)
- Section 4: Heat structure models
- Section 5: Trip system
- Section 6: Control system
- Section 7: Reactor kinetics model
- Section 8: Special techniques

Volume II: User's Guide and Input Requirements
Detailed instructions for code application and input data preparation are included in this volume. The volume is organized in eight sections:

- Section 1: Introduction
- Section 2: Hydrodynamics
- Section 3: Heat structures
- Section 4: Trips and controls
- Section 5: Reactor kinetics

- Section 6: General tables and component tables
- Section 7: Initial and boundary conditions
- Section 8: Problem control

An appendix to volume II is included containing a detailed explanation of all input cards.

<u>Volume III: Developmental Assessment</u>
This volume provides the results of developmental assessment cases that demonstrate and verify the models used in the code. The volume is divided into six sections:

- Section 1: Introduction
- Section 2: Developmental assessment matrix
- Section 3: Phenomenological cases
- Section 4: Separate effects cases
- Section 5: Integral effects cases
- Section 6: Summary and conclusions

<u>Volume IV: Models and Correlations</u>
This volume discusses in detail RELAP5-3D models and correlations. The volume is organized in eleven sections:

- Section 1: Introduction
- Section 2: Field equations
- Section 3: Flow regime maps
- Section 4: Closure relations for the fluid energy equations
- Section 5: Closure relations required by fluid mass conservation equations
- Section 6: Momentum equation closure relations
- Section 7: Flow process models
- Section 8: Special components models
- Section 9: Heat structures process models
- Section 10: Closure relations required by extra mass conservation fields
- Section 11: Steady-state

<u>Volume V: User's Guidelines</u>
This volume contains guidelines that have evolved over the past several years through the use of the RELAP5-3D code. The volume is organized in five sections:

- Section 1: Introduction
- Section 2: Fundamental practices
- Section 3: General practices
- Section 4: Specific practices
- Section 5: Pressurized water reactor example applications

The important phenomena in LB and SB LOCAs are identified based on the accident phase, and summarized in Table 1 and Table 2 of RAI-SNPB-04 response. The following tables describe how the important phenomena for LB and SB LOCAs are modeled in RELAP5-3D.

Table I. LB LOCA Phenomena and Modeling

Phenomena	Phenomena description	Model	RELAP5-3D Volume/Section
Asymmetries	A difference in T-H behavior that can be attributed to the geometrically asymmetric arrangement of hardware.	Field Equation/Basic Equation: All fluid flow equations	I-3.1.1 - I-3.1.4 & IV-2.1
Boiling – film	Boiling regime in which vapor blankets all or an appreciable portion of the heating surface.	Closure (constitutive) Relations (including regime maps): Wall-to-fluid energy exchange: film boiling	I-3.3.9 & I-3.3.10 & IV-4.2
Boiling – transition	A boiling regime that spans the boiling surface between critical heat flux and minimum film boiling	Closure (constitutive) Relations (including regime maps): Wall-to-phase energy exchange: transition boiling	I-3.3.9 & I-3.3.10 & IV-4.2
Condensation – interfacial	The process whereby steam is cooled due to contact with a colder liquid, resulting in a change of phase from vapor to liquid at the interface between the two phases.	Closure (constitutive) Relations (including regime maps): Interfacial energy exchange: all flow regimes	I-3.3.11 & IV-4.1
Entrainment / deentrainment	The process whereby liquid is captured (entrained) by a high-velocity steam flow. The process whereby liquid departs (deentrained) from a steam flow.	Closure (constitutive) Relations (including regime maps): Interfacial momentum exchange: all flow regimes.	I-3.3.6 & IV-6.1
Evaporation – interfacial	The process whereby a fluid changes from the liquid state to the vapor state by the addition of energy.	Closure (constitutive) Relations (including regime maps): Interfacial energy exchange: all flow regimes.	I-3.3.11 & IV-4.1
Flashing – interfacial	The process whereby fluid changes from the liquid state to the vapor state due to a reduction in the fluid pressure, which lowers the saturation temperature.	Closure (constitutive) Relations (including regime maps): Interfacial energy exchange: all flow regimes	I-3.3.11 & IV-4.1
Flow – countercurrent	The process whereby liquid flows opposite (counter) to the gas flow direction.	Closure (constitutive) Relations (including regime maps): Interfacial momentum exchange: all flow regimes. Special Process Models: CCFL	I-3.3.6 & IV-6.1 I-3.4.7 & IV-7.4
Flow - choked	The maximum possible flow through a flow constricting item of hardware, usually a nozzle, orifice, or break in a pipe.	Special Process Models: Critical flow model	I-3.4.1 & IV-7.2 & IV-7.3
Flow – multidimensional	Flow that has two or more dominant velocity vectors. Examples are multidimensional flows in a PWR core during reflooding.	Field Equation/Basic Equation: 3D vessel model	I-3.1.11
Heat conductance – fuel-clad gap	The overall thermal resistance to the flow of heat between the fuel pellets and cladding in a nuclear fuel rod.	Field Equation/Basic Equation: fuel-clad gap model	I-4.12 & IV-9.3

Phenomena	Phenomena description	Model	RELAP5-3D Volume/Section
Heat transfer – forced convection to vapor	Process of energy transport by the combined action of heat conduction, energy storage, and mixing motion.	Closure (constitutive) Relations (including regime maps): Wall-to-fluid energy exchange: single-phase vapor	I-3.3.9 & 3.3.10 & IV-4.2
Heat transfer – stored energy release	The process by which the energy within a solid structure is released to a lower energy state through one or more heat transfer processes, e.g., conduction and convection. Applies specifically to the transport of the energy residing in fuel rods operating at full power to the coolant following a reactor trip. The first peak is associated with the blowdown time period and is caused by the initial stored energy in the fuel rods and degraded fuel rod-to-coolant heat transfer.	Field Equation/Basic Equation: Conduction equation & fuel-clad gap	I-4 & IV-9.1 & I-4.12 & IV-9.3
Interfacial shear	The friction caused by the velocity difference between two phases at their interface.	Closure (constitutive) Relations (including regime maps): Interfacial momentum exchange: all flow regimes	I-3.3.6 & IV-6.1
Level	The vertical height of a column of single- or two-phase fluid.	Field Equation/Basic Equation: all: fluid flow equations	I-3.1.1 - I-3.1.4 & IV-3.2
Noncondensable effects	The impact of the presence of noncondensable gases upon heat transfer or any other phenomenon such as flow, condensation, flashing, and vapor volume expansion.	Closure (constitutive) Relations (including regime maps): Interfacial energy exchange: effect of noncondensables	IV-4.1.4 & IV-4.2.1
Oscillations	The periodic variation of any given hydraulic characteristic between two values.	Field Equation/Basic Equation: All fluid flow equations Closure (constitutive) Relations (including regime maps): Fluid momentum closure: all flow regimes Closure (constitutive) Relations (including regime maps): Fluid energy closure: all flow regimes	I-3.1.1 - I-3.1.4 & IV-2.1 I-3.3 & IV-6 I-3.3 & IV-4
Power-decay heat	Heat produced by the decay of radioactive nuclides.	Field Equation/Basic Equation: Power generation	I-7 & IV-9.4
Pump – performance, including degradation	The behavior of a pump under all normal and off-normal conditions.	Special Component Models: Pump component	I-3.5.4 & IV-8.1

Table 2. SB LOCA Phenomena and Modeling

Phenomena	Phenomena description	Model	RELAP5-3D Volume/Section
Condensation – fluid to surface	The process whereby steam is cooled due to contact with a colder surface, resulting in a change of phase from vapor to liquid at the surface.	Closure (constitutive) Relations (including regime maps): Wall-to-fluid energy exchange: condensation	I-3.3.9 & I-3.3.10 & IV-4.2
Condensation – interfacial	The process whereby steam is cooled due to contact with a colder liquid, resulting in a change of phase from vapor to liquid at the interface between the two phases.	Closure (constitutive) Relations (including regime maps): Interfacial energy exchange: all flow regimes	I-3.3.11 & IV-4.1
Entrainment / deentrainment	The process whereby liquid is captured (entrained) by a high-velocity steam flow. The process whereby liquid departs (deentrained) from a steam flow.	Closure (constitutive) Relations (including regime maps): Interfacial momentum exchange: all flow regimes	I-3.3.6 & IV-6.1
Flashing – interfacial	The process whereby fluid changes from the liquid state to the vapor state due to a reduction in the fluid pressure, which lowers the saturation temperature.	Closure (constitutive) Relations (including regime maps): Interfacial energy exchange: all flow regimes	I-3.3.11 & IV-4.1
Flow regime – break inlet	The characteristics of the flow at the break entrance, e.g., subcooled liquid, saturated, two-phase, stratified, vapor, etc.	Closure (constitutive) Relations (including regime maps): All flow regimes.	I-3.1.1- 3.1.4 & IV-3
Flow – countercurrent	The process whereby liquid flows opposite (counter) to the gas flow direction.	Closure (constitutive) Relations (including regime maps): All flow regimes. Special Process Model: CCFL	I-3.3.6 & IV-6.1 I-3.4.7 & IV-7.4
Flow – choked	The maximum possible flow through a flow constricting item of hardware, usually a nozzle, orifice, or break in a pipe.	Special Process Model: Choked flow model	I-3.4.1 & IV-7.2 & IV-7.3
Flow – gap	Flow through the hot leg to downcomer gap.	Field Equation/Basic Equation: All fluid flow equations	I-3.1.1 - I-3.1.4 & IV-3.1
Heat transfer – post-CHF	Heat transfer between the two-phase fluid and the heated surface in the liquid-deficient region downstream of the CHF point, i.e., the location at which the heat transfer condition of the two-phase flow substantially deteriorates.	Closure (constitutive) Relations (including regime maps): Wall-to-fluid energy exchange & Interfacial energy exchange: transition boiling, film boiling	I-3.3.9 & I-3.3.10 IV-4.1 & IV-4.2
Interfacial shear	The friction caused by the velocity difference between two phases at their interface.	Closure (constitutive) Relations (including regime maps): Interfacial momentum exchange: all flow regimes	I-3.3.6 & IV-6.1
Level	The vertical height of a column of single- or two-phase fluid.	Field Equation/Basic Equation: All fluid flow equations	I-3.1.1 - I-3.1.4 & I-3.3.1 & IV-3.2

Phenomena	Phenomena description	Model	RELAP5-3D Volume/Section
Oxidation	A chemical reaction that increases the oxidation content of a material. Of specific interest is cladding oxidation, which occurs at elevated temperatures, which can occur only under accident conditions.	Field Equation/Basic Equation: Heat structure: metal-water interaction	I-4.15
Power – 3D distribution	The axial, radial and azimuthal power variation in a core.	Field Equation/Basic Equation: Reactor Kinetics Models - 3D kinetics	I-7.2
Power – decay heat	Heat produced by the decay of radioactive nuclides.	Field Equation/Basic Equation: Power generation	I-7 & IV-9.4
Power – local peaking (fuel rod)	The ratio of power at a location (specific fuel rod) to the core average power.	Field Equation/Basic Equation: Reactor Kinetics Models - 3D kinetics	I-7.2
Pressure drop	The reduction in pressure with distance.	Field Equation/Basic Equation: All fluid flow equations Closure (constitutive) Relations (including regime maps): Momentum and energy equations closures	I-3.1.1 - I-3.1.4 & IV-3.1 I-3.3 & IV-6; I-3.3 & IV-4
Rewet	The post-dryout process in which liquid once again resumes intimate contact with a heated surface.	Closure (constitutive) Relations (including regime maps): Wall-to-fluid energy exchange: all flow regimes Closure (constitutive) Relations (including regime maps): Reflood heat transfer models: Wall-to-fluid heat transfer.	I-3.3.9 & I-3.3.10 & IV-4.2 IV-4.4.5 & IV-4.4.6
Stratification - horizontal	The variation of physical properties such as temperature or density across the vertical cross section of a fluid body having a primarily horizontal orientation, e.g., the cold leg of a nuclear steam supply system.	Field Equation/Basic Equation: Mass, momentum energy equations Closure (constitutive) Relations (including regime maps): Regime maps: stratified flow	I-3.1.1 - I-3.1.4 & IV-3.1 I-3.3.2 & IV-3.1

Reference:

[1]. RELAP5-3D Code Manual, Vol. I "Code Structure, System Models and Solution Methods". INEEL-EXT-98-00834, Revision 4.1, September 2013.

SNPB-3-11

Modeling of Important Phenomena

Please provide a summary of the important phenomena and discuss how the LTCC EM models these phenomena.
Criterion 3.2 Reference: SRP, III.3b

STP Response:

The RELAP5 series of codes has been developed at the Idaho National Laboratory (INL) under sponsorship of the U.S. Department of Energy, the U.S. Nuclear Regulatory Commission, members of the International Code Assessment and Applications Program (ICAP), members of the Code Applications and Maintenance Program (CAMP), and members of the International RELAP5 Users Group (IRUG). Specific applications of the code have included simulations of transients in light water reactor (LWR) systems such as loss of coolant, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip.

RELAP5-3D is the latest in the series of RELAP5 codes and it includes all the capabilities of the RELAP5 family in simulating the behavior of a reactor coolant system during transients such as hypothesized LOCA scenarios. The mission of the RELAP5-3D development program was to develop a code version suitable for the analysis of all transients and postulated accidents in LWR systems, including both large- and small-break loss-of-coolant accidents (LOCAs) [1].

RELAP5-3D includes many component models from which normal operation and transients in PWR systems can be simulated. The component models include pumps, valves, pipes, transient conduction-convection heat transfer systems, reactor kinetics, separators, annuli, pressurizers, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, fluid-wall friction, branching, and choked flow [1].

The RELAP5 code has been widely used for analysis of LOCA scenarios of different break sizes including large and small breaks. The use of the code has been extended from the initial phases of the accident to the long term core cooling phase [2].

Volume V of the RELAP5-3D code manual [3] describes the modeling techniques developed by INL and researchers of national and foreign institutions for small and large break LOCA analysis. In particular, Section 5 is dedicated to the analysis of steady-state and transients of PWR, and in particular to large break LOCA (Section 5.1.8). The capabilities of the current version of the code and its predecessors have also been assessed against experimental data from integral effect test facilities and referenced in Section 5.1.9 of Volume V.

References:

[1]. RELAP5-3D Code Manual, Vol. I "Code Structure, System Models and Solution Methods". INEEL-EXT-98-00834, Revision 4.1, September 2013.

[2]. NUREG/CR-6770 LA-UR-01-5561, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," Los Alamos National Laboratory, August 2002.

[3]. RELAP5-3D Code Manual, Vol. V "User's Guidelines". INEEL-EXT-98-00834, Revision 4.1, September 2013.

SNPB-3-12

Field Equations

Please define and provide a summary of the field equations for the LTCC EM. This should include identification of the of the conservation equation (e.g., mass, momentum) and the number of dimensions of the equation. For portions of the RCS model that change in nodalization (e.g., 1-0 to 3-0), a separate description may be necessary. Additionally, demonstrate that these equations are able to model the necessary phenomena.
Criterion 3.3 Reference: SRP, III.3b

STP Response:

The RELAP5-3D hydrodynamic model is a transient, two-fluid model for two-phase vapor/gas-liquid mixture flow using eight field equations (for eight primary dependent variables). The primary dependent variables are pressure (P), phasic specific internal energies (U_g, U_f), vapor/gas volume fraction (void fraction) (α_g), phasic velocities (v_g, v_f), noncondensable quality (X_n), and boron density (ρ_b). In the one-dimensional equation model, the independent variables are time (t) and distance.

The secondary dependent variables used in the equations are phasic densities (ρ_g, ρ_f), phasic temperatures (T_g, T_f), saturation temperature (T_s), and noncondensable mass fraction in noncondensable gas phase (X_{ni}) for the i-th noncondensable species.
The basic field equations for the two-fluid nonequilibrium model consist of:

- two phasic continuity equations,
- two phasic momentum equations, and
- two phasic energy equations.

The equations are written in differential stream tube form with time and one space dimension (one dimensional model) as independent variables and in terms of time and volume-average dependent variables.

Volume I of the RELAP5-3D code manual [1] provides a detailed description of the:

- field equations: Section 3.1,
- numerical formulation of the basic conservation of mass, momentum, and energy: Sections 3.1.1, and 3.1.2, and
- numerical scheme adopted: Sections 3.1.3-10.

The proposed LTCC EM uses exclusively one dimensional components to simulate the regions of the RCS. Only the one dimensional conservation equations are used[1].

The use of the two-fluid nonequilibrium model has been largely discussed and proven to be more convenient than previous simplified approaches (for example, the well known homogeneous-equilibrium approximation). Bestion [2] highlighted the need for a two-fluid

[1] The RELAP5-3D includes a three-dimensional set of conservation equations described in Section 3.1.11. Although these equations are not used in the EM.

model in the current generation system codes, identifying unacceptable drawbacks of previous models based on improvements of the Homogeneous Equilibrium Model.

The advantage of the two-fluid model is that both mechanical non-equilibrium (phasic slip) and thermal non-equilibrium (different temperatures of the phases) can be modeled. Mechanical non-equilibrium can be significant in some phases of a loss-of-coolant accident progression. Examples include behavior of the ECC water (when it is injected into the system it does not immediately mix and flow at the same velocity as the steam), cooling water flowing down the downcomer counter current to escaping steam, countercurrent flow of steam and water occurring in the steam generator tubes during reflux cooling, and stratified flow occurring in horizontal piping components with little interaction between liquid and vapor. Important thermal non-equilibrium phenomena requiring a two-fluid model during LOCA simulation include sub-cooled liquid with direct contact condensation after ECCS injection and superheated vapor during Post-CHF heat transfer in the core [3].

References:

[1]. RELAP5-3D Code Manual, Vol. I "Code Structure, System Models and Solution Methods". INEEL-EXT-98-00834, Revision 4.1, September 2013.

[2] Dominique Bestion, "System Code Models and Capabilities", THICKET 2008 – Session III – Paper 06, 2013.

[3]. RELAP5/MOD3 Code Manual Volume 6: Validation of Numerical Techniques in RELAP5/MOD3.0; A. S. Shieh V. H. Ransom R. Krishnamurthy, NUREG/CR-5535/Rev 1-Vol VI.

SNPB-3-13

Validation of Closure Relationships

For the closure relationships identified, please provide appropriate validation for the use of this relationship over its expected application domain. This validation should include comparisons to separate effects tests and/or integral test data and appropriately address the model's uncertainty. Where appropriate, discuss any similarity criteria, scaling rationale, assumptions, simplifications, and/or compensating errors.
Criterion 3.4, 3.8, 3.9, 4.3, 4.6, 5.2, 5.4, 5.5, 5.6 Reference: SRP, III.3b, d, e

STP Response:

The RELAP5-3D code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code is able to model the coupled behavior of the reactor coolant system and the core during loss-of-coolant accident scenarios. The code development has benefitted from extensive application and comparison to data from: phenomenological (PT); separate effects (SET); and integral effects (IET) test cases (including LOFT, PBF, Semiscale, ACRR, NRU, and other experimental programs). A developmental assessment has been performed for the RELAP5-3D computer code. This assessment used a combination of phenomenological, separate effects, and integral effects test cases to investigate how well selected code models perform. A detailed description of the validation matrix, the experimental data used, and the results of the code assessment are included in the RELAP5-3D code manual (Volume III).

Tables 1 and 2 summarize the benchmark validation performed on the RELAP5-3D code based on each of the important phenomena for LB and SB LOCA respectively. These phenomena are identified and described in the response to RAI-SNPB-04. Table 3 lists the IET cases included in the code assessment for both LB and SB LOCAs.

Table 1. LB LOCA Phenomena – Developmental Assessment

Phenomena	Vol IV Sec. #	Test Type	Test Name
Asymmetries	5.6	IET	LOFT L2-5 (3-D)
Boiling – film	4.11	SET	ORNL THTF Tests 3.07.9B, 3.07.9N, 3.07.9W
Boiling – transition	4.10 4.11 4.12	SET SET SET	Bennett Heated Tube Tests 5358, 5294 and 5394 ORNL THTF Tests 3.07.9B, 3.07.9N, 3.07.9W Royal Institute of Technology Tube Test 261
Condensation – interfacial	4.14 4.16	SET SET	FLECHT-SEASET Test 31701 UPTF Downcomer Countercurrent Flow Test 6, Run 131
Entrainment / deentrainment	3.5 4.14 4.16	PT SET SET	Bubbling steam through liquid FLECHT-SEASET Test 31701 UPTF Downcomer Countercurrent Flow Test 6, Run 131
Evaporation – interfacial	4.1 4.8 4.9	SET SET SET	Edwards' Pipe GE Level Swell, 1 ft. Test 1004-3 GE Level Swell, 4 ft. Test 5801-15
Flashing – interfacial	4.1 4.3 4.4	SET SET SET	Edwards' Pipe Marviken Test 22 Marviken Test 24
Flow – countercurrent	3.7 3.8 4.15 4.14 4.16	PT PT SET SET SET	Gravity wave 1-D Gravity wave 3-D Dukler air-water flooding FLECHT-SEASET Test 31701 UPTF Downcomer Countercurrent Flow Test 6, Run 131
Flow - critical	4.1 4.2 4.3 4.4 4.5 4.6	SET SET SET SET SET SET	Edwards' Pipe Marviken Test 21 Marviken Test 22 Marviken Test 24 Marviken Test Jet Impinging Test 11 Moby Dick air-water
Flow – multidimensional	3.3 3.12 3.13 3.14	PT PT PT PT	Water over Steam (3-D) Pure radial symmetric flow (3-D) Rigid body rotation (3-D) R-theta symmetric flow (3-D)
Heat conductance – fuel-clad gap	LB LOCA IET See Table 3		
Heat transfer – forced convection to vapor	LB LOCA IET See Table 3		
Heat transfer – stored energy release	LB LOCA IET See Table 3		
Interfacial shear	4.8	SET	GE Level Swell, 1 ft. Test 1004-3

Phenomena	Vol IV Sec. #	Test Type	Test Name
	4.9 4.5	SET SET	GE Level Swell, 4 ft. Test 5801-15 Marviken Test JIT 11
Level	3.4 3.5 3.6 4.13 4.14 4.8 4.9	PT PT PT SET SET SET SET	Fill-Drain Bubbling steam through liquid Manometer FLECHT-SEASET Test 31504 FLECHT-SEASET Test 31701 GE Level Swell, 1 ft. Test 1004-3 GE Level Swell, 4 ft. Test 5801-15
Noncondensable effects	3.6 4.16	PT SET	Manometer UPTF Downcomer Countercurrent Flow Test 6, Run 131
Oscillations	3.6	PT	Manometer
Power-decay heat	3.10	PT	Core power
Pump – performance, including degradation	4.21	SET	Full-scale reactor coolant pump

Table 2. SB LOCA Phenomena – Model Validation

Phenomena	Manual Vol/Sec	Test Type	Test Name
Condensation – fluid to surface	4.17	SET	MIT Pressurizer Test ST4
Condensation – interfacial	4.13 4.14	SET SET	FLECHT-SEASET Test 31504 FLECHT-SEASET Test 31701
Entrainment / deentrainment	3.5 4.14 4.16	PT SET SET	Bubbling steam through liquid FLECHT-SEASET Test 31701 UPTF Downcomer Countercurrent Flow Test 6, Run 131
Flashing – interfacial	4.1 4.3 4.4	SET SET SET	Edwards' Pipe Marviken Test 22 Marviken Test 24
Flow regime – break inlet	SB LOCA IET See Table 3		
Flow – countercurrent	3.7 3.8 4.15 4.13 4.14	PT PT SET SET SET	Gravity wave 1-D Gravity wave 3-D Dukler air-water flooding FLECHT-SEASET Test 31504 FLECHT-SEASET Test 31701
Flow – critical	4.1 4.2 4.3 4.4 4.5 4.6	SET SET SET SET SET SET	Edwards' Pipe Marviken Test 21 Marviken Test 22 Marviken Test 24 Marviken Test JIT 11 Moby Dick air-water
Flow – gap	SB LOCA IET See Table 3		
Heat transfer – post-CHF	4.12 [1] VII- 2.3.1.4	SET IA[2] SET SET	Royal Institute of Technology Tube Test 261 Royal Institute of Technology Tube Test
Interfacial shear	4.8 4.9 4.5 [1] VII-2.3.1.5 [1] VII-2.3.2.2 & 5	SET SET SET IA-SET IA: IET	GE Level Swell, 1 ft. Test 1004-3 GE Level Swell, 4 ft. Test 5801-15 Marviken Test JIT 11 Kreisingt test facility BETHSY 6-inch cold leg break
Level	3.5 4.13 4.14 4.8 4.9	PT SET SET SET SET	Bubbling steam through liquid FLECHT-SEASET Test 31504 FLECHT-SEASET Test 31701 GE Level Swell, 1 ft. Test 1004-3 GE Level Swell, 4 ft. Test 5801-15
Oxidation	See reference [2] and [3]		
Power – 3D distribution	Not currently identified		
Power – decay heat	3.10	PT	Core power
Power – local peaking (fuel rod)	Not currently identified		

[2] IA: Independent Evaluation

Phenomena	Manual Vol/Sec	Test Type	Test Name
Pressure drop	[1] VII-2.3.1.2	IA: SET	Low Flow and Natural Circulation Experiment at the WSRC
Rewet	5.2 5.4 5.5 5.6	IET IET IET IET	ROSA-IV Test SB-CL-18 LOBI Test A1-04R LOFT Experiment L2-5 (1-D) LOFT Experiment L2-5 (3-D)
Stratification - horizontal	[4]	IET	2-in LOCA at Krško Nuclear Power Plant

Table 3. IET Benchmark

Test	Case	Experiment Type	Phenomena
IET III-5.4	LOBI Test A1-04R	LB LOCA	Blowdown phase of a CL LB LOCA in a PWR
IET III-5.5	LOFT L2-5 (1-D)	LB LOCA	Double-ended CL LB LOCA
IET III-5.6	LOFT L2-5 (3-D)	LB LOCA	Double-ended CL LB LOCA
IET III-5.1	LOFT L3-7	1-in. SB LOCA	1-in. CL SB LOCA
IET III-5.2	ROSA-IV Test SB-CL-18	6-in. SB LOCA	6-in. (5%) CL SB LOCA
IET III-5.3	Semiscale NC S-NC-1	Loop natural circulation	Steady-state single-phase NC
IET III-5.3	Semiscale NC S-NC-2	Loop natural circulation	Steady state single-phase, two-phase, and reflux natural circulation
IET III-5.3	Semiscale NC S-NC-3	Loop natural circulation	the effective heat transfer area during two-phase loop natural circulation.
IET III-5.3	Semiscale NC S-NC-10	Loop natural circulation	Steady state single-phase, two-phase, and reflux natural circulation
[1] IET-IA (VII-3.3.2.2) R5M3.1	PMK-2 PMK-2 7.4% Cold Leg Break Experiment (SPE-4)	SB LOCA	SBLOCA in the cold leg (7.4% cold leg break)
[1] IET-IA (VII-3.3.2.3) R5M3.1	BETHSY 6.2TC	SB LOCA	6 inch cold leg break (5% break area
[1] IET-IA (VII-3.3.2.4) R5M3.1	LOFT L3-6.	SB LOCA	offtake pipe connected at a right angle to the cold leg of the active loop. The break orifice had a break area corresponding to a break diameter of 4 inches
[1] IET-IA (VII-2.3.2.2 & 2.3.2.5) R5M3.0	BETHSY Test 9.1b/ISP27 6.2 TC	SBLOCA	0.5% Cold Leg Break 5.0% (6 inch) cold leg break
[1] IET-IA (VII-2.3.2.4) R5M3.0	BETHSY 4.1a-TC and 5.1a.	Natural convection	single-phase natural circulation, two-phase natural circulation, and reflux at various secondary conditions
[1] IET-IA (VII-2.3.2.3) R5M3.0	LSTF SB-CL-18	SB LOCA	5% CL SB LOCA
[1] IET-IA (VII-2.3.2.6) R5M3.0	Semiscale S-NH-1	SB LOCA	0.5% small break loss-of-coolant accident in the cold leg

References:

[1] RELAP5/Mod3 code manual, Vol. 7: Summaries and reviews of independent code assessment reports, NUREG/CR-5535, INEL-95/0174

[2]. Thomas K.S. Liang, Huan-Jen Hung, Chin-Jang Chang, Lance Wang, "Development of LOCA Calculation Capability with RELAP5-3D in Accordance with the Evaluation Model Methodology", Icone-9, 2001.

[3]. Thomas K. S. Liang, Development of an Appendix K Version of RELAP5-3D and Associated Deterministic-Realistic Hybrid Methodology for LOCA Licensing Analysis, http://www.intechopen.com/

[4]. I. Parzer, B. Mavko, Analysis of RELAP5/MOD3.3 Prediction of 2-Inch Loss-of-Coolant Accident at Krŝko Nuclear Power Plant, NUREG/IA-0222.

SNPB-3-14

Simplifying and Averaging

Please provide a summary of the key simplifying and averaging assumptions used in the generation of the mathematical models used in the L TCC EM and demonstrate that they are appropriate for the accident scenarios being modeled.
Criterion 3.5 Reference SRP, III.3b

STP Response:

The RELAP5-3D code manual (Volume I) provides a detailed description of the field equations (eight state equations) and the simplification and averaging assumptions used in solving these equations in space and time [1].

The RELAP5-3D hydrodynamic model is a transient, two-fluid model for flow of a two-phase vapor/gas-liquid mixture that can contain noncondensable components in the vapor/gas phase and/or a soluble component (i.e., boron) in the liquid phase. A one-dimensional as well as a multi-dimensional hydrodynamic model is included in the code, although only one-dimensional components are used in this EM. The two-fluid equations of motion that are used as the basis for the RELAP5-3D hydrodynamic model are formulated in terms of volume and time-averaged parameters of the flow. A detailed description of the equations formulation is included in Section 3.1 of Volume 1 [1]. Phenomena that depend upon transverse gradients, such as friction and heat transfer, are formulated in terms of the bulk properties using empirical transfer coefficient formulations. In situations where transverse gradients cannot be represented within the framework of empirical transfer coefficients, such as subcooled boiling, special purpose models developed for the particular situation are employed.

The set of basic equations are described in Section 3.1.1 of Volume 1 [1]; and differencing (finite differences) to a more convenient set of differential equations upon which to base the numerical scheme are described in Section 3.1.2 [1].

The system model is solved numerically using a semi-implicit finite-difference technique. The numerical technique is based on replacing the system of differential equations with a system of finite difference equations partially implicit in time. The terms evaluated implicitly are identified as the scheme is developed. In all cases, the implicit terms are formulated to be linear in the dependent variables at new time; a linear time-advancement matrix is solved by direct inversion using the default border-profile lower upper (BPLU). The difference equations are based on the concept of a control volume (or mesh cell, or node) in which mass and energy are conserved by equating accumulation to the rate of mass and energy in through the cell boundaries minus the rate of mass and energy out through the cell boundaries plus the source terms. The velocities at boundaries are most conveniently defined through use of momentum control volumes (cells) centered on the mass and energy cell boundaries. The scalar properties (pressure, specific internal energies, and void fraction) of the flow are defined at cell centers, and vector quantities (velocities) are defined on the cell boundaries.

A detailed description of the semi-implicit numerical method adopted, the guidelines followed in developing the numerical approximations, and the time advancement for the semi-implicit scheme is reported in Sections 3.1.3 and 3.1.4 [1].

Reference:

[1]. RELAP5-3D Code Manual, Vol. I "Code Structure, System Models and Solution Methods". INEEL-EXT-98-00834, Revision 4.1, September 2013.

SNPB-3-16

Single Version of the Evaluation Model

Please confirm that a single version of the EM was used during the simulations of the given accident scenarios. This includes confirming that the code version was frozen and the manner for calculating or obtaining inputs did not change.
Criterion 4.1 Reference: SRP III.d

STP Response:

Yes, a single version of the EM was used. Code version was controlled in accordance with 0PGP07-ZA-0014 "Software Quality Assurance Program" and the manner of controlling inputs did not change.

SNPB-3-19

Initial Test Cases

Please provide a summary of the assessment cases performed in order to demonstrate that RELAP5-3D has been installed and is being used appropriately.
Criterion 4.7 Reference: SRP, III.3d

STP Response:

The RELAP5-3D software has been installed and currently is currently being applied in conformance with the South Texas Project Electric Generating Station Software Quality Assurance Program 0PGP07-ZA-0014 revision 10 [1]. A RELAP5-3D SQA package (per procedure) has been developed using the guidance of paragraph 6.3 of [1]. The SQA documentation [2] has been developed per Addendum 4 of the procedure (Procured Software Developed by Industry Regulators or Industry Organizations). Addendum 4 requires the preparation of a test plan following the guidelines included in Addendum 12, and a test case or test report formatted per Addendum 13. The SQA package [2] includes a summary of the test cases performed to demonstrate that the RELAP-3D software version has been installed and is being used appropriately.

The table below shows the list of run-time environment (RTE) test problems included in the SQA package.

Table 1. SQA Package - RTE Test Problems

Case #	Input Name	Description
1	3dflow.i	Basic one dimensional flow test for implementing 3-D capabilities
2	ans05.i	Long term decay heat study with Proposed 2005 ANS Standard data (ans05-4)
3	ans79.i	Long term decay heat study with Proposed 1979 ANS Standard data (ans79-3)
4	edhtrk.i	Pipe blowdown plus heat structures coupled to the pipe, a heat structure with a simple analytic solution, reactor kinetics and control components with a few trips
5	edhtrkn.i	Similar to edhtrk.i with nearly implicit advancement used instead of semi-implicit advancement
6	edhtrt.i	Similar to edhtrk.i with time dependent actinide and fission concentration as a function of power history
7	edrst.i	Restart file to check similarity with the results of edhtrk.i
8	fldrn2.i	Transient file for analyzing water replacing steam in control volume
9	refbun.i	LBLOCA with reflood
10	todcnd.i	5 pipe cells with a hot wall (2 separate heat structures)
11	typ1200n2.i	Simulation of a four loop pressurized water reactor undergoing a small break. Timestep card control word is 15.

References:

[1]. 0PGP07-ZA-0014 “Software Quality Assurance Program” STI 33856275 rev.10, 04/17/2014.

[2] RELAP5-3D Version 4.1.3 Software Quality Assurance Package, Rev.0, STI 34280651, FSUG File No. D60.

Attachment 7

Affidavit for Withholding for Response to RAI-18

ALION
SCIENCE AND TECHNOLOGY

ALION Science & Technology

AFFIDAVIT

We, Dominic Muñoz, Project Manager and Martin Rozboril, Jr. Assistant Vice President Division Manager (AVPDM) state as follows:

(1) We, Dominic Muñoz, Project Manager, and Martin Rozboril, Jr. AVPDM, Nuclear Services, ALION Science & Technology ("Alion") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in all revisions of ALION Science & Technology report "Erosion Testing of Small Pieces of Low Density Fiberglass Debris-Test Report," ALION-REP-ALION-I006-04, with the latest revision to date, Rev. 1, dated November 17, 20011. Information from this report was used to support the South Texas Project January 2016 follow-up to 2009 RAI-18. Two versions of the follow-up to 2009 RAI-18 are being provided to the NRC. The first version of follow-up RAI-18 with information from the erosion report, ALION-REP-ALION-I006-04 included for NRC technical review, and the second version of RAI-18 with proprietary information from ALION-REP-ALION-I006-04 is redacted for public release.

(3) In making this application for withholding of proprietary information of which it is the owner, Alion relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Alion's competitors without license from Alion constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future Alion customer-funded development plans and programs, resulting in potential products to Alion;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4) a, and (4) b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by Alion, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Alion, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Alion is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or their delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Alion are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The document identified in paragraph (2), above, is classified as proprietary because it contains "know-how" and "unique data" developed by Alion within our research and development programs. The development of this document, supporting methods and data constitutes a major Alion asset in this current market.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial

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harm to Alion's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Alion's comprehensive BWR/PWR GSI-191 analysis base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and experimental methodology and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, analytical and experimental costs comprise a substantial investment of time and money by Alion.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Alion's competitive advantage will be lost if its competitors are able to use the results of the Alion experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Alion would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Alion of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 6th day of May 2016.

Dominic Muñoz
Project Manager
ALION Science & Technology

Martin Rozboril, Jr.
Assistant Vice President
Division Manager, Nuclear Services
ALION Science & Technology

Attachment 8

Definitions and Acronyms

ANS	American Nuclear Society
ARL	Alden Research Laboratory
ASME	American Society of Mechanical Engineers
BA	Boric Acid
BAP	Boric Acid Precipitation
BC	Branch Connection
BEP	Best Efficiency Point
B-F	Bimetallic Welds
B-J	Single Metal Welds
BWR	Boiling Water Reactor
CAD	Computer Aided Design
CASA	Containment Accident Stochastic Analysis
CCDF	Complementary Cumulative Distribution Function or Conditional Core Damage Frequency
CCW	Component Cooling Water
CDF	Core Damage Frequency
CET	Core Exit Thermocouple(s)
CHLE	Corrosion/Head Loss Experiments
CHRS	Containment Heat Removal System
CLB	Cold Leg Break or Current Licensing Basis
CRMP	Configuration Risk Management Program
CS	Containment Spray
CSHL	Clean Strainer Head Loss
CSS	Containment Spray System (same as CS)
CVCS	Chemical Volume Control System
DBA	Design Basis Accident
DBD	Design Basis Document
D&C	Design and Construction Defects
DEGB	Double Ended Guillotine Break
DID	Defense in Depth
DM	Degradation Mechanism
ECC	Emergency Core Cooling (same as ECCS)
ECCS	Emergency Core Cooling System
ECWS	Essential Cooling Water System (also ECW)
EM	Evaluation Model
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure(s)
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ESF	Engineered Safety Feature
FA	Fuel Assembly(s)
FHB	Fuel Handling Building
GDC	General Design Criterion(ia)
GL	Generic Letter
GSI	Generic Safety Issue
HHSI	High Head Safety Injection (ECCS Subsystem)
HLB	Hot Leg Break
HTVL	High Temperature Vertical Loop
HLSO	Hot Leg Switchover
ID	Inside Diameter
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-Service Inspection
LAR	License Amendment Request
LBB	Leak Before Break
LBLOCA	Large Break Loss of Coolant Accident
LCO	Limiting Condition for Operability
LDFG	Low Density Fiberglass
LERF	Large Early Release Frequency
LHS	Latin Hypercube Sampling
LHSI	Low Head Safety Injection (ECCS Subsystem)
LOCA	Loss of Coolant Accident
LOOP/LOSP	Loss of Off Site Power
LTCC	Long Term Core Cooling
MAAP	Modular Accident Analysis Program
MAB/MEAB	Mechanical Auxiliary Building or Mechanical Electrical Auxiliary Building
MBLOCA	Medium Break Loss of Coolant Accident
NIST	National Institute of Standards and Technology
NLHS	Non-uniform Latin Hypercube Sampling

Definitions and Acronyms

NPSH	Net Positive Suction Head, (NPSHA – available, NPSHR – required)
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
PCI	Performance Contracting, Inc.
PCT	Peak Clad Temperature
PDF	Probability Density Function
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owner's Group
PWSCC	Primary Water Stress Corrosion Cracking
QDPS	Qualified Display Processing System
RAI	Request for Additional Information
RCB	Reactor Containment Building
RCFC	Reactor Containment Fan Cooler
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RI-ISI	Risk-Informed In-Service Inspection
RMI	Reflective Metal Insulation
RMTS	Risk Managed Technical Specifications
RoverD	Risk over Deterministic Methodology
RVWL	Reactor Vessel Water Level
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SC	Stress Corrosion
SI/SIS	Safety Injection, Safety Injection System (same as ECCS)
SIR	Safety Injection and Recirculation
SR	Surveillance Requirement
SRM	Staff Requirements Memorandum
SSE	Safe Shutdown Earthquake
STP	South Texas Project
STPEGS	South Texas Project Electric Generating Station
STPNOC	STP Nuclear Operating Company
TAMU	Texas A&M University
TF	Thermal Fatigue
TGSCC	Transgranular Stress Corrosion Cracking
TS	Technical Specification(s)
TSB	Technical Specification Bases
TSC	Technical Support Center
TSP	Trisodium Phosphate
UFSAR	Updated Final Safety Analysis Report
UHSN	Upper Head Spray Nozzle
UNM	University of New Mexico
USI	Unresolved Safety Issue
UT	University of Texas (Austin)
V&V	Verification and Validation
VF	Vibration Fatigue
WCAP	Westinghouse Commercial Atomic Power
ZOI	Zone of Influence