ATTACHMENT 2

IN-VESSEL THERMAL-HYDRAULIC ANALYSIS ASSOCIATED WITH

LICENSE AMENDMENT FOR A RISK-INFORMED APPROACH TO ADDRESS

GENERIC SAFETY ISSUE 191 AND GENERIC LETTER 2004-02

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

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STP Nuclear Operating Company
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A.1 INTRODUCTION

In September 2004, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (Reference 1), as a result of the NRC evaluation of Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance." The GL 2004-02 requested that licensees for PWRs perform evaluations of the emergency core cooling system (ECCS) and the containment spray system (CSS) to assess the potential for debris entrained in the circulated containment pool to block the ECCS recirculation flow path and within the reactor and fuel assemblies following a loss-of-coolant accident (LOCA).

In December 2004, the Nuclear Energy Institute (NEI) published NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" (Reference 2), providing a method for licensees to resolve some aspects of the concerns discussed in GL 2004-02. The NRC staff's safety evaluation (SE) of NEI 04-07 (Reference 3) found that additional guidance was needed in the area of blockage in the reactor vessel in order to adequately address the downstream effects of debris that passes through the ECCS sump strainer(s).

In response to the NRC SE's conclusions on NEI 04-07, the Pressurized Water Reactor Owners Group (PWROG) sponsored development of Topical Report (TR) WCAP-16793-NP-A, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid" (WCAP-16793) (Reference 4). WCAP-16793 provided a methodology to evaluate the effects of debris and chemical precipitates on core cooling when the ECCS is aligned to the containment sump. The objective of WCAP-16793 was to provide guidance on demonstrating that long-term core cooling (LTCC) would be maintained following a LOCA to satisfy the requirements of Title 10 of the *Code of Federal Regulations* Section 50.46 (10 CFR 50.46). WCAP-16793, Revision 2 (Reference 4), was approved with limitations and restrictions specified in the incorporated NRC SE (Reference 5).

On November 21, 2007, the NRC issued "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses" (References 6 and 7), for guidance to licensees preparing supplemental responses to GL 2004-02. The Revised Content Guide provided information to licensees on how to evaluate the effects of debris carried downstream of the containment sump screen and into the reactor vessel, and show that the in-vessel effects evaluation is consistent with, or bounded by, the industry generic guidance (WCAP-16793), as modified by NRC staff comments on that document. The Revised Content Guide also identified NRC staff information needs on the application of the methods, the exceptions to WCAP-16793, and the summary of the evaluation of those areas.

By letter dated March 8, 2005 (Reference 8), the STP Nuclear Operating Company (STPNOC) submitted its 90-day response to GL 2004-02 for the South Texas Project, Units 1 and 2 (STP).

Table 1[1] contains the key correspondence which specifically addresses the in-vessel thermal-hydraulic effects of debris on LTCC for STP. This table is not a complete list of material submitted on the in-vessel thermal-hydraulic effects review, but it is a list of documents relevant to the current approach.

Table 1: List of Correspondence Related to Thermal-Hydraulic Analysis

Author	Document	Date	Reference
NRC	GL 2004-02	September 13, 2004	1
NEI	PWR Sump Performance Evaluation Model (EM)	December 2004	2
NEI	SE for PWR Sump Performance EM	December 2004	3
PWROG	WCAP-16793-NP, Revision 2	July 2013	4
NRC	Revised Guidance Letter	November 21, 2007	7
NRC	Revised Guidance on GL 2004-02	November 2007	6
STPNOC	90-Day Response to GL 2004-02	March 8, 2005	8
INL	RELAP5-3D Manuals	July 2014	9
STPNOC	Reactor Coolant System (RCS) Thermal Hydraulics (TH)	August 20, 2015	10
NRC	Draft Quality Assurance (QA) Request for Additional Information (RAI)	October 21, 2015	11
NRC	Draft TH RAIs	December 11, 2015	12
NRC	RAI – Round 3	April 11, 2016	13
NRC	Methodology Audit Report	April 13, 2016	14
NRC	QA Audit Report	May 11, 2016	15
STPNOC	Part 1 of Response to RAI Round 3	May 11, 2016	16
STPNOC	Part 2 of Response to RAI Round 3	June 16, 2016	17
STPNOC	Part 3 of Response to RAI Round 3	July 21, 2016	18
STPNOC	RAI Round 3 Response Supplement	October 20, 2016	19
STPNOC	RAI Round 3 Response Supplement	November 9, 2016	20

The NRC staff issued an RAI specifically on the issue of in-vessel thermal hydraulic effects. This was considered RAI, Round 3; however, due to the STPNOC methodology change, previous in-vessel thermal-hydraulic RAI questions were superseded. See the NRC staff's letter dated December 12, 2016, regarding closeout of RAI questions no longer applicable to the GSI-191 review (Reference 21).

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[1] For clarity, correspondence from the NRC is highlighted in gray. Due to the nature of this pilot program, much of the documentation submitted was changed due to a change in the approach by STPNOC.

General information for each RAI question is given in Table 2 including the question number, topic, associated SE section, and the reference number(s) of its response.

Table 2: List of RAIs

Question	Subject	Section	Reference of Response
SNPB-3-1	Clad oxide	A.4	16, 19
SNPB-3-2	Accident scenario progression	A.4.2.1.2	18, 19
SNPB-3-3	Core bypass flow	A.4.2.1.4	16
SNPB-3-4	Important phenomena	A.4.2.1.3	16
SNPB-3-5	Debris at grid spacers	A.4.2.1.3	16, 19
SNPB-3-6	Initial and boundary conditions	A.4.2.1.4	18
SNPB-3-7	Initial and boundary conditions for long-term	A.4.2.1.4	18, 19
SNPB-3-8	Phenomena modeled	A.4.2.2.2	16
SNPB-3-9	Reference and limits of closure relationships	A.4.2.2.3	17
SNPB-3-10	User manual	A.4.2.2.4	17
SNPB-3-11	Modeling of important phenomena	A.4.2.3.2	16
SNPB-3-12	Field equations	A.4.2.3.3	16
SNPB-3-13	Validation of closure relationships	A.4.2.3.4	16, 19
SNPB-3-14	Simplifications and averaging	A.4.2.3.5	16
SNPB-3-15	Level of detail	A.4.2.3.6	18
SNPB-3-16	Single version of the EM	A.4.2.4.1	16
SNPB-3-17	Validation of the EM	A.4.2.4.2	18, 19
SNPB-3-18	Mesh size sensitivity	A.4.2.4.7	18, 19
SNPB-3-19	Initial test cases	A.4.2.6.1	16, 19
SNPB-3-20	Specific sensitivity studies	A.4.2.4.7	18, 19, 20
SNPB-3-21	Important sources of uncertainty	A.4.2.5.1	18
SNPB-3-22	Uncertainties and design margin	A.4.2.5.1	18, 19
SNPB-3-23	Quality assurance program for the EM	A.4.2.6.1	18
SNPB-3-24	Input verification	A.4.2.6.1	18
SNPB-3-25	Proper convergence	A.4.2.6.1	18
SNPB-3-26	Non-physical results	A.4.2.6.1	18
SNPB-3-27	Realistic results	A.4.2.6.1	18
SNPB-3-28	Boundary conditions as prescribed	A.4.2.6.1	18
SNPB-3-29	Thoroughly understood results	A.4.2.6.1	18
SNPB-3-30	Quality assurance program documentation	A.4.2.6.2	18
SNPB-3-31	Independent peer review	A.4.2.6.2	18
SNPB-3-32	Important sources of uncertainty	A.4.2.5.1	18

A.2 REGULATORY EVALUATION

Generic Letter (GL) 2004-02 requested that holders of operating licenses for PWRs perform evaluations of the ECCS and the CSS recirculation functions considering the effects of debris following a LOCA. These evaluations are to include the potential for debris blockage at flow restrictions within the ECCS recirculation flow path downstream of the sump strainer, including potential blockage at fuel assembly inlet debris strainers and other potential flow restrictions, such as the fuel assembly spacer grids. Debris blockage at these locations has the potential to impede or prevent the flow of coolant to the reactor core, potentially leading to inadequate LTCC.

The acceptance criteria for the performance of a nuclear reactor core following a LOCA are found in 10 CFR 50.46. The regulations under 10 CFR 50.46(a)(1)(i) state that the ECCS cooling performance must be calculated in accordance with an acceptable EM. This EM is defined in 10 CFR Subsection 50.46(c)(2):

> An *evaluation model* is defined as the calculational framework for evaluating behavior of a system of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

The EM must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA, make comparisons to applicable experimental data, and must identify, quantify, and assess uncertainties in the analysis method and inputs.

The acceptance criterion dealing with the long-term cooling phase of the accident recovery is in 10 CFR 50.46(b)(5), which reads as follows:

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

As stated in Section A.1, “Introduction,” of this In-Vessel Thermal-Hydraulic Analysis, the NRC staff endorsed WCAP-16793 as an acceptable evaluation method to show compliance with the long-term core cooling requirements in 10 CFR 50.46 considering the effects of debris in the ECCS recirculating fluid. In the SE for WCAP-16793, the NRC staff specified that meeting 10 CFR 50.46(b)(5) requires: (1) acceptance criteria for LTCC once the core has quenched and reflooded, and (2) the mission time that should be used in evaluating debris ingestion effects on the reactor fuel.

To summarize, long-term cooling capability must be provided despite potential challenges from chemical effects (e.g., boron precipitation,[2] interaction of debris with chemicals from coatings) or physical effects (e.g., debris), as demonstrated by no significant increase in calculated peak cladding temperature (PCT). After quench and reflood, moderate increases in cladding temperature, on the order of 200 to 400 degrees Fahrenheit (°F) could be acceptable, if appropriately justified. In addition, adequate core cooling performance during the ECCS mission time is demonstrated when bulk and local temperatures are shown to be stable or continuously decreasing with the additional assurance that any debris entrained in the cooling water supply would not be capable of affecting the stable heat removal mechanism due to sump strainer clogging or downstream effects.

In WCAP-16793, the acceptance criteria for LTCC following core quench and reflooding are given as the following:

1. The maximum clad temperature shall not exceed 800 °F following core quench and reflooding

2. The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The acceptance criteria do not represent, nor are they intended to be, new or additional LTCC requirements beyond the requirements in 10 CFR 50.46. Instead, they allow demonstration that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal. The 800 °F temperature was determined based on autoclave data that demonstrated oxidation and hydrogen pickup to be acceptable at and below 800 °F. A discussion of the technical basis for the 800 °F temperature is given in Appendix A of WCAP-16793. The 0.050-inch limit for oxide plus deposits was selected to preclude the formation of deposits that would bridge the space between adjacent rods and block flow between fuel channels.

The licensee performed a number of simulations to demonstrate that these criteria have been met, and the NRC staff reviewed these simulations. To assure the quality and uniformity of NRC staff reviews, the NRC created NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP) (Reference 22), to guide the staff in performing its reviews. Regulatory guidance for the review of design basis accident evaluation methodologies is provided in Section 15.0.2 of the SRP, “Review of Transient and Accident Analysis Methods” (Reference 23). Similar guidance is also set forth for the industry in Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” December 2005 (Reference 24).

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[2] Section 8 of WCAP-16793 states that the effects of boron precipitation on LTCC are being addressed by the PWROG in a separate program. Refer to SE Section 3.4.2.8, “Impact of Debris,” for a description of the program and the NRC staff evaluation.

A.3 TECHNICAL EVALUATION OVERVIEW

A.3.1 NRC Staff Evaluation Method

In order to demonstrate that the impacts of in-vessel debris have been appropriately captured, STPNOC stated it performed analyses confirming that the following acceptance criteria, set forth during the NRC's review of WCAP-16793, have been satisfied:

1. The maximum clad temperature shall not exceed 800 °F following core quench and reflooding

2. The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The purpose of the NRC staff's review was to determine if there is adequate core cooling during the long-term period following a LOCA (i.e., after reflood and core quench) such that the maximum clad temperature would not exceed 800 °F and the cladding oxide would not exceed 0.050 inches.

The licensee's submittal proposed to demonstrate that both of these acceptance criteria were satisfied using scientific computer simulations of the LTCC phase of the LOCA. STPNOC's computer simulations for the LTCC phase of the accident used the RELAP5-3D platform, an EM that has not previously been reviewed and approved by the NRC staff for use in this manner. Thus, the NRC staff's technical evaluation focused on determining whether the EM, when used in the manner prescribed by STPNOC, resulted in appropriate simulations of the given scenario such that the NRC staff would have confidence in the outcomes. Then, the outcomes were compared against the acceptance criteria in WCAP-16793.

It is important to note that the NRC staff's review of the EM was "simulation" focused and not "code" focused. Thus, in its scientific computer simulation review,[3] the NRC staff was concerned with determining if the simulation results were trustworthy and how trustworthy they needed to be for the intended purpose.

This distinction is important, as the guidance typically used by the NRC staff for reviews like this (e.g., SRP Section 15.0.2, Regulatory Guide 1.203) focuses on reviewing the EM. When such reviews are performed, a plant-specific simulation is not usually reviewed since the goal is to evaluate the model for a variety of future uses by the nuclear industry.

Thus, the NRC staff did not review the STPNOC LTCC EM for general use. The NRC staff's review focused on those simulations performed by STPNOC to demonstrate adequate LTCC capability in the presence of debris for STP. In that context, the NRC staff reviewed the simulations produced by the STP LTCC EM and determined whether there is reasonable assurance that those simulations are adequate representations of the STP LTCC scenario.

[3] The topic of scientific computer simulation review is discussed in more detail in Kaizer, Heller, and Oberkampf (Reference 25).

A.3.2 Scope of the Review

The licensee deterministically modeled the phenomena during the long-term phase following small and medium hot-leg breaks using RELAP5-3D. RELAP5-3D has never been submitted for NRC review and approval, and this review is not intended to provide a generic review. The NRC staff restricted its review to focus on whether RELAP5-3D is an acceptable EM to predict the phenomena occurring during the long-term cooling phase following a LOCA at STP. This review focused on ensuring that the acceptance criteria were met for STP's current licensed operating conditions. Use of RELAP5-3D by other plants, for other purposes, or with different key inputs would require prior review and approval by the NRC.

RELAP5-3D was used by the licensee to predict the blowdown, refill, and reflood phases of the LOCA; however, the NRC staff reviewed those phases only to determine if they would provide a reasonable estimate of the initial condition for the long-term phase and only for the specific scenarios considered by STPNOC. Thus, the ability of RELAP5-3D to accurately simulate phenomena impacting the figures of merit[4] during blowdown, refill, and reflood is beyond the scope of this review for those phenomena which are not important for calculating the initial conditions of the long-term phase. Additionally, the ability of the EM to accurately simulate this scenario for other inputs is beyond the scope of this review.

As stated above, the review for in-vessel thermal-hydraulic effects on ECCS and CSS considering the impacts of debris, focused on RELAP5-3D for small and medium hot-leg breaks. The licensee used alternative methods to evaluate cold-leg breaks, and large hot-leg breaks for the in-vessel thermal-hydraulic evaluations. A summary of these different methods is shown below in Table 3, "Accident Scenarios."

The NRC staff discussed the licensee's use of the RoverD methodology to evaluate the impacts of debris on cold-leg breaks in Enclosure 3, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 212 and 198 to Facility Operating License Nos. NPF-76 and NPF-80" (Main SE). Main SE Section 3.4.2.7, "Debris Transport," provides the NRC staff's evaluation of the licensee's analysis showing that, for all sizes of cold-leg breaks, 7 grams per fuel assembly (gm/FA) was the maximum that could reach the core. The NRC staff found this analysis acceptable because this debris amount met the WCAP-16793 criteria. The SE for WCAP-16793 provides the NRC staff's conclusion that debris quantities of less than 15 gm/FA will not result in core inlet blockage compromising core coolant flow.

For large hot-leg breaks, in order to simplify the thermal-hydraulic analysis, the licensee assumed that core damage would result for any hot-leg break greater than 16 inches, thus, a risk-informed assessment (probabilistic risk assessment) was used. Table 3 provides an overview of the different break scenarios and how they are treated, and shows that only small and medium hot-leg breaks are considered in this in-vessel thermal-hydraulic analysis.

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[4] A figure of merit is calculated during the analysis and is a primary variable for drawing conclusions about the analysis. An analysis may have several figures of merit.

Table 3: Accident Scenarios

	Hot-leg Break	Cold-leg Break
Small Break	LTCC EM Analysis (In-Vessel Thermal-Hydraulic Analysis)	RoverD Analysis (Main SE)
Medium Break	LTCC EM Analysis (In-Vessel Thermal-Hydraulic Analysis)	RoverD Analysis (Main SE)
Large Break (> 16")	Risk-informed Analysis (Main SE)	RoverD Analysis (Main SE)

A.3.3 System Description

The facility is a four-loop PWR with a Westinghouse-designed nuclear steam supply system. During a LOCA, regardless of break location, the ECCS pumps are aligned to inject borated water into three of the four RCS cold-legs. The source for this water is from stored locations, like the refueling water storage tank (RWST). Water that is pumped into the reactor vessel by the ECCS is subsequently discharged through the break into the containment where it collects in the containment building basement and the ECCS sumps. When the stored water supply is exhausted, the CSS and ECCS are realigned to draw coolant from the containment sump. The coolant discharged from the RCS and from the CSS is then circulated back into the RCS to provide for continued LTCC without the need for additional cooling water.

There are two separate categories of LOCAs depending on whether the break is upstream or downstream of the core (cold-leg side or hot-leg side, respectively). The quantity of debris carried into the core, the quantity of debris deposited on fuel cladding surfaces, and the head available to drive coolant into the core are greatly dependent upon the location of the pipe break. The effect of the different break locations is discussed in WCAP-16793.

In the event of a hot-leg break, the coolant pumped into the cold-leg is forced into the reactor pressure vessel, down the downcomer and up through the reactor core toward the break. During the LTCC period, core flow, plus a small amount of core bypass flow, is equal to the total ECCS flow delivered to the cold-leg. However, this ECCS flow, and thus the flow through the core, may vary depending on the number of operating ECCS pumps.

In the event of a cold-leg break, ECCS coolant injected into the failed loop will exit the RCS through the break while coolant injected into the intact loop will enter the downcomer annulus. This ensures that the downcomer is filled, at minimum, to the bottom of the cold-leg nozzle. During a cold-leg break, once the core has been recovered, the flow of coolant entering the core is that required to replenish boil-off (i.e., less than 1.5 gallons per minute per fuel assembly) (Reference 4). The excess coolant flows around the downcomer annulus and exits the reactor pressure vessel through the failed pipe. Therefore, the LTCC period following a cold-leg break represents a minimum core flow condition.

Debris build-up at the core inlet and in the fuel assemblies following either a hot-leg or cold-leg break could impact heat transfer from the fuel cladding and could add to the resistance in the core inlet that must be overcome to provide adequate cooling flow into the core.

A.4 TECHNICAL EVALUATION

A.4.1 Clad Oxide Calculation

Clad Oxidation
The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.
Acceptance Basis from WCAP-16793-NP, Revision 2

The clad oxide thickness limit provided in WCAP-16793 prevents deposits from filling the space between adjacent fuel rods and blocking coolant flow.

The licensee provided justification that it met this criterion in response to SNPB-3-1 (Reference 16). In that response (Reference 27), the licensee referenced responses to previous questions 31 and 36 which referred to a LOCA Deposition Model (LOCADM) analysis demonstrating that the total thickness of deposition remained less than 50 mils. However, it was not clear to the NRC staff that this LOCADM analysis was applicable to STP, as it appeared to assume a much lower fiber loading than could be justified for STP.

Therefore, the licensee supplemented its response to SNPB-3-1 (Reference 19) and provided additional details on the STP-specific analysis performed. The analysis performed with LOCADM assumed that 91 grams of fibrous debris per fuel assembly (gm/FA) bypassed the sump strainers. This was much greater than the amount that could be reasonably expected to bypass the strainer since the maximum amount that could bypass the strainer and still be considered a success at the strainer was less than half this amount. Because the licensee performed a plant-specific analysis that conservatively assumed a larger quantity of fiber per fuel assembly than would actually be expected for the scenarios under consideration, the NRC staff determined that LOCADM is appropriate for STP.

The licensee used LOCADM with a conservative amount of debris bypassing the sump strainers and confirmed that the thickness of clad oxide and deposits of material was less than 0.050 inches. The NRC staff determined that there is reasonable assurance that the clad oxidation will not exceed 0.050 inches in any fuel region because the licensee used LOCADM with a conservative amount of debris in the analysis. Therefore, the NRC staff concludes that the clad oxidation criterion is satisfied.

A.4.2 Long-Term Core Cooling Evaluation Model

As stated previously, the NRC staff's review of the EM was performed following the guidance of SRP Section 15.0.2. Section 15.0.2 of the SRP directs the reviewer to examine the EM, which is defined as the calculational framework for evaluating the behavior of the RCS during a postulated accident or transient, and includes the computer programs, mathematical models, assumptions, and procedures on how to treat the input and the output, as well as many other factors. Section 15.0.2 of the SRP organizes the review into the six categories shown in Table 4.

Table 4: SRP Section 15.0.2 Review Categories

Section	
A.4.2.1	Accident Scenario Identification Process
A.4.2.2	Documentation
A.4.2.3	Evaluation Model Development
A.4.2.4	Code Assessment
A.4.2.5	Uncertainty Analysis
A.4.2.6	Quality Assurance Program

In order to demonstrate that the maximum clad temperature will not exceed 800 °F following the core quench and reflood stage, STPNOC performed multiple simulations using the LTCC EM. The LTCC EM uses the RELAP5-3D computer code. The focus of the NRC staff's review was to determine if the code options, inputs, and models were appropriate and if using appropriate input parameters resulted in an accurate and conservative RELAP5-3D simulation.

The NRC staff's review focused on ensuring the LTCC EM simulations performed by STPNOC were reasonable representations of the actual thermal-hydraulic phenomenon in the STP core following a LOCA. The NRC staff's review of the EM was restricted to those simulations already performed by STPNOC, and did not consider future simulations which could be performed except for very small modifications to the initial set of three simulations. The NRC staff therefore placed limitations on the use of the LTCC EM by STPNOC to perform the analysis using alternative inputs. These limitations are discussed in Section A.4.3, "Conclusions."

A.4.2.1 Accident Scenario Identification Process

The accident scenario identification process is a structured process used to identify the key figures of merit or acceptance criteria for the accident. It is also used to identify and rank the reactor component and physical phenomena modeling requirements based on their (a) importance to acceptable modeling of the scenario and (b) impact on the figures of merit for the calculation (e.g., PCT and maximum, average cladding oxidation thicknesses).

In general, STPNOC considered six similar accident scenarios. However, only two scenarios were assessed deterministically and used its LTCC EM. The other four scenarios were addressed by other means and were not considered in this review. A summary of the accident scenarios and the SE sections where the NRC staff review is documented is shown in Table 3.

For the LTCC EM, the key figure of merit is the PCT. However, other calculated quantities (e.g., mass flow rates, pressures, heat fluxes) are important for ensuring that the resulting simulation behaves in a reasonable manner and that the resulting post-reflood PCT has been adequately simulated. Therefore, the focus of the accident scenario identification process is to describe the important phenomena in each scenario, so the EM can be evaluated in terms of its ability to model those phenomena. Table 5 provides the SRP review criteria topics and the sections providing the NRC staff's review.

Table 5: Accident Scenario Identification Process Review Categories

Accident Scenario Identification Process	
A.4.2.1.1	Structured Process
A.4.2.1.2	Accident Progression
A.4.2.1.3	Phenomena Identification and Ranking
A.4.2.1.4	Initial and Boundary Conditions

A.4.2.1.1 Structured Process

Structured Process

The process used for accident scenario identification should be a structured process.

SRP Section 15.0.2, Subsection III.3c

The licensee provided a description of the structured process used to identify and define the accident scenario in response to SNPB-3-2 (References 18 and 19) and SNPB-3-4 (Reference 16). The NRC staff's review of the STPNOC process for accident scenario identification determined that the process addressed three areas:

1. The description of the accident scenarios; evaluated by the NRC staff in Section A.4.2.1.2, "Accident Progression," of this In-Vessel Thermal-Hydraulic Analysis.

2. Identification of the important phenomena from these scenarios; evaluated by the NRC staff in Section A.4.2.1.3, "Phenomena Identification and Ranking," of this In-Vessel Thermal-Hydraulic Analysis.

3. Identification of the important aspects of the boundary conditions; evaluated by the NRC staff in Section A.4.2.1.4, "Initial and Boundary Conditions," of this In-Vessel Thermal-Hydraulic Analysis.

Because the licensee considered the accident scenario, identified the important phenomena, and identified the important boundary conditions, the NRC staff determined that the process used for the accident scenario identification is a structured process; therefore, this criterion is satisfied.

A.4.2.1.2 Accident Progression

> **Accident Progression**
>
> *The description of each accident scenario should provide a complete and accurate description of the accident progression.*
>
> SRP Section 15.0.2, Subsection III.3c

In its response to SNPB-3-2 (References 18 and 19), the licensee provided a description of the accident progression for the largest break considered deterministically as well as justifications for certain initial and boundary conditions. The justifications related to the input and boundary conditions are evaluated in Section A.4.2.1.4, "Initial and Boundary Conditions," of this In-Vessel Thermal-Hydraulic Analysis.

The main phases of the 16-inch hot-leg break: break and blowdown, refill and reflood, pre-blockage and LTCC, and core blockage and post-blockage LTCC, roughly follow those of the standard cold-leg break LOCA accident progression, with some exceptions. The phases of the 16-inch hot-leg break are described below. The following STP information should be noted for this break progression:

- Both units at STP are 4-loop Westinghouse plants, each with three independent trains of safety injection (SI).
- Safety injection flow is injected into the RCS in loops A, B, and C between the steam generator and the core in the cold-leg.
- The break is located in loop B between the core and the steam generator in the hot-leg.

<u>Phase 1 - Break and Blowdown</u>

The licensee stated that the first phase is defined to start when the hot-leg breaks and the reactor depressurizes as water is expelled from the RCS through the 16-inch break. Since the water's temperature is above the saturation temperature as the RCS blows down, depressurization causes voids to form in the RCS. However, there is substantial coolant flow through the core, since all SI flow must pass through the core to exit out of the break in the hot-leg. Because of this, the core collapsed liquid level never drops below the bottom of the core.

<u>Phase 2 - Refill and Reflood</u>

The licensee continued with the second phase stating that it starts when the core collapsed liquid level starts increasing and ends when the core is completely flooded (i.e., the collapsed liquid level reaches the top of the core). Because this is a hot-leg break (where SI must flow through the core to reach the break) rather than a cold-leg break (where the SI can bypass the core), the licensee stated that the beginning and end of this phase can be difficult to identify. During this phase, the accumulators inject and eventually the low head SI pumps are able to complete the flooding of the core.

Phase 3 - Pre-Blockage and Long-term Core Cooling

The licensee stated that the third phase is defined to start when the core is completely flooded (i.e., the collapsed liquid level has reached the top of the core) and ends when the ECCS suction switches over from the RWST to the sump. During this time, the SI flow completes filling up the reactor pressure vessel and RCS loops and starts filling up the steam generators. The steam generators, which were unable to release their stored energy to the RCS during the first three phases because of voiding on the primary side of the steam generator u-tubes, act as heat sources that must be cooled by the SI flow. In its supplemental response submitted in Reference 19, STPNOC asserted that the large quantity of secondary-side mass in the steam generators, which consists of both water and metal mass, did not interfere with the refilling of the steam generator u-tubes. In the sensitivity study STPNOC provided to support this assertion, conservatively large masses were added to the secondary side of the steam generator which more than accounted for the secondary metal mass in the steam generators. Even with these additional masses and the increased energy transferred to the SI flow, there was no impact on the PCT for the LTCC EM.

Phase 4 - Core Blockage and Post-Blockage Long-Term Core Cooling

The licensee described phase 4 as when the SI flow switches over from the RWST to the sump, thus causing any debris which accumulated in the sump and bypassed the sump strainer to be pumped into the core. The licensee assumed the time between sump switchover and core blockage is 360 seconds for this analysis. The justification for this time delay is evaluated by the NRC staff in Section A.4.2.1.4, “Initial and Boundary Conditions,” of this In-Vessel Thermal-Hydraulic Analysis.

After 360 seconds, the debris generated by the LOCA and transported to the RCS is conservatively assumed to completely block the bottom of the core, including the barrel-baffle bypass region. In reality, as the core begins to block, a portion of the SI flow would be diverted around the core and into the steam generator u-tubes. The fraction of flow diverted around the core increases as a function of core blockage until the bottom of the core is completely blocked, at which point all of the SI flow is forced through the steam generator u-tubes. However, for the simulation, this gradual blockage is not assumed and instead the licensee assumes the core is instantaneously blocked at 360 seconds following sump switchover.

Following sump switchover, STPNOC specified all SI must flow through one of the steam generators. Any SI flowing to steam generator 3 (i.e., the loop with the break) must flow in through the cold-leg of the steam generator, through the u-tubes, down the hot-leg and exit the RCS on the steam generator side of the break. Any SI flowing to steam generators 1, 2, or 4 must flow in through the cold-leg of the steam generator, through the u-tubes, down the hot-leg, and into the core to the upper plenum before it is able to flow into the hot-leg of loop 3 and out the vessel side of the break.

Following blockage, the temperature of the fluid in the core increases and boiling may occur. Fluid will then flow from the core into the upper plenum due to the change in density, caused either by a phase change to steam or by the decreasing density of liquid water as the temperature increases. This fluid, likely a two-phase mixture, will combine with the SI flowing into the upper plenum from the steam generators and some of the mixture will flow back into the core through the upper core plate. Eventually, because of the coolant added to the system by the ECCS, the mixture must flow into the hot-leg of the broken loop and exit the RCS on the vessel side of the break.

The NRC staff reviewed the licensee's description of each phase of a hot-leg break LOCA that included the key physical phenomena, the key figures of merit, and the progression of each during the phase. This description was logical and consistent with the NRC staff's experience with these phases of the event. The NRC staff concludes that the licensee provided a complete and accurate description of the accident progression; therefore, this criterion is satisfied.

A.4.2.1.3 Phenomena Identification and Ranking

> **Phenomena Identification and Ranking**
>
> *The dominant physical phenomena influencing the outcome of the accident should be correctly identified and ranked.*
>
> SRP Section 15.0.2, Subsection III.3c

The licensee provided justification for the above criterion in response to SNPB-3-4 (Reference 16). It should be noted that at the time of this response, the licensee intended to use the LTCC EM for small, medium, and large (i.e., larger than 16-inch breaks) hot-leg breaks as well as small cold-leg breaks. Therefore, the licensee provided a description of the important phenomena for those set of breaks. Only the phenomena associated with hot-leg breaks 16 inches and smaller were considered for this review since STPNOC changed its request to make all breaks larger than 16 inches to be part of the risk-informed review (the NRC staff's review of this is provided in the Main SE).

The licensee's listing of phenomena was based on work performed under the code scaling, applicability, and uncertainty (i.e., CSAU) methodology to identify the important phenomena occurring during a large break LOCA (LBLOCA) (Reference 29). The listing included a description of the important phenomena and a discussion of the phase of the accident during which each phenomenon occurred. The NRC staff found that the phenomena identified encompassed all phenomena the staff would expect to be important during a hot-leg break.

The licensee also identified those phenomena expected to be most important during the long-term cooling phase (phase 4). The NRC staff determined that the licensee's identification of both heat transfer of natural convection and the counter current flow limitation (CCFL) would be the most important phenomena during the long-term phase. However, the NRC staff also identified other phenomena which could be important during phase 4, including:

- Blockage on grid spacers in the fueled regions of the core
- Core uncovery
- Heat stored in the steam generators

<u>Blockage on grid spacers in the fueled region of the core</u>

The licensee's analysis assumed some amount of debris makes it past the strainer and blocks the bottom of the core, so it is reasonable to assume that the SI flow has the potential to carry some amount of debris into the core. This debris could cause additional blockages in the core, resulting in local heat ups. The licensee provided a justification that debris in the fueled region of the core was not an important consideration in response to SNPB-3-5 (Reference 16).

In its response, the licensee referenced a previous submittal (Reference 30). In the referenced analysis, the licensee demonstrated that the amount of crud which could be expected to be released would be less than 2 pounds. The NRC staff reviewed the licensee submittal and determined that this small amount of fine particles would have minimal impact on the flow through the grid spacers. However, though crud deposition is an important consideration, the analysis discussed by the licensee did not address the potential for fibrous debris deposition in the core.

In its response (Reference 16), the licensee also referenced WCAP-16793 to address the ability of fibrous debris to collect at grid spacers. In Section 3.4.4 of the SE for WCAP-16793, the NRC staff concluded that when the quantity of debris is within the acceptance limits specified in WCAP-16793 (e.g., 15 gm/FA), there is reasonable assurance that flow of coolant will not be impeded by debris collecting at grid spacers. This conclusion was based on the observation during testing that while fiber was deposited on grid spacers in the core, it did not impede cooling. It was not clear to the NRC staff that the same conclusion made in WCAP-16793 would be applicable to the licensee, since the expected quantity of fiber at STP is higher than what was tested in WCAP-16793.

In its supplemental response to SNPB-3-5 submitted in Reference 19, the licensee discussed the potential for debris beyond 15 gm/FA to enter the core. The licensee described the following conservatisms in its modeling approach:

- The maximum amount of fiber that is available to reach the core is estimated as a conservatively high amount of 50 gm/FA. This amount is expected to be conservative as debris greater than 50 gm/FA could only occur if the sump strainer fails.

- The core is assumed to be 100 percent blocked as soon as the in-vessel fiber amount reaches 15 gm/FA. As demonstrated by testing, a much larger fiber loading would be needed to fully block the bottom of the core provided chemicals have not arrived. Additionally, once fiber begins to block the bottom of the core, it forms a bed which acts as a filter and easily traps additional fiber. This leads to the conclusion that the amount of fiber which would be trapped at the bottom of the core would likely be well in excess of 15 gm/FA, reducing the fiber amount available for deposition in the fueled region of the core.

- Any SI flow which is discharged from the core following sump switchover would be re-filtered through the ECCS sump screens, which would reduce the amount of fiber injected into the vessel. This re-filtering is ignored in the current analysis.

- The testing which supports the 15 gm/FA was conducted in such a way that any re-filtering of the coolant through the ECCS suction strainers was also ignored. If this was accounted for, the licensee stated the actual amount of fiber would have been greatly reduced due to the SI being re-filtered through the ECCS sump strainers.

Assuming that the maximum amount of fiber entering the core would be 50 gm/FA, and 15 gm/FA would be deposited on the core support plate, 35 gm/FA would remain to be deposited in the core above the bottom grid, deposited elsewhere in the RCS, or flow out of the break. The licensee noted that only a small portion of this flow would actually enter the core, since the only SI flow to enter the core would be the flow to overcome decay heat. Much of the SI flow would exit the break, thus, the licensee estimates that only a small amount of debris, about 5 gm/FA, would flow from the upper plenum, through the upper core plate, and into the core. However, the NRC staff considers the flow dynamics to be complicated by the conservative core blockage assumptions in the analysis as discussed below.

While it is conservative to assume that the core is fully blocked once the debris entering the vessel reaches 15 gm/FA, this means that, for a hot-leg break, the only other flow path is around the steam generators, through the hot-legs, and into the upper plenum. As only a small fraction of the SI flow will be needed to replace that which is pushed out of the core due to natural convection, only a small amount of debris would actually enter the core. While that scenario is conservative in that it reduces the amount of SI flow into the core, this reduction in SI flow also reduces the amount of debris capable of entering the core, therefore, the NRC staff does not consider this scenario to be likely.

The NRC staff considers it more likely that the core will not block with 15 gm/FA deposited on the core support plate; and further, that there will be substantial flow through the barrel-baffle region, which also will not block. While the core is generally assumed to block below the core support plate, the blockage would actually occur below the first spacer. It is reasonable to assume that some amount of fiber would be deposited on these grid spacers, but it is likely that the distribution would not be uniform on all grid spacers and all grid spacers may not block. If blockage occurred uniformly on all grid spacers, it would likely not happen simultaneously across the core. Further, there is no known blockage mechanism in the barrel-baffle region as the holes are too large for the fiber to bridge across. The barrel-baffle region not only provides a flow path directly to the top of the core, it also provides the capability for coolant to be delivered into the core at various axial elevations because of the horizontal flow holes in the core baffle, which are also likely to remain unblocked by fibrous debris. Thus, there is a reasonable probability that much more of the SI flow (and hence, the associated debris) would actually enter the vessel than assumed in the “conservative” case.

The NRC staff considered both extremes wherein either (a) many of the flow channels are blocked and only a very small amount of SI flow would be expected to enter the fueled region of the core, or (b) few of the flow channels are blocked and a large amount of SI would be expected to enter the fueled region of the core. The staff finds that, given the open-lattice configuration of a PWR core which allows for multiple paths of cross flow and the analysis suggesting that the core region would be very well mixed, there is reasonable assurance that any fiber deposited in the fueled region of the core will have a minimal impact on core heat transfer. In situation (a), a minimal amount of fiber is able to flow into the core. In situation (b), a large amount of fiber will flow into the core, but this fiber will be carried by a large amount of SI flow which would aid in mixing the fluid in the core and prevent localized heating.

<u>Core Uncovery</u>

As long as the core remained covered with a two-phase mixture, then the core temperature would be kept near saturation and the heat transfer would be limited to two-phase convection. However, following LTCC, should the two-phase level drop below the top of the core, the fuel would experience a heat-up to temperatures greater than saturation, and other heat transfer regimes should be considered as highly-ranked phenomena. In the license’s base case, the two-phase level never dropped below the top of the core. However, the base case assumed a cosine axial power shape. In response to SNPB 3-20 (Reference 18), the licensee demonstrated that, following sump switchover and core blockage, both the top-skewed and bottom-skewed axial power shapes would result in higher PCTs than the cosine axial power shape. Further, in both instances, the PCT rose above the saturation temperature in the core. This suggested that the limiting case should assume a top-skewed axial power shape, as the sensitivity study showed that it reached the highest post-reflood PCT. Additionally, in that study, the two-phase level dropped below the top of the core, making heat transfer regimes other than the ones initially considered by the licensee highly ranked phenomena. It should be noted that in all cases, the PCT did not remain above saturation temperature for an extended period (i.e., it was a transient effect which did not last long enough to significantly impact temperature) of time, and the cladding temperature remained well below the limit of 800 °F.

In a supplemental response to SNPB 3-20 (References 19 and 20), the licensee provided additional details to support the assumption that the core would not experience temperatures above saturation, and, therefore, consideration of other heat transfer phenomena was not necessary. The main argument provided by the licensee was that the assumption of complete core blockage was conservative. Testing conducted by Westinghouse[5] demonstrated that the holes of the barrel-baffle region would not block due to the larger size of the holes and the velocity of the SI flow through them. The licensee performed a sensitivity study assuming a top-skewed axial power shape with the barrel-baffle flow path opened. This sensitivity study demonstrated that the core remained at saturated conditions following sump switchover and core blockage. The NRC staff found that, given the scenario with the barrel-baffle region open is the most realistic and supported by test data, it would be very unlikely that the core would experience temperatures above saturation. Additionally, this study did not model the horizontal flow holes connecting the barrel-baffle region and the core, which is an additional conservatism. Modeling these holes would further increase the SI flowing into the core and improve heat transfer, further reducing the likelihood of observing temperatures above saturation in the core.

<u>Heat stored in steam generators</u>

The licensee stated that the energy stored in the steam generators could impact the PCT in two main ways. First, the energy transferred to the primary side could cause boiling, which would increase the pressure drop and consequently increase the time it takes to fill a steam generator. Second, the energy transferred to the primary side could cause a reduction in the subcooling of the SI flow before it has the opportunity to cool the core. The licensee provided justification that the heat stored in the steam generators was not an important consideration in response to SNPB-3-2 (Reference 19). In its response, the licensee performed a sensitivity study and increased the secondary side metal mass by accounting for the mass of the u-tubes twice. This resulted in increasing in the total heat transferred from the steam generators to the primary side by a factor of 6. Even with this large increase in heat transferred, the resulting LTCC PCT remained within a small range of the saturation temperature. Given this sensitivity study, the NRC staff concludes that the heat stored in the steam generators has minimal impact on the PCT.

In summary, the licensee provided a list of the phenomena, a description of those phenomena, and the justifications of ranking those phenomena. The NRC staff reviewed this information and, as discussed above, determined that the dominant physical phenomena of "core uncovery" and "blockage on grid spacers in the fueled regions of the core" were ranked above "heat stored

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[5] See WCAP-17788, Volume 1, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Section 6.4.1; and Volume 5, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) - Autoclave Chemical Effects Testing for GSI-191 Long-Term Cooling," Section 5.6.

in steam generators." Thus, the phenomena have been identified, described, and ranked; therefore, this criterion is satisfied.

A.4.2.1.4 Initial and Boundary Conditions

Initial and Boundary Conditions

The description of each accident scenario should provide complete and accurate description of the plant initial and boundary conditions.

SRP Section 15.0.2, Subsection III.3c

The licensee provided information on this criterion in response to SNPB-3-6 (Reference 18). Specifically, the licensee provided the key parameters assumed in its simulations and confirmed these parameters with a direct comparison to plant data, or, if plant data were unavailable, with the results from a steady state run of the approved plant transient analysis code RETRAN. Though the licensee's verification assured the accuracy of most of the key initial and boundary conditions, the NRC staff asked questions about the following parameters:

- Treatment of core bypass flow
- Time from sump switchover to full-core blockage (360 seconds)
- Bounding break size (16-inch)
- Initial and boundary conditions for phase 4 - post-blockage LTCC

Core bypass flow

During the review, the licensee did not clearly indicate whether it credited the bypass flow in the barrel-baffle region or other core bypass flow paths. The licensee provided a clarification in response to SNPB-3-3 (Reference 16) by discussing the six flow paths which constitute the core bypass flow, but did not specify how each would be treated. In its supplemental response to SNPB-3-22 (Reference 18), the licensee addressed how each of the core bypass flow paths were modeled, and clarified that the flow in the barrel-baffle region is conservatively assumed to be blocked once the core is blocked at the time of sump switchover. While the barrel-baffle bypass flow is not credited in the official base-case, a sensitivity study was performed in response to SNPB-3-20 (Reference 20), which showed the impact of allowing flow in the barrel-baffle region. Based on the foregoing information, the NRC staff finds that the licensee's description of the core bypass flow paths is complete and accurate.

Time from sump switchover to full-core blockage (360 seconds)

The licensee provided justification that the time from sump switchover to full-core blockage was conservative in its response to SNPB-3-2 (Reference 19). In that response, the licensee clarified that the time between sump switchover and core blockage of 360 seconds corresponds to the time needed for 15 gm/FA to enter the RCS following sump switchover. In the analysis, this blockage is applied instantaneously at 360 seconds; it does not build up gradually, as would be expected in reality. The NRC staff finds this treatment of blockage timing conservative for two reasons.

First, the assumption that the core fully blocks once 15 gm/FA collects at the core inlet is a conservative assumption. As previously discussed, testing was used to demonstrate that

15 gm/FA applied at the core inlet would not significantly impede core cooling. The testing did not quantify how much fiber would be necessary to fully block the core, though testing demonstrated that, without chemical effects, a much higher value than 15 gm/FA could be supported.

Second, in reality, as the fiber layer started to build up on the lower spacers in the core, the pressure drop through the core would gradually increase before the core inlet became fully blocked. This would divert flow through other paths, including the steam generators. Thus, not all of the SI flow entering the RCS would actually make it to the core inlet, and once the RCS reached 15 gm/FA the amount of fiber at the core inlet would likely be less than that amount.

For the two reasons discussed above, the NRC staff finds the licensee's assumption of 360 seconds from sump switchover to full-core blockage as reasonable. Thus, the NRC staff finds the licensee's description of the sump switchover time is complete and accurate.

<u>Break size (16-inch)</u>

The licensee provided a justification that the 16-inch hot-leg break bounds all smaller breaks in response to SNPB-3-2 (Reference 19). In its response, the licensee noted that unlike medium and small cold-leg breaks where PCT is not correlated to break size, in medium and small hot-leg breaks, PCT is correlated to break size, with higher PCTs occurring at larger break sizes. This is because in smaller breaks, the pressure remains high for longer and ECCS flow is injected at a slower rate, allowing the RWST to drain slower and delaying the time before sump switchover. Therefore, smaller breaks will result in a lower decay power when sump switchover occurs and will also result in a slower buildup of debris in the RCS, since the SI injection would be ultimately limited to the break flow.

The licensee also performed a sensitivity study for three break sizes (16 inches, 6 inches, and 2 inches). This sensitivity demonstrated that the largest break size had the earliest sump switchover and therefore the highest decay power at core blockage. The licensee noted that the same four phases (break and blowdown, refill and reflood, pre-blockage and LTCC, and core blockage and post-blockage LTCC) are experienced in each case, with the break size mostly impacting the length of each phase. However, the licensee recognized that the figure of merit for determining the limiting break should not be PCT, but the core collapsed liquid level. This is because, in the analysis, the fuel remains covered and the PCT is therefore close to the saturation temperature. Thus, for an event such as the 2-inch break where the core never completely depressurizes, the saturation temperature remains high. The NRC staff reviewed the scenario and determined that the most limiting break would be the one resulting in core uncovery in the long-term phase with the highest PCT (or the break which brings the system closest to core uncovery during the long-term phase). Therefore, the limiting hot-leg break is the 16-inch break since it bounds all smaller breaks in the hot-leg piping. The NRC staff finds that the licensee's description of the bounding break size to be complete and accurate.

<u>Initial and boundary conditions for phase 4 - post-blockage long-term core cooling</u>

The licensee used the LTCC EM to simulate the entire event, however, the NRC staff considered the first three phases of the accident – break and blowdown, refill and reflood, pre-blockage and LTCC – only inasmuch as they provided the initial and boundary conditions for the fourth phase, post-blockage long-term core cooling. In other words, the NRC staff did not review the ability of the EM to adequately capture the PCT during blowdown or reflood, since the core is completely quenched at the start of the phase and the PCT would have no impact on the fourth phase. Instead, the NRC staff considered which aspects of the modeling of the first three phases would have the largest impact on the fourth phase.

The licensee stated that because the sump switchover occurred after reflood, the conditions at the beginning of phase 4 were relatively constant. In general, the core would be between the ECCS injection temperature and saturation temperature. The NRC staff found three areas in which the previous phases could have an impact on the fourth phase: the delay after full-core blockage when SI flow reaches the core, the decay power in the core at the time of blockage, and the amount of subcooling in the core.

- Concerning the delay after full-core blockage: The NRC staff noted that the licensee conservatively increased the delay after full-core blockage when SI flow reached the core by ignoring the gradual build-up of debris and ignoring most alternate flow paths. In reality, the debris would gradually build up on the bottom-most assembly spacer grid, which would result in a higher pressure drop through the core. This higher pressure drop would result in more flow being diverted to the steam generators. Because the steam generators become effectively the only path for SI flow to enter the core once the core inlet is blocked, the primary side of the steam generator u-tubes must be completely full to establish core cooling. In the current analysis, the steam generator u-tubes do not completely fill with water until blockage occurs, because the major flow path is through the core and out the break. This increases the delay between core blockage and when SI flow can reach the core. If the debris build-up were modeled as a gradual increase instead of a step change, additional flow would be diverted to the steam generators, causing them to fill faster and reducing the delay between core blockage and when SI flow reaches the core. Further, as detailed in response to SNPB-3-3 (Reference 16), the licensee ignored alternate flow paths which would also decrease this time delay and could prevent a time delay altogether. Therefore, the licensee made conservative assumptions concerning the delay after full-core blockage.

- Concerning the decay power in the core at the time of blockage: The NRC staff noted that the licensee ensured a conservatively early time for full-core blockage by reducing the time to sump switchover. This was accomplished by minimizing volume in the RWST and assuming all SI trains were fully operational and not operating at minimal efficiency. Since the licensee reduced the time to sump switchover and blockage, decay heat in the core would be higher requiring a higher cooling capacity. Therefore, the licensee made conservative assumptions concerning the decay power in the core at the time of blockage.

- Concerning the amount of subcooling in the core: The NRC staff noted that the licensee conservatively reduced the subcooling in the core at the start of the fourth phase. This was accomplished by maximizing the RWST temperature and ignoring any ECCS cooling. Since the licensee assumed lower cooling capacity of the RWST coolant and no ECCS cooling, the licensee conservative treated the amount of subcooling in the core.

The NRC staff concludes that the licensee (1) applied initial and boundary conditions that reflect plant operating conditions, (2) used justifiable initial and boundary conditions for key inputs such as break size and time to sump switchover, and (3) modeled the first three phases of the event in such a way as to result in conservative initial conditions for the fourth phase. The fourth phase conservative assumptions included increasing the time delay between full-core blockage and when SI flow reaches the core, ensuring an early blockage time with a resulting higher heat load in the core, and reducing the subcooling in the core. The NRC staff therefore determined that the licensee provided complete and accurate descriptions of the plant initial and boundary conditions, and that the initial and boundary conditions for the accident scenario reflect real or conservative plant conditions, thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.2 Documentation

The development of an EM for use in reactor safety licensing calculations requires a substantial amount of documentation including (a) the EM, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the Quality Assurance Program (QAP).

Section 15.0.2, Subsection III.3.a of the SRP contains seven review criteria for the NRC staff's documentation assessment. The review criteria topics and subsections with the NRC staff's review criteria assessments are listed in Table 6.

Table 6: Documentation Review Categories

Documentation	
A.4.2.2.1	Necessary Documentation
A.4.2.2.2	Theory Manual
A.4.2.2.3	Closure Relationships
A.4.2.2.4	User Manual
A.4.2.2.5	Options for Licensing Calculations
A.4.2.2.6	Required Input
A.4.2.2.7	Accident-Specific Guidelines

A.4.2.2.1 Necessary Documentation

Necessary Documentation

The documentation should be reviewed to determine if (i) all documentation listed in Section II.1 above has been provided [the evaluation model, the accident scenario identification process, the code assessment, the uncertainty analysis, a theory manual, a user manual, and the quality assurance program], (ii) the evaluation model overview provides an accurate roadmap of the evaluation model documentation, (iii) all documentation is accurate, complete, and consistent and, (iv) all symbols and nomenclature have been defined and consistently used.

SRP Section 15.0.2, Subsection III.3a

In Chapter 5 of Attachment 1-3 to Supplement 2 (Reference 10), STPNOC provided an initial overview of the LTCC EM. Additional documentation, including the licensee's changes to the LTCC EM, is captured in the following list:

- RELAP5-3D code manuals (Reference 9)

- Relevant STPNOC RAI responses pertaining to the LTCC EM:

  - RAI Response – Part 1 (Reference 16)
  - RAI Response – Part 2 (Reference 17)
  - RAI Response – Part 3 (Reference 18)
  - RAI Response – Supplement (Reference 19)
  - RAI Response – Supplement (Reference 20)

Based on the NRC staff's review of the above references, the licensee provided all documentation necessary for the NRC staff to complete its documentation review, the licensee's EM overview provides an accurate roadmap of the EM documentation, the licensee's EM is adequately described for the simulations performed and the documents were accurate, complete, and consistent, with all symbols and nomenclature defined and used consistently. Therefore, the NRC staff concludes that this criterion is satisfied.

A.4.2.2.2 Theory Manual

Theory Manual

The theory manual should be a self-contained document that describes the field equations, closure relationships, numerical solution techniques, and simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods.

SRP Section 15.0.2, Subsection III.3a

The licensee provided information on the theory manual in response to SNPB-3-8 (Reference 16). In the licensee's response, it provided a brief description of each key phenomena, how each phenomenon was modeled, and where more information on each phenomenon can be found in the RELAP5-3D theory manual (Reference 9).

The licensee provided the theory manual for RELAP5-3D, a listing of the important phenomena and the link between those phenomena and how they are modeled in the LTCC EM, and other docketed correspondence describing the inputs and other assumptions (References 16, 17, 18, 19, and 20).

The NRC staff determined that the theory manual for RELAP5-3D, and any particular input, model, and parameter selections resulting from its use in the LTCC EM as specified in the RAI responses, is a self-contained document containing the field equations, closure relationships, numerical solution techniques, and associated simplifications and approximations for these equations and techniques. The NRC staff concludes the licensee provided an appropriate theory manual, and, therefore, this criterion is satisfied.

A.4.2.2.3 Closure Relationships

Closure Relationships
The theory manual should identify the pedigree or origin of closure relationships used in the code and the limits of applicability for all models in the code.
SRP Section 15.0.2, Subsection III.3a

The licensee provided information on the closure relationships in response to SNPB-3-9 (Reference 17). In the licensee's response, it provided pointers to the relevant volume of the RELAP5-3D manual, which describes the various closure models for use in the code. This information, along with the phenomena mapping given in response to SNPB-3-8 and the detailed information provided in the RELAP5-3D manuals, allows the pedigree (i.e., history and origin) and limits (i.e., range of applicability) of each closure relationship to be determined.

The licensee provided the link between the important phenomena and their associated technical references in the RELAP5-3D manual, thus, the NRC staff finds that the closure relationships have been appropriately documented. Therefore, the NRC staff concludes that this criterion is satisfied.

A.4.2.2.4 User Manual

User Manual
The user manual should provide guidance for selecting or calculating all input parameters and code options.
SRP Section 15.0.2, Subsection III.3a

The licensee provided information on the user manual in response to SNPB-3-10 (Reference 17). It should be noted that the user manual was provided early in the review process and describes four scenarios for its use. It was only after the submittal of the RAI response that the licensee decided to address the small cold-leg break using a different approach. The user manual given to the NRC staff by the licensee provides information on how to execute the input decks, rather than providing instructions for building an input deck as would normally be expected in a user manual for an EM. Thus, in and of itself, the user manual is of limited use for performing new or additional simulations using the same LTCC EM. However, since the scope of review was restricted to consider the LTCC EM and the simulations which were generated by STPNOC for use only by STPNOC, future use of LTCC EM under different conditions were not considered in this review.

Since the user manual is mainly important for future use of the LTCC EM, the NRC staff determined the user manual to be irrelevant for this review. By limiting the scope of the review to only those simulations described in the licensee's RAI responses, and by restricting the LTCC EM use to STP only, the NRC staff concludes that this criterion does not apply.

A.4.2.2.5 Options for Licensing Calculations

> **Options for Licensing Calculations**
>
> *The guidance in the [user] manual should specify the required and acceptable code options for the specific licensing calculations.*
>
> SRP Section 15.0.2, Subsection III.3a

Because the LTCC EM is only used to perform the simulations described in the RAI responses provided by the licensee, the NRC staff determined that consideration of future licensing calculations was not needed. The NRC staff concludes that this criterion does not apply.

A.4.2.2.6 Required Input

> **Required Input**
>
> *The required input settings are hardwired into the input processor so that the code stops with an error message if the required input is not provided or if the input is not within an acceptable range of values or that administrative controls (an independent reviewer QA check) are in place that accomplish the same purpose.*
>
> SRP Section 15.0.2, Subsection III.3a

The LTCC EM makes use of the RELAP5-3D computer code. The NRC staff obtained a copy of RELAP5-3D code and confirmed that the code would stop with error messages and warnings if certain input was not provided or if the simulation provided erroneous results. While it is not feasible to confirm that every error in input will result in an error message, the wide use of RELAP5-3D, the wide use of the computer code which is it based on (RELAP5/MOD3), and its further development provides confidence that many such potential input errors have been discovered and corrected. Additionally, the limited use of the LTCC EM, restricted to the simulations already performed and slight variations from these simulations, would likely not result in any new errors. Therefore, the NRC staff concludes the computer code, RELAP5-3D, input setting are hardwired into the input processor and will provide the appropriate error messages when the required input is not provided; therefore, this criterion is satisfied.

A.4.2.2.7 Accident-Specific Guidelines

> **Accident-Specific Guidelines**
>
> *Computer codes that are used for multiple accidents and transients should include guidelines that are specific to each transient or accident.*
>
> SRP Section 15.0.2, Subsection III.3a

The licensee provided justification for the use of the LTCC EM for the simulations of the 16-inch hot-leg breaks (and various sensitivity studies), however, complete accident-specific guidelines were not provided. This is because the NRC staff approval is limited to only those simulations already submitted to the NRC. Thus, future use of the LTCC EM by this licensee or others requires prior review and approval by the NRC staff.

The NRC staff concludes that this criterion is met since accident specific guidelines were provided for the STP simulations performed and the licensee is not authorized to use the computer codes for accidents and transients outside the simulations reviewed by the NRC staff.

A.4.2.3 Evaluation Model Development

As discussed in Section A.2, "Regulatory Criteria," of this In-Vessel Thermal-Hydraulic Analysis, 10 CFR 50.46 defines an EM. Section 15.0.2, Subsection III.3.b of the SRP contains eight review criteria for EMs. The review criteria topics and the subsections that provide the NRC staff's assessments are listed in Table 7.

Table 7: Evaluation Model Review Categories

Subsection	
A.4.2.3.1	Previously Reviewed and Accepted Codes and Models
A.4.2.3.2	Physical Modeling
A.4.2.3.3	Field Equations
A.4.2.3.4	Validation of the Closure Relationships
A.4.2.3.5	Simplifying and Averaging Assumptions
A.4.2.3.6	Level of Detail in the Model
A.4.2.3.7	Equations and Derivations
A.4.2.3.8	Similarity and Scaling

A.4.2.3.1 Previously Reviewed and Accepted Codes and Models

Previously Reviewed and Accepted Codes and Models

It should be determined if the mathematical modeling and computer codes used to analyze the transient or accident should have been previously reviewed and accepted.

SRP Section 15.0.2, Subsection III.3b

The LTCC EM makes use of the RELAP5-3D computer code, which is the latest in the RELAP5 series of computer codes created by Idaho National Laboratory. While other computer codes based on RELAP5 have been submitted, reviewed, and accepted by the NRC staff (Reference 26), RELAP5-3D has not. Additionally, previous approvals of LOCA codes in EMs focused on the blowdown, refill, and reflood phases of a LOCA, rather than the LTCC phase.

The LTCC EM submitted by the licensee focuses on LTCC with debris. The NRC staff is not aware of any computer codes which have been specifically reviewed and accepted for LTCC analysis. Therefore, the review of the LTCC EM using RELAP5-3D is considered a new review and will not directly rely on the NRC's acceptance of previously reviewed computer codes. However, in certain instances, the licensee compared RELAP5-3D predictions to those of previously reviewed and accepted methods; each of these instances is addressed individually.

The NRC staff determined that the mathematical modeling and computer codes used to analyze the accident have not been previously reviewed and accepted. Therefore, the NRC staff reviewed those mathematical models and computers codes in this In-Vessel Thermal-Hydraulic Analysis. The NRC staff concludes that this criterion is satisfied.

A.4.2.3.2 Physical Modeling

Physical Modeling

The physical modeling described in the theory manual and contained in the mathematical models should be adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the initial and boundary conditions were accurate in response to SNPB 3-8 and SNPB-3-11 (Reference 16). In those responses, the licensee provided a brief description of RELAP5-3D's capability to model certain physical phenomena. Additionally, the licensee provided a description of the accident progression (discussed in Section A.4.2.1.2, "Accident Progression") as well as an identification of the highly ranked phenomena (discussed in Section A.4.2.1.3, "Phenomena Identification and Ranking"). Because the focus of the LTCC EM was on the behavior of the core following sump switchover and full-core blockage, the NRC considered the physical modeling of the first three phases inasmuch as they provide reasonable initial and boundary conditions for the fourth phase. As detailed in the section on accident progression, the phenomena associated with the first three phases of the event are within the general scope of the standard LOCA analysis for which RELAP5-3D was developed. Additionally, the phenomena associated with phase 4 of the event are a subset of those phenomena which are considered important in the first three phases.

Because the scenarios under consideration are expected to experience the same phenomena as those which occur during a LOCA, and because RELAP5-3D was developed with models for all key LOCA phenomena (including those identified by the licensee and reviewed by the NRC staff as important in phase 4), the NRC staff determined that the physical phenomena modeled and described in the theory manual are adequate to calculate the accident scenario considered. The NRC staff concludes that this criterion is satisfied.

A.4.2.3.3 Field Equations

Field Equations

The field equations of the evaluation model should be adequate to describe the set of physical phenomena that occur in the accident.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the field equations were accurate in response to SNPB-3-12 (Reference 16). In that response, the licensee provided a description of the field equations being used. The licensee's EM employs a two-fluid model for two-phase flow with seven total field equations (mass, momentum, energy, and mixture energy), in one-dimensional form. Thus, while the computer code is called RELAP5-3D, only 1D components were used in the LTCC EM.

Because the licensee used an industry-standard two-fluid model which can account for non-equilibrium effects between the vapor and liquid phases and the standard 1D implementation of the field equations, the NRC staff determined that the field equations adequately model the physical phenomena of interest. The NRC staff concludes that this criterion is satisfied.

A.4.2.3.4 Validation of the Closure Relationships

Validation of the Closure Relationships

The range of validity of the closure relationships should be specified and should be adequate to cover the range of conditions encountered in the accident scenario.

SRP Section 15.0.2, Subsection III.3b

The licensee provided information on the validation of the closure relationships in response to SNPB-3-13 (References 16 and 19). In the initial response, the licensee provided a map from the phenomena modeled during a LOCA to the validation of those phenomena in the RELAP5-3D manual. In the supplemental response, the licensee discussed the validation of the key closure relationships for the LTCC EM and separated the closure relationships into five main areas: (1) flow regime maps, (2) energy closure relations, (3) momentum closure relations, (4) flow process models, and (5) other models (e.g., models for special components, reactor kinetics).

As discussed in Section A.3.1, "NRC Staff Evaluation Method," of this In-Vessel Thermal-Hydraulic Analysis, the NRC staff did not consider every closure model used in the LTCC EM, but instead focused on two sets of relationships: (a) those closure relationships of key physical phenomena for phase 4 (identified in Section A.4.2.1.3, "Phenomena Identification and Ranking"), and (b) all other closure relationships used in the LTCC EM. These are discussed below.

<u>Key Closure Relationships</u>

There were two highly ranked phenomena identified in phase 4 of the LTCC EM: natural convection heat transfer and counter current flow limitation (CCFL). For the first highly ranked phenomenon, the licensee stated that the heat transfer in the natural convection flow regime was modeled using the Chen correlation for both saturated and subcooled nucleate boiling. Because this correlation was used previously and is commonly used to predict such heat transfer, the NRC staff determined that this closure model generates appropriate predictions of the underlying phenomena.

For the second highly ranked phenomenon, the licensee stated that the counter current flow limitation was modeled using the Wallis CCFL correlation with smooth edges. The CCFL model is applied at the top of the core. The licensee performed a sensitivity study comparing three different CCFL models which could be used at this location. The data used to assess the various CCFL models was obtained from multiple tests (Reference 31). In general, a test was defined as two spaces separated by a horizontal plate which contained holes. The tests were characterized by the plate thicknesses, the various diameters of the holes, the number of holes, and pitch between the holes. When comparing the superficial velocities of gas and liquid, the number of holes seemed to be the largest predictor of the data's behavior, as more holes would support higher superficial velocities and hence better mixing.

Each of the three correlations studied—Wallis with sharp edges, Wallis with smooth edges, and Bankoff—was well correlated to a different data set (i.e., group of data with similar corresponding number of holes). The Wallis correlation with sharp edges conservatively predicted all the data (i.e., calculated a minimum of superficial velocities and therefore reduced mixing). While this correlation seemed appropriate in situations in which there were few holes, it greatly under-predicted the superficial velocities in situations where there were many holes. Wallis with smooth edges over-predicted the superficial velocities in data with a few number of holes, but under-predicted the superficial velocities in data with a moderate number of holes and greatly under-predicted the superficial velocities of data with a large number of holes. This was the correlation chosen by the licensee for the LTCC EM.

While Bankoff provided the best prediction of the data with a large number of holes, it significantly over-predicted the superficial velocities from the data with a moderate and a small number of holes. Thus, though Bankoff was likely the most realistic choice for the number of holes that exist in the upper core plate, the licensee chose to use the more conservative Wallis with smooth edges CCFL correlation. To demonstrate the impacts of the CCFL model, the licensee also performed a sensitivity study where each of the three models was used for a simulation. The Wallis/sharp model had the least mixing between the core and the upper plenum, and this resulted in core uncovery which occurred at various times after full-core blockage. The Wallis/smooth model did not show core uncovery, but resulted in fluctuations of the PCT, likely due to fluctuations of pressure in the core caused by void formation. The Bankoff model resulted in no such fluctuations. This was likely due to the high mixing predicted by this model as void formation in the core would not prevent liquid from reaching the core from the upper plenum, and therefore the PCT remained relatively constant and consistent. Because the Wallis with smooth edges was demonstrated to result in an accurate prediction of the relevant test data, and would likely result in an over-prediction of conditions causing CCFL which would reduce the mixing to the core, the NRC staff determined that this closure model will generate appropriate predictions of the underlying phenomena.

The NRC staff also notes that the meshing of the core alone is likely a significant conservatism. In the analysis, there is little time in the long-term phase when there is not water in the upper plenum, and therefore the only impedance to cooling the core is the CCFL which could occur on the upper core plate. It is likely that if CCFL could be completely excluded, then the entire simulation could be simplified to ensure that any delay in the SI flow reaching the core following full-core blockage would not result in core uncovery. However, the potential for CCFL does add a significant complication in that there can be adequate cooling flow in the upper plenum, but that flow may not be able to penetrate through the upper core plate and into the core itself. However, the NRC staff notes that the core noding used is likely to exaggerate the impacts of CCFL. In the simulations, the core itself is modeled using multiple axial nodes but a single radial node and the upper core plate is a single node on top of the core. Thus, any disturbance in the flow between the core and the upper plenum is experienced over the entire node. In reality, while CCFL may occur in hotter channels which generate significant amounts of steam (e.g., those commonly found in central regions of the core), it would not be likely to occur in lower power channels (e.g., those commonly found in the core periphery). Therefore, a likely SI flow path is down through the core support plate on the periphery, down the periphery fuel bundles, and then into the center of the core to make up for any loss due to boil off. Given the open lattice nature of the core, this flow path would in reality be almost unaffected by CCFL. However, due to the manner in which the core is meshed and the CCFL model is applied, this flow path is not possible in the simulation. The NRC staff finds that this is likely to be a conservatism in the licensee's analysis.

<u>Other closure relationships</u>

The closure relationships for the LTCC EM are found in Chapter 4 of the RELAP5-3D manual. The NRC staff reviewed the appropriate closure relationships and found that many of the closure models are commonly used to model phenomena under similar conditions as those found in the LTCC EM. The NRC staff determined that the other closure models will generate appropriate predictions of the underlying phenomena because these closure models are mature, the RELAP5 series of codes has wide use, and the phenomena observed during most of the simulations is well known and consistent with expectations (i.e., appropriate behavior during blowdown, refill, reflood).

Based on its review, the NRC staff determined that the range of validity of the closure relationships have been specified and are adequate to predict their underlying phenomena, because the key phenomena are relatively simple in nature, and because of likely conservatisms in the treatment of both core radial noding and CCFL modeling. Thus, the NRC staff determined that the closure relationships are adequate to cover the range of condition encountered in the accident scenario; therefore, this criterion is satisfied.

A.4.2.3.5 Simplifying and Averaging Assumptions

Simplifying and Averaging Assumptions

The simplifying assumptions and assumptions used in the averaging procedure should be valid for the accident scenario under consideration.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the simplifying and averaging assumptions were accurate in response to SNPB-3-14 (Reference 16). In the response, the licensee referenced the theory manual for RELAP5-3D (Reference 9) and provided a brief summary of the field equations used in the RELAP5-3D analysis. They also provided a brief description of how those equations were solved and a reference describing how they were derived.

The NRC staff reviewed the licensee's responses and determined that the simplifying assumptions used in the averaging procedure are valid for the accident scenario because the LTCC EM uses a well-documented and commonly used approach in averaging, and the scenarios under consideration do not contain phenomena that would challenge the averaging approach used in RELAP5-3D. Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.3.6 Level of Detail in the Model

Level of Detail in the Model
The level of detail in the model should be equivalent to or greater than the level of detail required to specify the answer to the problem of interest.
SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the level of detail in the models were accurate to simulate the problem of interest in response to SNPB-3-15 (Reference 18). In that response, the licensee provided a summary of the level of detail included in the LTCC EM and confirmed that the current modeling is consistent with STP's licensing basis analysis.

The NRC staff reviewed the licensee's responses and determined that STPNOC used an appropriate level of detail in the LTCC EM because the overall approach STPNOC described is consistent with that of a typical LOCA analysis; the phenomena of interest which most influence the figure of merit are known, well studied, and commonly modeled phenomena; and STPNOC performed appropriate sensitivity studies confirming the behavior of the LTCC EM (see Section A.4.2.4.7, "Sensitivity Studies"). Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.3.7 Equations and Derivations

Equations and Derivations
The equations and derivations should be correct.
SRP Section 15.0.2, Subsection III.3b

The manual for the RELAP5-3D code does not provide the specific derivation of equations, but instead relies on references for the development of such equations, since those derivations have been previously performed. The NRC staff determined that the equations and derivations are correct because these equations were previously derived and recorded in numerous references, and the key phenomena are well understood. Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.3.8 Similarity and Scaling

Similarity and Scaling
The similarity criteria and scaling rationales should be based on the important phenomena and processes identified by the accident scenario identification process and appropriate scaling analyses. Scaling analyses should be conducted to ensure that the data and the models will be applicable to the full scale analysis of the plant transient.
SRP Section 15.0.2, Subsection III.3b

In the analysis provided by the licensee supporting the modeling choices employed in the LTCC EM, one set of assessment data was used. The NRC staff reviewed the assessment data and determined that use of this data would be applicable to the full scale analysis of the plant transient because the assessment data was obtained over a wide range of conditions, and the data which was most applicable to reactor scale was conservatively predicted (as detailed in the discussion of the CCFL correlation in Section A.4.2.3.4, "Validation of the Closure Relationships"). Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.4 Code Assessment

The code assessment considers all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario. The focus in Section A.4.2.3, "Evaluation Model," above, was on the field equations, the closure relationships, and the phenomena they model. The field equations and closure relationships were considered separately. In this section, the focus shifts to the combined use of all field equations and closure relationships in generating the figures of merit.

Section 15.0.2, Subsection III.3.d of the SRP contains eight review criteria for the code assessment. The review criteria topics and the sections providing the NRC staff's review are listed in Table 8.

Table 8: Code Assessment Review Categories

Subsection	
A.4.2.4.1	Single Version of the Evaluation Model
A.4.2.4.2	Validation of the Evaluation Model
A.4.2.4.3	Range of Assessment
A.4.2.4.4	Numerical Solution
A.4.2.4.5	Code Tuning
A.4.2.4.6	Compensating Errors
A.4.2.4.7	Sensitivity Studies
A.4.2.4.8	Assessment Data

A.4.2.4.1 Single Version of the Evaluation Model

Single Version of the Evaluation Model

All assessment cases should be performed with a single version of the evaluation model.

SRP Section 15.0.2, Subsection III.3d

The licensee stated that a single version of the EM was used for the submitted analysis in response to SNPB-3-16 (Reference 16). The NRC staff concludes that this criterion is satisfied because the licensee confirmed that a single version of the LTCC EM was used for the LTCC analysis.

A.4.2.4.2 Validation of the Evaluation Model

> **Validation of the Evaluation Model**
>
> *Integral test assessments must properly validate the predictions of the evaluation model for the full size plant accident scenarios. This validation should cover all of the important code models and the full range of conditions encountered in the accident scenarios.*
>
> SRP Section 15.0.2, Subsection III.3d

The licensee provided information on the validation of the EM in response to SNPB-3-17 (References 18 and 19). In that response, STPNOC provided a summary of the verification of the input models, and a summary of their judgment of the simulation results. The licensee also provided a discussion of the results of the simulations performed with the LTCC EM and how those results are consistent with the expected behavior following a LOCA. However, in general, there are no readily available test data which can be used to provide validation for the LTCC EM. Due to the lack of test data, the NRC staff used engineering judgment to conclude that the EM would produce an adequate prediction of the underlying phenomena. The staff based this conclusion on the following four considerations.

First, the NRC staff found that the analysis performed by the licensee contains a large number of conservatisms, including the following licensee assumptions:

- imposing a full-core blockage once the RCS system contains debris at 15 gm/FA, when the core would not be expected to fully block until some higher value of debris (as observed in testing supporting WCAP-16793) in the SE

- ignoring flow through the barrel-baffle region, even though test data demonstrates it remains unblocked

- ignoring flow through the holes between the barrel-baffle region and the core, even though these holes would likely remain unblocked

- biasing key input parameters conservatively including RWST volume, RWST temperature, and ECCS cooling

- using a conservative CCFL model and core modeling which is likely to exaggerate the impact of CCFL and underestimate the true mixing in the core

Second, the NRC staff found that even with these conservative assumptions, the resulting simulation of the long-term cooling phase was relatively simple. The majority of the SI flow was forced to flow through the upper plenum before it could flow out the break. There were few complex phenomena, and the only real complexity was caused by the potential for CCFL, which reduced the flow of the water from the upper plenum through the upper core plate and into the core.

Third, the NRC staff observed test data which demonstrated that the barrel-baffle region would remain open and would not be blocked by debris. A sensitivity study performed by the licensee demonstrated that with the barrel-baffle region unblocked, the core would not experience uncovery. A sensitivity study demonstrated that relaxing this conservatism, while keeping the others, reduced the complexity of the simulation to a simple mass balance equation because an alternative flow path was available for the SI flow to reach the core without being impacted by CCFL. It should be noted that this sensitivity only allowed flow through the barrel-baffle region. It did not allow flow through the core baffle holes, which would have further increased the coolant flow to the core.

Fourth, a sensitivity study by the licensee suggested that even if the barrel-baffle region could remained blocked, the core would not necessarily experience uncovery, provided additional radial channels were added to the core noding in the EM. While CCFL was found to restrict the flow from the upper plenum into the core in the analysis, it is possible that this results from an oversensitivity to CCFL caused by modeling the core – specifically, the top of the core where CCFL occurs – as a single radial node. Given the large flow area at the top of the core, the NRC staff finds it reasonable that CCFL would occur above the relatively hotter channels but likely not above the colder channels (e.g., the periphery). Above the colder channels, water from the upper plenum would flow down through the upper core plate and into the fuel bundles, where it would flow into the hotter portions of the core at elevations where CCFL would not be a consideration.

Based on the four conservatisms discussed above, the NRC staff concludes that the LTCC EM produced an accurate or conservative simulation and, therefore, this criterion is satisfied.

A.4.2.4.3 Range of Assessment

Range of Assessment

All code closure relationships based in part on experimental data or more detailed calculations should be assessed over the full range of conditions encountered in the accident scenario by means of comparison to separate effects test data.

SRP Section 15.0.2, Subsection III.3d

The issue of range of conditions was considered in the NRC staff's assessment of the validation of the closure relationships (see Section A.4.2.3.4, "Validation of the Closure Relationships") and the integral tests (see Section A.4.2.4.2, "Validation of the Evaluation Model"). Based on the conclusions in Sections A.4.2.3.4 and A.4.2.4.2, the NRC staff concludes that code closure relationships were assessed over the range of conditions encountered in the accident scenario, compared to separate test data, and found to be acceptable; therefore, this criterion is satisfied.

A.4.2.4.4 Numerical Solution

Numerical Solution

The numerical solution should conserve all important quantities.

SRP Section 15.0.2, Subsection III.3d

The NRC staff confirmed that the important quantities in mass, momentum, and energy are directly modeled. Further, the NRC staff confirmed that the QAP under which the simulations were performed, directed analysts to ensure proper convergence for each run (see the response to SNPB-3-25 in Reference 18).

The NRC staff determined that the numerical solution will conserve all important quantities such that the figures of merit can be adequately predicted because of the field equations (i.e., conservation equations) used by the licensee and the QAP direction to ensure proper convergence. Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.4.5 Code Tuning

Code Tuning

All code options that are to be used in the accident simulation should be appropriate and should not be used merely for code tuning.

SRP Section 15.0.2, Subsection III.3d

During the NRC staff's review of the LTCC EM, the NRC staff found the code options chosen to be appropriate and did not find any evidence of code tuning. Additionally, the sensitivity studies evaluated in Section A.4.2.4.7, "Sensitivity Studies," provide further evidence of the appropriateness of the code option choices.

Because the code options were appropriate for the scenarios simulated and the sensitivity studies demonstrated robustness in key aspects of performing the simulations, the NRC staff determined that the LTCC EM was not artificially tuned. The NRC staff concludes that this criterion is satisfied.

A.4.2.4.6 Compensating Errors

Compensating Errors

The reviewers should ensure that the documentation contains comparisons of all important experimental measurements with the code predictions in order to expose possible cases of compensating errors.

SRP Section 15.0.2, Subsection III.3d

There were no direct comparisons of code predictions to experimental measurements. This issue is addressed fully in Section A.4.2.4.2, "Validation of the Evaluation Model." However, the NRC staff considered the potential for compensating error in the NRC staff's assessments of the initial and boundary conditions (see Section A.4.2.1.4, "Initial and Boundary Conditions"), the assessment of the validation of the closure relationships (see Section A.4.2.3.4, "Validation of the Closure Relationships"). The NRC staff also considered compensating errors due to experimental measurements and those that may have resulted from making the analysis "more conservative." Based on the NRC staff's review of the licensee's submittal and discussions in Sections A.4.2.1.4, A.4.2.3.4, and A.4.2.4.2, in this In-Vessel Thermal-Hydraulic Analysis, the NRC staff concludes that this criterion is satisfied.

A.4.2.4.7 Sensitivity Studies

Sensitivity Studies

Assessments should be performed where applicable [specific test cases for LOCA to meet the requirements of Appendix K to 10 CFR Part 50 and TMI [Three Mile Island] action items for PWR small-break LOCA}.

SRP Section 15.0.2, Subsection III.3d

Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

Appendix K to 10 CFR Part 50

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

TMI [Three Mile Island] action items for PWR (Reference 28)

The NRC staff determined that the TMI action items are out of the scope of this review because the items were related to pressurized thermal shock, not PCT.

The licensee provided information on the following sensitivity studies in response to SNPB-3-18 (References 18 and 19) and SNPB-3-20 (References 18, 19, and 20). In those responses, the licensee provided details on the following sensitivity study topics:

- Core radial mesh sensitivity
- Core axial mesh sensitivity
- Appendix K decay heat with single worst failure and steam generator tube plugging
- Axial power shape
- Break size sensitivity
- Break orientation
- Open barrel-baffle region

Core radial mesh sensitivity

The licensee provided information on the core radial mesh sensitivity in response to SNPB-3-18 (References 18 and 19). In the sensitivity study, the licensee added a hot channel to the core region. This two-channel model was compared to the single core channel model (i.e., the base case). As expected, the conditions for CCFL occurred more frequently at the top of the hot channel than they did at the top of the average channel. This difference is greatest immediately after core blockage, when CCFL in the hot channel is nearly constant, while the average channel is not constant. Additionally, when the PCT is compared to the base case, both cases fluctuate in a similar manner near the saturation temperature, but the two-channel model is much smoother and at a slightly lower temperature than the base case. The NRC staff considers this likely because in the two-channel model, liquid can flow into the average channel and steam and liquid exit through the hot channel, but in the base case all liquid flowing into the core and all steam and liquid exiting the core must go through the same channel (i.e., the same node). The NRC staff finds that the licensee's treatment of the radial mesh of the core is acceptable because the single-channel model predicts conservatively compared to the simulated two-channel model, and because a model with a realistic number of channels would likely predict even better mixing between the core and upper plenum.

Core axial mesh sensitivity

The licensee provided information on the core axial mesh sensitivity in response to SNPB-3-18 (Reference 19). In this sensitivity study, the licensee reduced the axial mesh in the core from 21 axial nodes to 10 axial nodes. The timings of major events during the simulations and the resulting PCT from both cases were nearly identical. Because the sensitivity demonstrates that a change in axial mesh size does not impact the simulations results and because the licensee is using an axial mesh size similar to that used for LOCA analysis, the NRC staff finds that this treatment of the axial mesh of the core is acceptable.

Appendix K decay heat with single worst failure and steam generator tube plugging

The licensee provided a sensitivity study discussing the impacts of assuming Appendix K decay heat, single worst failure, and steam generator tube plugging in response to SNPB-3-20 (References 18, 19, and 20). This study included Appendix K decay heat (i.e., a 1.2 multiplier on the 1971 American Nuclear Society decay heat standard), the failure of a single train of ECCS, and steam generator tube plugging of 10 percent. In this study, the licensee did not assume some of the conservatisms found in the LTCC EM; namely, the licensee assumed a nominal RWST volume instead of the lower volume used in the base case, and nominal RWST and sump pool temperatures instead of the conservatively high temperatures used in the base case. A comparison to the base case showed that the two cases have very similar timings up through the reflood phase. However, sump switchover was delayed in the Appendix K sensitivity due both to the larger RWST volume and the reduced ECCS flow resulting from the assumed failure of a single train of ECCS. In both the base case and the Appendix K sensitivity study, the PCT and the final pressures are similar, and the temperature of the cladding is approximately the coolant's saturation temperature.

The NRC staff requested the Appendix K decay heat sensitivity study to determine the impacts of assuming Appendix K decay heat and single worst failure on the event. The staff reviewed the licensee's results and determined that a failed train of ECCS would delay the drainage of the RWST, resulting in delayed sump switchover/core blockage, and therefore a lower core power when blockage occurs. However, it is unclear from the licensee's response (1) how sensitive the timing of blockage is to the assumed decay heat level, (2) how much of an impact the failed ECCS train has on the blockage timing, and (3) how much of an impact the RWST volume has on the timing. While these items were not addressed in the licensee's sensitivity studies, the NRC staff finds that any increase in decay heat would have a minimal impact on the PCT, since the core would simply stabilize in the long-term phase at a slightly higher pressure and, therefore, slightly higher saturation temperature. As discussed above, the NRC staff finds the treatment of the 10 CFR 50 Appendix K decay heat sensitivity study is acceptable.

Axial power shape

The licensee provided information on the axial power shape in response to SNPB-3-20 (References 18, 19, and 20). This sensitivity study is evaluated in the discussion on core uncovery in Section A.4.2.1.3, "Phenomena Identification and Ranking."

Break size sensitivity

The licensee provided information on the break size sensitivity in response to SNPB-3-2 (Reference 19). This sensitivity study is evaluated in the discussion on bounding break size assumptions in Section A.4.2.1.4, "Initial and Boundary Conditions."

Break Orientation

The licensee provided information on the break size sensitivity in response to SNPB-3-20 (Reference 20). In its response, the licensee stated that the results of the study indicated that the break orientation had no significant impact on the resulting PCT. Because the break orientation had no impact on the results of the PCT, the NRC staff finds that the treatment of the break orientation is appropriate.

Open barrel-baffle bypass region

The licensee provided information on the barrel-baffle bypass sensitivity study in response to SNPB-3-20 (References 18, 19, and 20). The licensee also re-performed this sensitivity study assuming the worst power shape (top-skewed). As discussed in Section A.4.2.1.3, "Phenomena Identification and Ranking," the NRC staff determined that this sensitivity is likely to be the most realistic of all of the analysis performed, and demonstrates that there is no core uncovery during LTCC.

The NRC staff finds that appropriate test cases were performed because the comparisons demonstrated that even under-conservative conditions, the core temperature remains below the 800 °F acceptance criterion, thus, under more realistic but still conservative conditions, the core will likely not experience uncovery and will avoid exceeding the saturation temperature. Therefore, the NRC concludes that this criterion is satisfied.

A.4.2.4.8 Assessment Data

Assessment Data

Published literature should be referred to for sources of assessment data for specific phenomena, accident scenarios, and plant types.

SRP Section 15.0.2, Subsection III.3d

One set of assessment data were used in the licensee's analysis supporting the LTCC EM. Because the assessment data was obtained from published literature (see Reference 31) containing references to multiple other well-known publications, the NRC staff found this use of assessment data appropriate. The NRC staff concludes that this criterion is satisfied.

A.4.2.5 Uncertainty Analysis

Uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. Safety parameters are those parameters which have limits to ensure plant safety, such as the specified acceptable fuel design limits required by General Design Criterion 10, "Reactor design," in Appendix A to 10 CFR Part 50. Examples of safety parameters are PCT, cladding oxidation thickness, departure from nucleate boiling ratio, and critical power ratio.

No explicit uncertainty analysis was prescribed or performed for the LTCC EM. However, the NRC staff reviewed specific aspects of the LTCC EM to confirm that specific uncertainties would be accounted for in the analysis.

Section 15.0.2, Subsection III.3.e of the SRP contains three criteria for the uncertainty analysis. The review criteria topics and the subsections providing the NRC staff's review are listed in Table 9.

Table 9: Uncertainty Analysis Review Categories

Subsection	
A.4.2.5.1	Important Sources of Uncertainty
A.4.2.5.2	Experimental Uncertainty
A.4.2.5.3	Calculated and Predicted Results

A.4.2.5.1 Important Sources of Uncertainty

Important Sources of Uncertainty

The accident scenario identification process should be used in identifying the important sources of uncertainty. Sources of calculation uncertainties should be addressed, including uncertainties in plant model input parameters for plant operating conditions (e.g., accident initial conditions, set points, and boundary conditions). To address these uncertainties, demonstrate that the combined code and application uncertainty should be less than the design margin for the safety parameter of interest in the calculation.

SRP Section 15.0.2, Subsection III.3e

The licensee provided justification that important sources of uncertainty were identified in responses to RAI questions. First, the licensee identified and provided a description of the important uncertainties in response to SNPB-3-32 (Reference 18). Second, the licensee provided a discussion of the impact of the important uncertainties on the analysis in response to SNPB-3-22 (Reference 18). Third, the licensee described how the uncertainties were accounted for in the input in response to SNPB-3-21 (Reference 18). Finally, the licensee provided a discussion of how the important uncertainties were addressed in the input deck for the LTCC EM in response to SNPB-3-23 (Reference 18).

The following is a listing of the uncertainties identified by STPNOC, a summary of how those uncertainties were addressed, and the NRC staff's review each uncertainty.

<u>Initial Reactor Power</u>

The licensee used nominal reactor power (i.e., 3,853 megawatt-thermal) for the LTCC EM. The NRC staff determined previously in this In-Vessel Thermal-Hydraulic Analysis that the reactor power will have a minimal impact during the long-term phase and finds this treatment is acceptable. It should be noted that the licensee performed a sensitivity study on the decay heat used in the analysis, which is addressed in Section A.4.2.4.7, "Sensitivity Studies." This study demonstrated that a large change in decay heat (which is equivalent, for the figure of merit, to a change in the initial reactor power) may result in a small change in PCT. However, this change in PCT is due to the change in the saturation temperature. Because the PCT remains linked to the saturation temperature and well below the limit of 800 °F, the NRC staff finds that the licensee's treatment of the uncertainty is acceptable and, therefore, the initial reactor power uncertainty is acceptable.

Core Heat Structure Thermal Properties

No conservatisms were added to better address the core heat structure thermal properties, as the licensee stated these properties are of low significance during the long-term cooling phase. The NRC staff concluded that these properties would have a minimal impact on the PCT during the long-term phase and, at most could very slightly increase the heat transferred to the fluid during the long-term phase, since there is ample water to remove the heat during that phase, and any increase would be small and inconsequential. Thus, the NRC staff finds that no additional conservatism is needed and the licensee's treatment of the core heat structure thermal properties is acceptable; therefore, the core heat structure thermal properties uncertainty is acceptable.

Reactor Vessel Passive Heat Structures

The licensee did not add conservatism to better address the reactor vessel heat structures. However, the licensee included the metal mass from these structures in the LTCC EM and calculated the impact of the stored energy on the fluid in the RCS. The NRC staff notes that during the long-term cooling phase, most of the passive heat structures have been in contact with coolant for a substantial portion of the accident and have had ample opportunity to lose stored energy. Typically, the staff would expect to see the heat transferred from the steam generators to the primary coolant dominate over any remaining heat transferred from other portions of the RCS. The licensee performed a sensitivity study (discussed in Section A.4.2.1.3, "Phenomena Identification and Ranking") which demonstrated that the heat transferred from the steam generators had a minimal impact on the conditions in the core. The NRC staff finds that the licensee accounted for the passive heat structures in its LTCC EM and performed a sensitivity study which demonstrates that the remaining passive heat has little impact on conditions in the core. Therefore, the licensee's treatment of the uncertainty associated with the reactor vessel passive heat structures is acceptable and, therefore, the uncertainty is acceptable.

Reactor Core Axial Power Shape

The licensee performed a sensitivity study on the axial power shape. That sensitivity study and the use of the cosine axial power shape is evaluated in Section A.4.2.4.7, "Sensitivity Studies." The NRC staff found that the while the axial power shape had an impact on the given base case, that impact was due largely to other conservative assumptions, such as a blocked barrel-baffle region or the choice of modeling the core using a single radial node. When those assumptions were relaxed to result in a more realistic analysis, the sensitivity to the axial power shape was minimized. The NRC staff finds that the licensee appropriately accounted for the uncertainty associated with the reactor core axial power shape in its LTCC EM and, therefore, the reactor core axial power shape is acceptable.

Steam Generator Tube Plugging

The licensee performed a sensitivity study on steam generator tube plugging. However, in that sensitivity study, a number of additional parameters were varied simultaneously which were expected to have a much larger impact on the results than the tube plugging. In general, the NRC staff does not consider tube plugging to be an important uncertainty to capture, as its impact would likely be small. Further, it was not clear to the NRC staff if more plugging or less plugging would be conservative. More plugging increases the pressure drop through a steam generator, but removing tubes from service also means that less water is needed to fill the steam generator. The licensee chose to assume 0 percent tube plugging in the main analysis. Given that this would require the maximum amount of water to fill up the steam generators, which would cause the largest delay between full-core blockages and when SI flow reaches the core, the NRC staff finds the licensee's treatment of the uncertainty associated with steam generator tube plugging conservative and, therefore, the uncertainty is acceptable.

Vessel Flow Bypass Fractions

The licensee identified six bypass flows in response to SNPB-3-3 (Reference 16). Those six bypass flows are as follows:

1. Thimble tube flow through the fuel rods
2. Core former-to-fuel gap flows
3. LOCA holes flow between the barrel-baffle region and the core
4. Barrel-baffle flow between the bottom of the core and the top of the core
5. Cold-leg to hot-leg leakage flow
6. Upper head spray nozzle flow

The licensee conservatively chose to keep just one of the six bypass flows unblocked after full-core blockage: the upper head spray nozzle flow. The nodalization of this portion of the reactor vessel in the EM is discussed below. The licensee's analysis does not predict that it is an important flow path because the primary path for SI following core blockage is through the steam generators. The NRC staff finds the licensee's treatment of the uncertainty in the vessel flow bypass fractions is acceptable because of its conservative treatment (i.e., disregarding and minimizing) of other bypass flow paths which could provide substantial cooling to the core. Therefore, the vessel flow bypass fractions uncertainty is acceptable.

Core Nodalization

The licensee provided a sensitivity study of the core nodalization in response to SNPB-3-18 (References 18 and 19). That sensitivity study is evaluated in Section A.4.2.4.7, "Sensitivity Studies." Because the core nodalization used by STPNOC is very similar to the core nodalization commonly used for LOCA analysis by the industry, and because the sensitivity study verified the relative insensitivity of the PCT during the long-term phase to the core nodalization, the NRC staff finds that the licensee's treatment of the uncertainty in the core nodalization is acceptable. Therefore, the core nodalization uncertainty is acceptable.

Upper Head Nodalization

The licensee provided a description of the upper head nodalization in response to SNPB-3-22 (Reference 18). Early in the review process, the dominant SI flow path was into the cold-leg, into the top of the downcomer, through the upper head spray nozzles, into the upper head, down through the control rod guide tubes and into the top of the core. The licensee modified the nodalization of the upper head to better account for the standpipe effect of the control rod guide tubes. However, in the final scenarios considered by the licensee, this cooling path became inconsequential compared to the SI flow from the steam generator u-tubes. In light of the minimal impact of this flow path, the NRC staff finds that the new nodalization of the upper head adequately captures the standpipe effect from the guide tubes, and is therefore acceptable.

RWST Usable Volume

The licensee used an RWST volume which was conservatively biased lower than the expected usable volume, and even below the usable volume credited in a LOCA. Because this smaller volume is conservative for the analyses performed, the NRC staff finds this treatment of the RWST usable volume uncertainty is acceptable. Therefore, the RWST usable volume uncertainty is acceptable.

Decay Power Model

The licensee performed a sensitivity study on the core decay power. That sensitivity study and the use of the 10 CFR 50 Appendix K decay power is evaluated in Section A.4.2.4.7, "Sensitivity Studies." The NRC staff determined that any increase in decay heat would have a minimal impact on the PCT, since the core would be at a slightly higher pressure and, therefore, slightly higher saturation temperature. Thus, the NRC staff finds the licensee's treatment of the uncertainty associated with the decay power model is acceptable and, therefore, the decay power model uncertainty is acceptable.

Break Size

The licensee performed a sensitivity study on the various break sizes. That sensitivity study, and the use of the 16-inch break size, is evaluated in Section A.4.2.1.4, "Initial and Boundary Conditions." The NRC staff determined the 16-inch break most limiting compared to smaller breaks. The NRC staff finds that the licensee's treatment of the uncertainty associated with breaks size to be acceptable and, therefore, the break size uncertainty is acceptable.

Break Orientation

The licensee performed a sensitivity study on the break orientation. That sensitivity study and the use of the orientation chosen is evaluated in Section A.4.2.4.7, "Sensitivity Studies." The NRC staff determined that the break orientation has no impact on the PCT during LTCC. Thus, the licensee's treatment of the uncertainty in break orientation is acceptable and, therefore, the break orientation uncertainty is acceptable.

ECCS Flow Rate

The licensee did not consider a single worst failure of an SI train in its LTCC EM. This is conservative with respect to refill and reflood PCT since the failure means it takes longer to quench the core and PCT will be higher. However, it is non-conservative with respect to LTCC PCT since the extra train of SI causes the RWST to drain faster, which in turn causes complete core blockage sooner and at a higher decay heat power. Because the licensee treated the ECCS flow rate in a manner which would result in a higher than PCT than would occur if the failure of an ECCS train were assumed, the NRC staff finds this treatment of the uncertainty in the ECCS flow rate conservative and, therefore, the ECCS flowrate uncertainty is acceptable.

ECCS Injection Temperature

The licensee used higher than expected ECCS temperatures during both the SI phase (i.e., phases 1 and 2) and the recirculation phase (i.e., phases 3 and 4). The ECCS injection temperatures are evaluated in Section A.4.2.1.4, "Initial and Boundary Conditions." The NRC found that the licensee's use of higher than expected temperatures for the ECCS injection was a conservative assumption. Therefore, the licensee's treatment of the uncertainty associated with ECCS injection temperature is acceptable and, therefore, the uncertainty is acceptable.

Core Barrel-Baffle Bypass Fraction

The licensee modeled the core barrel-baffle bypass as fully blocked which is a conservative assumption such that no cooling can reach the core through this flowpath. Because this assumption is conservative (for more details, see Section A.4.2.4.7, "Sensitivity Studies"), the NRC staff determined that the licensee's treatment of the uncertainty associated with the core barrel-baffle bypass fraction is acceptable and, therefore, the uncertainty is acceptable.

Core Blockage Fraction

The licensee initiated blockage at the time of sump switchover, and assumed 100 percent blockage 360 seconds later. Because 360 seconds is a conservatively early estimate of the time it would take to completely block the core and because assuming the entire core is blocked is a conservative assumption, as it reduces the ability for coolant to flow into the core, the NRC staff found that the licensee's treatment of the uncertainty associated with core blockage fraction and time to core blockage is acceptable; therefore, the uncertainty is acceptable.

Plant Set Points and Delays

The licensee used nominal values (i.e., the values given in the technical specifications) for the various plant set points and delays. Because these values are consistent with the technical specification values, the NRC staff found that the licensee's treatment of the uncertainty associated with plant set points and delays is acceptable; therefore, the uncertainty is acceptable.

CCFL Parameters

The CCFL parameters were evaluated in in Section A.4.2.3.4, "Validation of the Closure Relationships." The NRC staff determined that the CCFL model chosen conservatively predicts CCFL by reducing the mixing between the core and the upper plenum below what would be reasonably expected. Therefore, the NRC staff finds the uncertainty associated with CCFL parameters is acceptable.

Because the licensee addressed the key sources of uncertainty as discussed above, and because with these uncertainties addressed, there is significant margin to the PCT limit, the NRC staff determined that important sources of uncertainty are identified and accounted for in an appropriate manner. These analyses demonstrate that the combined code and application uncertainty would be less than the design margin for the PCT (i.e., there is ample margin to the actual PCT limit). The NRC staff concludes that this criterion is satisfied.

A.4.2.5.2 Experimental Uncertainty

> **Experimental Uncertainty**
>
> *The uncertainties in the experimental data base should be addressed. Data sets and correlations with experimental uncertainties that are too large when compared to the requirements for evaluation model assessment should not be used.*
>
> SRP Section 15.0.2, Subsection III.3e

The licensee did not perform a direct assessment of experimental uncertainty during its review since it is incumbent on the NRC staff to perform this assessment. The NRC staff considered the results of a number of experiments from a published document (Reference 31) to address uncertainty in support of the CCFL closure model. For the CCFL study reviewed, the NRC staff determined that since the experiments conducted by different researchers at different facilities and times were found to be repeatable and similar (approximately 10 percent difference), then there is no need to further evaluate the experimental uncertainties of the provided data. The NRC staff concludes that this criterion is satisfied.

A.4.2.5.3 Calculated and Predicted Results

> **Calculated and Predicted Results**
>
> *For separate effects tests and integral effects tests, the differences between calculated results and experimental data for important phenomena should be quantified for bias and deviation.*
>
> SRP Section 15.0.2, Subsection III.3e

Test data is not available to provide validation for the LTCC EM so the NRC staff used engineering judgment to conclude that the EM would produce an adequate prediction of the underlying phenomena. This is the specific focus of Section A.4.2.4.2, "Validation of the Evaluation Model," and, holistically, of the LTCC EM safety evaluation.

For CCFL, the review was more a review of what coefficients were most applicable based on existing experiments, than a review of the experimental data itself. As no separate effects or integral effects tests were used to support the validation of the LTCC EM, the NRC staff concludes that this criterion does not apply to this review.

A.4.2.6 Quality Assurance Program

The QAP covers, in part, the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the LTCC EM. The QAP also ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

Section 15.0.2, Subsection III.3.f of the SRP contains three review criteria for the QAP. The review criteria topics and the subsection providing the NRC staff's reviews are listed in Table 10.

Table 10: Quality Assurance Plan Review Categories

Subsection	
A.4.2.6.1	Appendix B Quality Assurance Program
A.4.2.6.2	Quality Assurance Documentation
A.4.2.6.3	Independent Peer Review

A.4.2.6.1 Appendix B Quality Assurance Program

Appendix B Quality Assurance Program

The evaluation model should be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.

SRP Section 15.0.2, Subsection III.3f

The licensee provided a discussion that the EM was maintained under a QAP that meets the requirements of 10 CFR 50 Appendix B in response to SNPB-3-19 (References 16 and 19) and SNPB-3-23 through SNPB-3-29 (Reference 18). In its response to SNPB-3-23, the licensee confirmed that the RELAP5-3D analysis performed for the LTCC EM was in compliance with the STP Appendix B QAP. In its response to SNPB-3-24, the licensee confirmed that the input values are compared against the reference source and are controlled using the QAP.

The licensee provided details on the correct installation and execution of RELAP5-3D in response to SNPB-3-19 (References 16 and 19). In that response, the licensee confirmed that there were no technical differences between the Idaho National Laboratory's results (i.e., the creators of RELAP5-3D) and the licensee's results for the given set of set cases.

The NRC staff noted that Volume 5, Section 2.2.4.2 of the RELAP5-3D theory manual (Reference 9) describes how RELAP5-3D results should be analyzed, and specifically discusses the following criteria:

- Code convergence
- Non-physical results
- Realistic results
- Boundary conditions are well behaved
- Thoroughly understood results

In its responses to SNPB-3-25 through SNPB-3-29 (Reference 18), the licensee provided a discussion of how the QAP would assure each of the above criteria for the RELAP5-3D results. In its response to SNPB-3-25, the licensee stated that proper convergence is often ensured by requiring a qualified analyst to perform the analysis, adding that the "mass error" is typically a good figure of merit to observe to ensure convergence. In response to SNPB-3-26 through SNPB-3-29, the licensee stated that ensuring that the results are physically appropriate and realistic, that the boundary conditions behave as prescribed, and that the results are thoroughly understood are also considerations of the analyst performing the analysis. The NRC staff considers that such reliance on qualified analysts to be common industry practice, and it is also one of the reasons why an independent peer review is part of the QAP. For the NRC's evaluation of the peer review, see Section A.4.2.6.3, "Independent Peer Review."

Initially, the NRC staff's RAI questions were written assuming that the LTCC EM would be used generically for future analyses. However, during the course of the review, the licensee changed its methodology and the NRC staff also changed its review strategy to limit the scope as discussed in Section A.3.2, "Scope of the Review." Future use of the LTCC EM would require additional review and acceptance by the NRC staff, as the NRC only reviewed the licensee's application of its QAP to the RELAP5-3D analyses as appropriate for this limited use by the licensee.

Because the licensee provided a process which addresses quality assurance and because the NRC staff verified aspects of the licensee's application of its QAP, the NRC staff finds that a simulation performed under the program would achieve a sufficient quality for use in reactor safety licensing analysis. The NRC staff concludes that this criterion is satisfied.

A.4.2.6.2 Quality Assurance Documentation

Quality Assurance Documentation

The quality assurance program documentation should include procedures that address all of these areas [design control, document control, software configuration control and testing, and corrective actions].

SRP Section 15.0.2, Subsection III.3f

The licensee provided justification that its QAP documentation includes procedures to address all relevant areas in response to SNPB-3-30 (Reference 18). In its response, the licensee referenced the response to SNPB-3-23, which stated that the QAP for the simulations requires compliance with all the elements of the STP 10 CFR 50 Appendix B program including: error reporting, qualification for personnel, software quality assurance, and records management.

Because the licensee is using a QAP consistent with its 10 CFR 50 Appendix B program, the NRC staff finds that the QAP documentation for RELAP5-3D includes procedures to address all relevant areas of interest. The NRC staff concludes that this criterion is satisfied.

A.4.2.6.3 Independent Peer Review

Independent Peer Review

Independent peer reviews should be performed at key steps in the evaluation model development process.

SRP Section 15.0.2, Subsection III.3f

The licensee provided justification that an independent peer review was performed at key steps in the execution of the LTCC EM in response to SNPB-3-31 (Reference 18). In its response, the licensee clarified that the QAP specified separate roles for the preparer and the checker in generating the simulations, both of whom must be fully qualified for performing the procedure in question.

The NRC staff finds that the QAP appropriately applied an independent peer review because the licensee confirmed that multiple layers of review were conducted by different individuals. Further, the licensee submitted the simulation results to the NRC staff for an additional independent peer review with acceptable results. The NRC staff concludes that this criterion is satisfied.

A.4.3 Conclusions

The NRC staff made the following conclusions based on the referenced evaluations provided in this In-Vessel Thermal-Hydraulic Analysis:

- Based on the staff's evaluation in Section A.4.2.1, "Accident Scenario Identification Process," the NRC staff finds that that the accident scenario identification process is a structured process and is appropriately used to identify the key figures of merit for the accident.

- Based on the staff's evaluation in Section A.4.2.2, "Documentation," the NRC staff finds that the documentation provided is sufficient to describe the LTCC EM.

- Based on the staff's evaluation in Section A.4.2.3, "Evaluation Model Development," the NRC staff finds that the individual field equations and closure relationships are adequate for modeling the phenomena determined to be important for the chosen scenario.

- Based on the staff's evaluation in Section A.4.2.4, "Code Assessment," the NRC staff finds that the code assessment demonstrates that the LTCC EM is adequate for analyzing the chosen scenario.

- Based on the staff's evaluation in Section A.4.2.5, "Uncertainty Analysis," the NRC staff finds that the uncertainties in the inputs and models are appropriately accounted for such that the results are expected to bound possible outcomes for the accident.

- Based on the staff's evaluation in Section A.4.2.6, "Quality Assurance Program," the NRC staff finds that the STPNOC QAP assures all relevant actions in the development and maintenance of the EM have been taken.

Based on the above, the NRC staff concludes that the STPNOC LTCC EM and the simulations performed specifically for STP, Units 1 and 2, provide a conservative analysis for debris impacts on LTCC for hot-leg breaks 16 inches in diameter and smaller. Further, the simulations performed with this EM along with those from LOCADM demonstrates that the stated acceptance criteria from WCAP-16793 have been satisfied:

- The EM used for the LTCC analysis demonstrates that the maximum clad temperature remains at the saturation temperature and therefore shall not exceed 800 °F following core quench and re-flooding.

- The LOCADM analysis demonstrates that the thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The NRC staff's conclusions in the LTCC EM are specific to STP and the analysis performed. Future use of this in-vessel thermal-hydraulic EM was not considered, because use of the EM outside the simulations reviewed could invalidate the NRC staff's conclusions. If the input and modeling assumptions are made less conservative, such as decreasing the severity of accident conditions (e.g., the amount of debris generated, post-LOCA PCT or oxide thickness), or if the relative importance of specific models or flow paths are changed, then the NRC staff's assessment is no longer applicable.

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