EMF-2310(NP)(A)
Revision 1

SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors

May 2004

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DOCUMENT SYSTEM
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Framatome ANP, Inc.

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EMF-2310(NP)
Revision 1

SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors

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Concurred:	James Mallay J. F. Mallay, Director Regulatory Affairs	7/9/03 Date
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Approved:	J Cudlin J. J. Cudlin, Manager Analysis Services/PWR Safety Analysis	07/09/03 Date
Approved:	Dan Pruitt D. W. Pruitt, Manager Safety Analysis Methods	7/3/2003 Date
Approved:	R E Collingham R. E. Collingham, Manager BWR Reload Engineering and Methods Development	7/7/2003 Date

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U.S. Nuclear Regulatory Commission
Report Disclaimer

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This document contains a total of 352 pages (including the pages before item 1 above).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 19, 2004

Mr. James F. Mallay
Director, Regulatory Affairs
Framatome ANP
3815 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT EMF-2310(P), REVISION 1, "SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR PRESSURIZED WATER REACTORS" (TAC NO. MC0329)

Dear Mr. Mallay:

On August 12, 2003, Framatome ANP (FANP) submitted Topical Report (TR) EMF-2310(P), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," to the Nuclear Regulatory Commission (NRC) staff. On March 17, 2004, an NRC draft safety evaluation (SE) regarding our approval of EMF-2310(P) was provided for your review and comments. By letter dated April 1, 2004, FANP commented on the draft SE. A call was held on May 3, 2004, to discuss the staff disposition of the comments. The staff's disposition of FANP's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The staff has found that EMF-2310(P) is acceptable for referencing in licensing applications for Westinghouse and Combustion Engineering designed pressurized water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed SE. The SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC's TR website, we request that FANP publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, draft SE comments, and original TR pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the TR identification symbol.

If the NRC's criteria or regulations change so that its conclusions in this letter, that the TR is acceptable, are invalidated, FANP and/or the licensees referencing the TR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the TR without revision of the respective documentation.

Sincerely,

/RA/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Safety Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

EMF-2310(P), REVISION 1, "SRP CHAPTER 15 NON-LOCA METHODOLOGY

FOR PRESSURIZED WATER REACTORS"

FRAMATOME ANP

PROJECT NO. 728

1.0 INTRODUCTION

By letter dated August 12, 2003 (Reference 1), Framatome ANP (FANP) requested review and approval for referencing in licensing actions Topical Report (TR) EMF-2310(P), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," in particular EMF-2310, Section 5.6, "CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)."

The noted section has been revised to address the dilution front model used when the residual heat removal (RHR) system is in operation, all control rods are inserted in Modes 4 and 5, and complete mixing of the fluid is assumed prior to entry of the diluted fluid into the core.

EMF-2310(P) methodology incorporates S-RELAP5 as the systems analysis code and was previously reviewed and approved by the NRC staff for application to Chapter 15 non-loss-of-coolant accident (non-LOCA) events on May 11, 2001 (Reference 2).

2.0 REGULATORY BASIS

The regulatory bases for the boron dilution events are found in the General Design Criteria (GDC) (Reference 3) and the Standard Review Plan (SRP) (Reference 4). The specific applicable GDCs are:

(1) GDC 10, *Reactor Design*

(2) GDC 15, *Reactor Coolant System Design*

(3) GDC 26, *Reactivity Control System Redundancy and Capability*

The applicable SRP Section is 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)."

3.0 TECHNICAL EVALUATION

FANP has revised Section 5.6 of EMF-2310(P), Revision 0 in three areas. Each is discussed below.

3.1 The Dilution Front Model will be Used when the RHR System is in Operation

When one or more reactor coolant pumps are operating it is assumed that complete, instantaneous mixing of boron with the reactor coolant system (RCS) water occurs. Section 3.3 of this safety evaluation discusses this further. For modes where the RHR, or shutdown cooling system, is in operation, flow rates may not be sufficient to assure complete mixing of the reactor coolant system. Under these conditions the mixing front approach is applied.

The mixing front approach assumes that the diluent mixes with the RCS and results in reduced boron concentration at the mixing location. The dilution is then viewed as a series of dilution fronts progressing through the RCS. Dilution mixture transit time to the bottom of the core is based on the volume and the flow rates of both the diluent and RCS flows. The result is that dilution flows are fully mixed in the lower plenum prior to entrance into the core.

The NRC staff has reviewed the model as presented in EMF-2310(P), Revision 1, Section 5.6, and finds it acceptable. If operator action is required to terminate the transient, the time to dilution below the critical concentration must provide sufficient margin that the operator has the following times to take corrective action:

(a) During refueling: 30 minutes.

(b) During startup, cold shutdown, hot standby, and power operation: 15 minutes.

3.2 All Control Rods will be Assumed to be Inserted in Modes 4 and 5

Control rod insertion is permitted in Modes 4 and 5, but during refueling operations the analysis must assume withdrawal of all control rods. This is stated in SRP Section 15.4.6, Acceptance Criteria, parameter assumption (vi).

FANP has stated that if a plant has procedures that increase the shutdown boron requirements to compensate for a stuck rod, then the critical boron concentration is determined assuming that all rods are inserted for Modes 4 and 5. Otherwise, the critical boron concentration is determined using the assumption that the most reactive rod is stuck in the fully withdrawn position.

The NRC staff finds this consistent with the requirements of GDC 26 and the guidance of SRP Section 15.4.6, Acceptance Criteria and, therefore, is acceptable.

3.3 Complete Mixing of the Fluid is Assumed Prior to Entry of the Diluted Fluid into the Core

Support of the complete mixing model is based on supporting calculations performed with the STAR-CD computational fluid dynamics (CFD) code for the International Standard Problem ISP-43. ISP-43 is a voluntary participation problem of a test performed at the University of

Maryland 2x4 Thermal-Hydraulic Loop. The test was performed by holding the vessel coolant at a constant temperature of 347K (165°F) while injecting water into one cold leg. Mixing was determined through thermocouple measurements. Boron was not injected in this test, but the measure of success in predicting the test is to predict the temperature distribution as measured by the exit of the downcomer.

Results of the STAR-CD simulation indicate very close agreement with the test data over most of the range of the test. The initial temperature, the end state temperature, and time of the end state temperature are predicted very accurately. There is a few percent difference in the slope of the temperature decay as the entering fluid mixes. The difference is not significant, however, and demonstrates that the complete mixing assumption is valid for the flow conditions in the test.

FANP, in Attachment A of Reference 1, has stated that "[t]he analysis of a boron dilution event depends on the rate of dilution and the plant design. The plant layout dictates whether the dilution can be treated symmetrically or asymmetrically.... If the charging line for residual heat removal flow is not in the same cold leg as the dilution flow, or if the RHR flow is distributed across the other cold legs, the boron dilution event is asymmetrical." FANP review of the specific application of the EMF-2310(P) methodology must be performed to ensure the situation warrants use of the complete mixing assumption.

4.0 CONDITIONS

The parameters and assumptions used in the analysis should be suitably conservative. The following values and assumptions, as delineated in SRP Section 15.4.6, are considered acceptable, and should be evaluated if appropriate:

(1) For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent, or justified amount, to account for power-measurement uncertainty.

(2) The boron dilution is assumed to occur at the maximum possible rate.

(3) The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

(4) All fuel assemblies are installed in the core.

(5) A conservatively low value is assumed for the reactor coolant volume.

(6) For analyses during refueling, all control rods are withdrawn from the core.

(7) For analyses during power operation, the minimum shutdown margin allowed by the technical specifications is assumed to exist prior to the initiation of boron dilution.

(8) For each event analyzed, a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

(9) Conservative scram characteristics are assumed, i.e., maximum delay time with the most reactive rod held out of the core.

5.0 CONCLUSIONS

The NRC staff concludes that the FANP methodology described in this TR is capable of addressing the thermal-hydraulic response of the boron dilution event in a conservative manner and is, therefore, approved for reference in licensing actions.

6.0 REFERENCES

(1) Letter from Framatome ANP to NRC, Requesting Review of EMF-2310(P) Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," August 12, 2003 (ADAMS Accession No. ML032460852).

(2) Letter from NRC to Framatome ANP, Acceptance for Referencing of Licensing Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology For Pressurized Water Reactors" (TAC No. MA7192), May 11, 2001 (ADAMS Accession No. ML033580677).

(3) Title 10 of the *Code of Federal Regulations* Appendix A to Part 50, General Design Criteria for Nuclear Power Plants.

(4) NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, April 1996.

Attachment: Resolution of Comments

Principal Contributor: Ralph Landry

Date: May 19, 2004

RESOLUTION OF COMMENTS

ON DRAFT SAFETY EVALUATION FOR EMF-2310, REVISION 1,

"SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR

PRESSURIZED WATER REACTORS"

By letter dated April 1, 2004, Framatome ANP (FANP) provided comments on the draft safety evaluation (SE) for EMF-2310, Revision1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors." A call was held with FANP on May 3, 2004, to discuss the staff's disposition of the comments. The following is the staff's resolution of those comments.

1. FANP Comment: Delete the last two sentences of the last paragraph of Section 3.3, and replace with Insert 1.

   NRC Action: This comment was partially adopted into the final SE, as agreed upon during the May 3, 2004, call between FANP and NRC. The last sentence was deleted. The second to last sentence was left as modified by inserting "FANP" as the first word.

2. FANP Comment: Delete Section 4.0, and replace with Insert 2. The boron dilution event analysis does not use the code S-RELAP5. FANP requests that the conditions in Section 4.0 of the SE be deleted since they are primarily related to the presumed use of S-RELAP5 for the analysis of the boron dilution event.

   NRC Action: The comment was partially adopted into the final SE, as agreed upon during the May 3, 2004, call between FANP and NRC. The first paragraph and conditions 1, 2, and 3, were deleted. The second paragraph was reworded to state "The parameters and assumptions used in the analysis should be suitably conservative. The following values and assumptions, as delineated in SRP Section 15.4.6, are considered acceptable, and should be used if appropriate."

   The 9 parameters were not deleted, because they are assumptions in SRP Section 15.4.6. The rewording of the second paragraph meets the intent of FANP's comment to delete the parameters.

3. FANP Comment: FANP considered that sufficient information was provided in the submittal to justify the use of the complete mixing model under asymmetric conditions, and requests that the NRC specifically approve the use of the complete mixing model for this situation by replacing the conclusion paragraph of Section 5.0 with Insert 3.

   NRC Action: The comment was partially adopted into the SE, as agreed upon during the May 3, 2004, call between FANP and NRC. The conclusion paragraph of Section 5.0 now states: "The NRC staff concludes that the FANP methodology described in this TR is capable of addressing the thermal-hydraulic response of the boron dilution event in a conservative manner and is, therefore, approved for reference in licensing actions."

4. FANP Comment: FANP proposes (for the purpose of clarity) to modify the first sentence in Topical Report EMF-2310(P), Revision 1, Section 5.6 to read: "The analysis of the boron dilution event does not use the system code S-RELAP5." This modification will be made in the approved version of the topical report.

   NRC Action: This modification is acceptable to the NRC.

FRAMATOME ANP, Inc.

August 12, 2003
NRC:03:044

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Request for Review of EMF-2310(P) Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

Ref.: 1. Letter, J. F. Mallay (Framatome ANP) to Document Control Desk (NRC), "Interim Report of Evaluation of a Deviation Pursuant to 10 CFR 21.21(a)(2)," NRC:01:017, April 27, 2001.

Ref.: 2. Letter, J. F. Mallay (Framatome ANP) to Document Control Desk (NRC), "Closure of Interim Report 01-001, 'Boron Dilution Analyses - Instantaneous Mixing and Dilution Front Models'," NRC:02:008, January 31, 2002.

Framatome ANP requests the NRC's review and approval for referencing in licensing actions the topical report EMF-2310(P) Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors." One CD containing a proprietary version of the report and one CD containing a non-proprietary version of the report are enclosed. We request that the NRC approve this report by December 31, 2003.

Section 5.6 "CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)" of EMF-2310(P)(A) Revision 0 is the only portion of the approved topical report which has been revised. This section has been modified to address the three items listed below.

1. The dilution front model will be used when the RHR system is in operation.

2. All control rods will be assumed to be inserted in modes 4 and 5.

3. Complete mixing of the fluid is assumed prior to entry of the diluted fluid into the core. This assumption has been validated by computational fluid dynamics (CFD) calculations.

Framatome ANP provided an interim report of a deviation pursuant to 10 CFR 21.21(a)(2) in Reference 1. The deviation involved the methodology for evaluation of a boron dilution event described in EMF-2310(P)(A) Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors." In Reference 2, Framatome ANP reported that it had completed the evaluation of the deviation and had concluded the deviation was not reportable under 10 CFR 21. Framatome ANP also determined that the boron dilution methodology described in

EMF-2310(P)(A) Revision 0 needed to be revised. Section 5.6 of the topical report has been revised to reflect three assumptions to be used in future analyses.

The technical justification for changes 1 and 2 is provided in the topical report and the technical justification for change 3 (assumption of complete mixing of the fluid at the entrance to the core) is provided in Attachment A to this letter.

Framatome ANP considers some of the information contained in the enclosed report to be proprietary. As required by 10 CFR 2.790(b), an affidavit is enclosed to support the withholding of the information from public disclosure.

Very truly yours,

James F. Mallay

James F. Mallay, Director
Regulatory Affairs

Enclosures

cc: D. G. Holland
E. S. Peyton
Project 728

A F F I D A V I T

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is James F. Mallay. I am Director, Regulatory Affairs, for Framatome ANP ("FANP"), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by FANP to determine whether certain FANP information is proprietary. I am familiar with the policies established by FANP to ensure the proper application of these criteria.

3. I am familiar with the FANP information provided with the letter number NRC:03:044 dated August 12, 2003, and referred to herein as “Document.” Information contained in this Document has been classified by FANP as proprietary in accordance with the policies established by FANP for the control and protection of proprietary and confidential information.

4. This Document contain information of a proprietary and confidential nature and is of the type customarily held in confidence by FANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure.

6. The following criteria are customarily applied by FANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FANP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for FANP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for FANP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by FANP, would be helpful to competitors to FANP, and would likely cause substantial harm to the competitive position of FANP.

7. In accordance with FANP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside FANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FANP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 12th day of August, 2003.

Ella F. Carr-Payne

Ella F. Carr-Payne
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/05

ELLA F. CARR-PAYNE
Notary Public
Commonwealth of Virginia
My Commission Exps. Aug. 31, 2005

Attachment A
Evaluation of Mixing during a Boron Dilution Event

Introduction and Summary

The analysis of a boron dilution event depends on the rate of dilution and the plant design. The plant layout dictates whether the dilution can be treated symmetrically or asymmetrically. In an asymmetric boron dilution event, the dilution takes place in only one of the cold legs of a PWR. If the charging line for residual heat removal flow is not in the same cold leg as the dilution flow, or if the RHR flow is distributed across the other cold legs, the boron dilution event is asymmetrical.

Treatment of the asymmetric behavior of boron dilution becomes a consideration only when the reactor coolant system flow rate becomes small enough that the instantaneous mixing model can no longer be applied. For conditions where the dilution front model applies, the behavior of asymmetric boron dilution is no different than the symmetric boron dilution event if the diluted sector is adequately mixed with the undiluted sectors at the core inlet.

The purpose of this analysis is to demonstrate that sufficient mixing exists to permit the symmetric dilution front model to be applied under asymmetric boron dilution conditions. The criterion used to demonstrate adequate mixing is to show that at least 85 percent mixing of the diluted flow with the remainder of the reactor coolant has occurred at the core inlet using a computational fluid dynamics code.

Analysis Results

Computational fluid dynamics (CFD) has been used to determine the degree of steady-state mixing during an inadvertent initiation of a charging pump resulting in a boron dilution event. CFD has been shown to adequately predict dilution and STAR-CD has been benchmarked by Framatome ANP against an International Standard Problem, ISP-43 (discussed later), to confirm the applicability of the model to boron dilution.

A simple model is built of a typical PWR that does not include any of the internals. A cross sectional view of a typical model is shown in Figure 1. All cells are fluid cells and the model is extended past the lower plenum to allow the use of a pressure boundary at the exit.

The STAR-CD default turbulence model is used and the fluid properties are held constant with no buoyancy effects. The dilution of borated water is simulated by thermal mixing. Thus, the property of interest is the temperature of the region. That is to say, the boron concentration is not modeled, only the fluid enthalpy as mixing takes place. This approach is consistent with ISP-43. The analysis is performed as a steady-state analysis where cold water is introduced into one leg of the plant with RHR cooling flow equally distributed in all cold legs.

The result of importance is the temperature profile at the exit of the lower plenum. An example of the temperature distribution is shown in Figure 2. For a 4-loop plant the region is divided into quadrants, for a 3-loop plant it is divided into thirds. The average temperature is calculated for each region and the percent mixing is determined based on the ratio of the worst case sector temperature to the perfectly mixed temperature; i.e.,

$$\%Mixed = (1 - \frac{T_{totally\_mixed} - T_{worst\_case}}{T_{totally\_mixed} - T_{cold\_leg}}) * 100$$

The results of the mixing calculations are summarized in Table 1.

The use of the 85 percent value to show satisfactory mixing is based on the following considerations. The CFD model used to analyze the mixing is conservative. It does not account for mixing caused by buoyancy effects that result from density differences as a function of boron concentration and temperature. The modeling of ISP-43 showed that buoyancy effects enhanced the mixing, and inclusion of these effects was necessary to obtain good agreement with the experiment. The model also does not account for mixing caused by wake turbulence and jets from support structures in the lower plenum, or the mixing that results from flow interaction with the lower tie plate and core support plate below the core. Finally, in the LOFT boron dilution experiments, L6-6, the experimentalists went to considerable lengths to set up the experiment to be as isothermal as possible, and to eliminate loop flow effects that would enhance mixing by conducting the experiment with the reactor coolant system drained so that the surface of the water was at the level of the tube sheet. Despite these measures, the estimate of the time to criticality was significantly underestimated – by 30% – as a consequence of unanticipated mixing. Because of the inherent conservatisms in the CFD boron dilution model and calculation, it is reasonable to assume that complete mixing has taken place at the core inlet if the sector average mixing from the CFD simulation exceeds 85 percent.

<u>Benchmark Results</u>

The nuclear industry routinely uses standard problems to demonstrate the adequacy of computer codes for predicting plant transients. The Committee on the Safety of Nuclear Installations within the Organization for Economic Cooperation and Development (OECD) sponsors these International Standard Problems and they are based on tests performed in research facilities around the world. One of these International Standard Problems, ISP-43, was used to demonstrate the ability of computational fluid dynamics (CFD) techniques to predict the boron mixing in the downcomer of a reactor vessel.

The ISP-43 test was performed at the University of Maryland 2x4 Thermal-Hydraulic Loop. This loop is a scale model of the Three Mile Island Unit 2 pressurized water nuclear reactor. The test was performed by holding the reactor coolant in the reactor vessel at 347 K and injecting water into one cold leg at a colder temperature and then using thermocouples to determine the mixing. Note that no boron was injected into the vessel; mixing was determined strictly by the change in temperature. The measure of success was the ability to predict the transient average temperature as measured at the exit of the downcomer, labeled in the test as Level 4.

STAR-CD's meshing program, Pro-am, was used to model the vessel. This program permitted the sizing of the cells needed to model the flow phenomena through various area changes. The model contains about 640,000 hexahedral cells. A cross-sectional view is shown in Figure 3. STAR-CD's AMG solver and the RNG k-epsilon turbulence model were chosen.

The benchmark transient calculation was performed using two different treatments for fluid modeling. In the first case, constant fluid properties (independent of temperature) were used and buoyancy was turned off. (This treatment is the one used for analysis of the boron dilution event.) The second case uses temperature dependent viscosity and density, and the buoyancy is optionally turned on. When comparing these modeling assumptions to the measurements of ISP-43, it is found that the first case conservatively underestimates the amount of mixing and the second case, with buoyancy included, agrees with the experimental results. A comparison of the two cases with the benchmark test results at Level 4 is shown in Figure 4.

Table 1 Summary of Results

Plant Type	RHR Flow (gpm)	Charging Flow (gpm)	Mixing %
3 Loop Westinghouse	2800	250	88.3
	1200	132	88.4
	1200	321	93.1
4 Loop Westinghouse	2800	250	97.2
	1200	132	95.1
CE	3000	147	86.9
	600	98	93.3

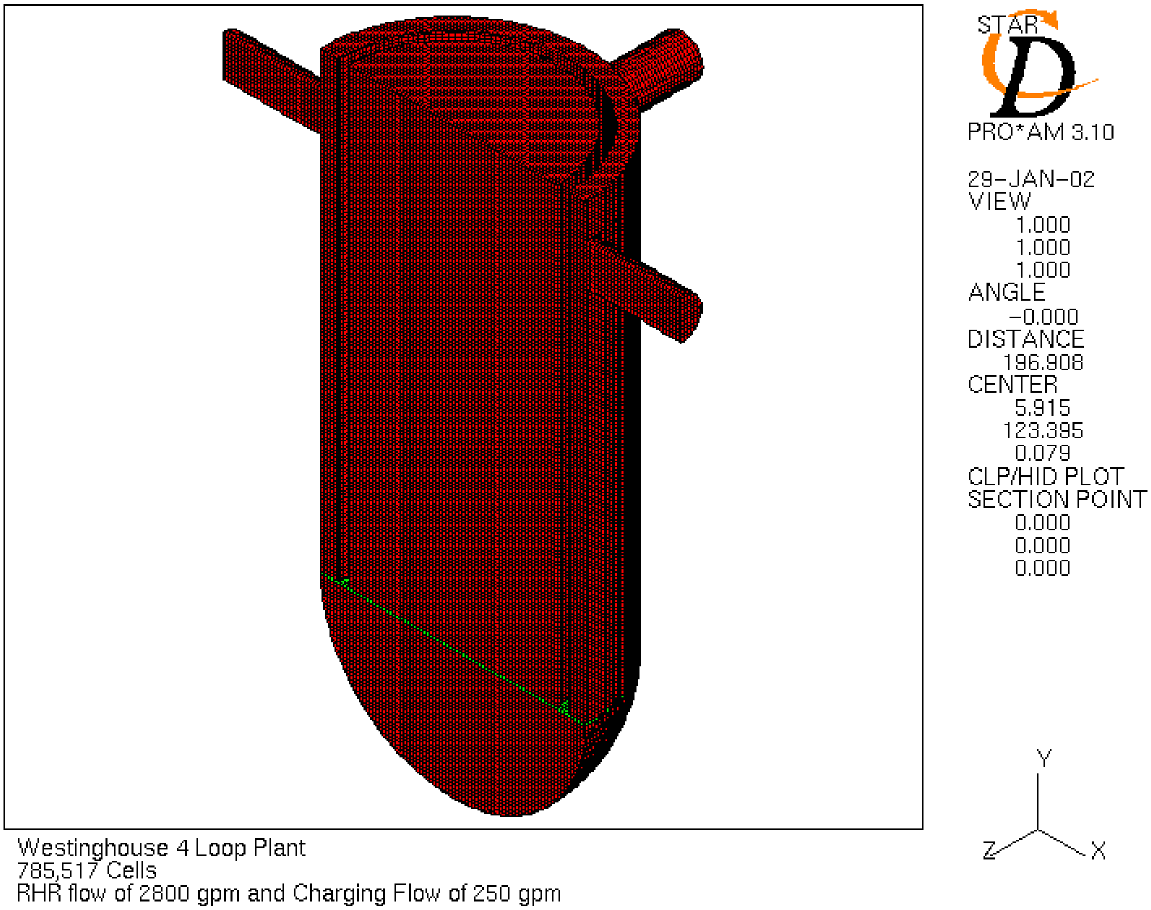

Figure 1 Cross Section of the Vessel Model

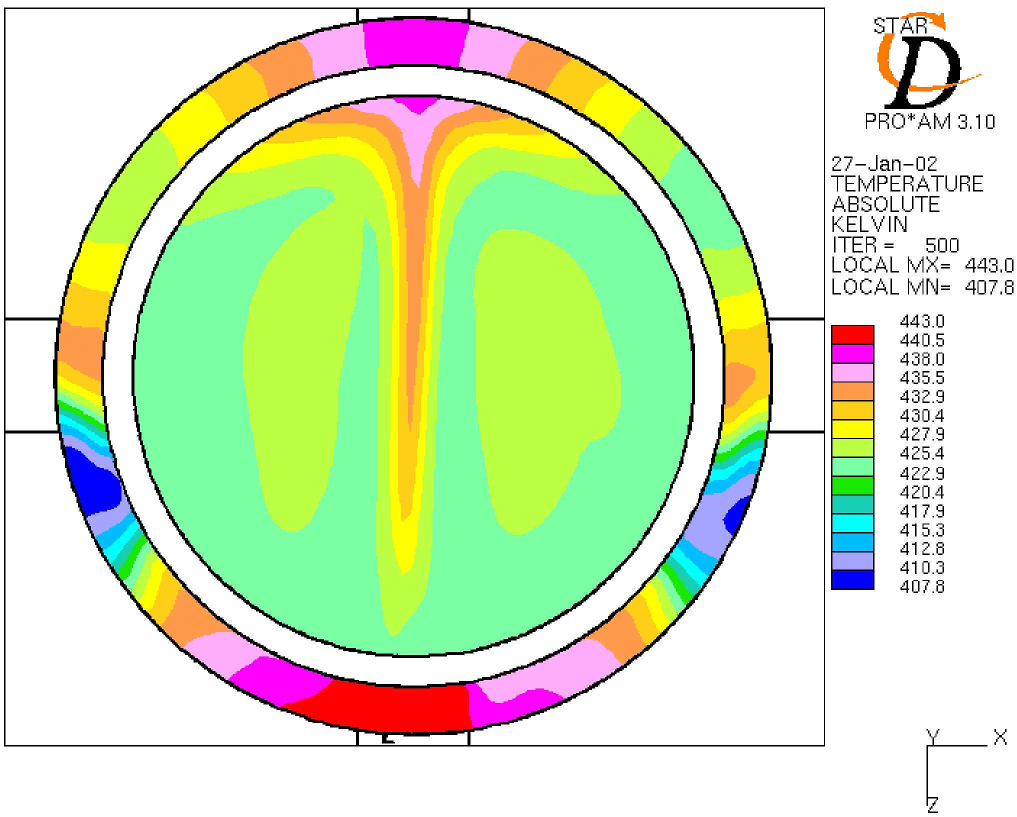

Figure 2 Temperature Distribution for the Lower Plenum

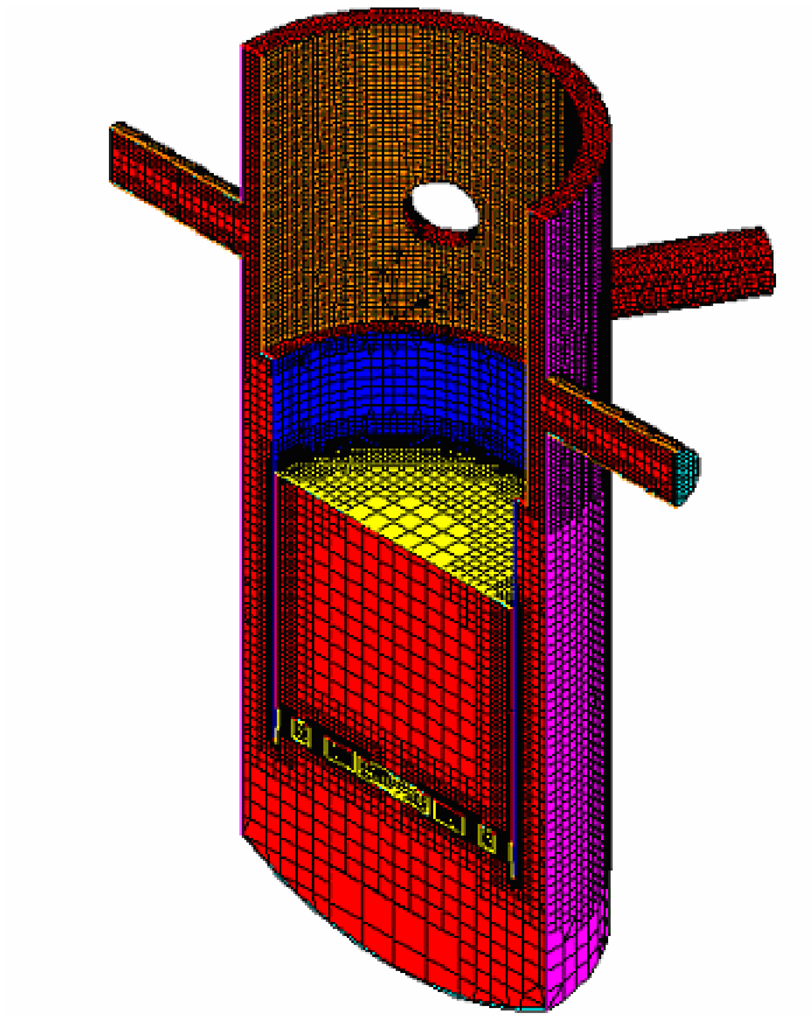

Figure 3 Clipped View of the ISP-43 Test Vessel Model

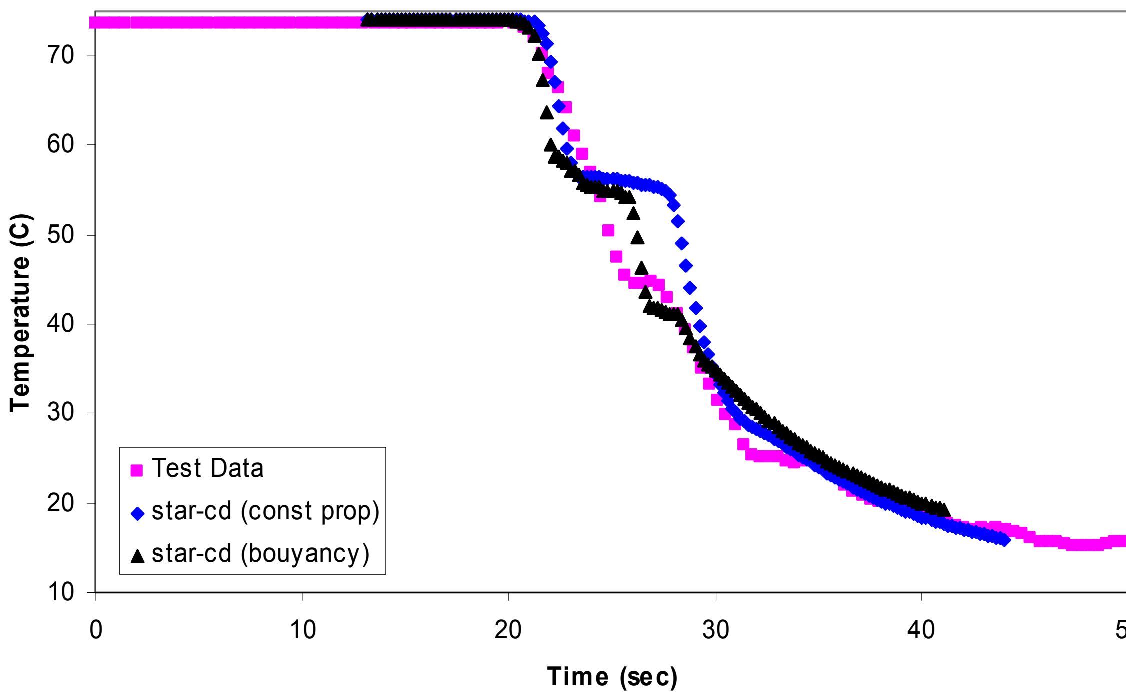

Figure 4 Test Results Compared to Analysis

FRAMATOME ANP

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April 1, 2004
NRC:04:016

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Topical Report EMF-2310(P), Revision 1, "SRP, Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors."

Ref.: 1. Letter, Stephen Dembek (NRC) to James F. Mallay (Framatome ANP), "Draft Safety Evaluation for Topical Report EMF-2310(P), Revision 1, 'SRP, Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors', (TAC No. MC0329)" March 17, 2004.

Ref.: 2. Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), "Request for Review of EMF-2310(P), Revision 1, 'SRP, Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors'," NRC:03:044, August 12, 2003.

Framatome ANP has reviewed the draft SER provided in Reference 1. We request changes to the draft SER primarily in two areas: use of the complete mixing model and the fact that S-RELAP5 is not used for the boron dilution event.

First, we believe that sufficient information was provided in the Reference 2 submittal to justify the use of the complete mixing model under asymmetric conditions. Framatome ANP requests that the NRC specifically approve the use of the complete mixing model for this situation.

Second, the boron dilution event analysis does not use the code S-RELAP5. We request that the conditions in Section 4.0 of the SER be deleted since they are primarily related to the presumed use of S-RELAP5 for the analysis of the boron dilution event. Comments on each of the nine conditions are provided below.

Condition 1 – For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent, or justified amount, to account for power-measurement uncertainty.

Comment 1 – This condition is only applicable if the code S-RELAP5 is used.

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Condition 2 – The boron dilution is assumed to occur at the maximum possible rate.

Comment 2 – This is a requirement which is stated in the topical report itself. The maximum unborated water charging rate is assumed for the analysis.

Condition 3 – The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution.

Comment 3 - This condition is only applicable if the code S-RELAP5 is used. The severity of the boron dilution event in not significantly dependent on moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution because the pertinent phenomenon is the fluid mixing. A requirement to maximize the boron concentration is stated in the topical report itself. This requirement meets the intent of this condition.

Condition 4 – All fuel assemblies are installed in the core.

Comment 4 – The calculations assume that all of the fuel assemblies are in the core to determine the core reactivity. This is part of the methodology and does not require a condition to invoke it.

Condition 5 – A conservatively low value is assumed for the reactor coolant volume.

Comment 5 - This is a requirement which is stated in the topical report itself. The minimum RCS volume is assumed for the analysis.

Condition 6 – For analyses during refueling, all control rods are withdrawn from the core.

Comment 6 – The requirement in the topical report is that the minimum shutdown margin for refueling is assumed in the analyses. During refueling, this shutdown margin is preserved by a "refueling boron" concentration. The plant licensing basis defines the acceptable control rod configuration that this refueling boron must protect. Consequently, acceptable control rod configurations during refueling range from rods withdrawn to inserted depending on the plant. The method already accounts for the necessary control rod assumption (i.e., as defined by the plant licensing basis) by relying on the minimum shutdown margin.

Condition 7 – For analyses during power operation, the minimum shutdown margin allowed by the technical specifications is assumed to exist prior to the initiation of boron dilution.

Comment 7 - This is a requirement which is stated in the topical report itself. The minimum shutdown margin is assumed for the analysis in each mode of operation, including power operation.

Condition 8 – For each event analyzed, a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

Comment 8 – For each event analyzed, a conservatively high reactivity addition rate is assured by maximizing the flow rate. The functional relationship between differential boron worth and boron concentration is incorporated into the analysis by determining the initial and critical boron concentrations using a neutronics simulator, which inherently accounts for changes in differential boron worth. This is part of the methodology and does not require a condition to invoke it.

Condition 9 – Conservative scram characteristics are assumed, i.e., maximum delay time with the most reactive rod held out of the core.

Comment 9 - This condition is only applicable if the code S-RELAP5 is used.

Framatome ANP proposes (for the purpose of clarity) to modify the first sentence in the topical report EMF-2310(P) Revision 1, section 5.6 to read, "The analysis of the boron dilution event does not use the system code S-RELAP5." This modification will be made in the approved version of the topical report.

A copy of the draft SER is presented in Attachment A to this letter which shows the proposed deletions to the draft SER. The locations of three inserts to the draft SER are also shown in attachment A. The content of the inserts is shown in Attachment B to this letter.

Very truly yours,

James F. Mallay, Director
Regulatory Affairs

Enclosures

cc: M. C. Honcharik
R. R. Landry
Project 728

Attachment A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 17, 2004

Mr. James F. Mallay
Director, Regulatory Affairs
Framatome ANP
3815 Old Forest Road
Lynchburg, VA 24501

SUBJECT: DRAFT SAFETY EVALUATION FOR TOPICAL REPORT EMF-2310(P), REVISION 1, "SRP, CHAPTER 15 NON-LOCA METHODOLOGY FOR PRESSURIZED WATER REACTORS" (TAC NO. MC0329)

Dear Mr. Mallay:

By letter dated August 12, 2003, Framatome ANP submitted Topical Report (TR) EMF-2310(P), Revision 1, "SRP, Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," to the staff for review. Enclosed for Framatome ANP's review and comment is a copy of the staff's draft safety evaluation (SE) for the TR.

Pursuant to 10 CFR 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After ten working days, the draft SE will be made publicly available and an additional ten working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

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

If you have any questions, please contact Michelle C. Honcharik at 301-415-1774.

Sincerely,

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Draft Safety Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

EMF-2310(P), REVISION 1, "SRP CHAPTER 15 NON-LOCA METHODOLOGY

FOR PRESSURIZED WATER REACTORS"

FRAMATOME ANP

PROJECT NO. 728

1.0 INTRODUCTION

By letter dated August 12, 2003 (Reference 1), Framatome ANP (FANP) requested review and approval for referencing in licensing actions Topical Report (TR) EMF-2310(P), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," in particular EMF-2310, Section 5.6 "CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)."

The noted section has been revised to address the dilution front model used when the residual heat removal (RHR) system is in operation, all control rods are inserted in Modes 4 and 5, and complete mixing of the fluid is assumed prior to entry of the diluted fluid into the core.

EMF-2310(P) methodology incorporates S-RELAP5 as the systems analysis code and was previously reviewed and approved by the NRC staff for application to Chapter 15 non-loss-of-coolant accident (non-LOCA) events on May 11, 2001 (Reference 2).

2.0 REGULATORY BASIS

The regulatory bases for the boron dilution events are found in the General Design Criteria (GDC) (Reference 3) and the Standard Review Plan (SRP) (Reference 4). The specific applicable GDCs are:

(1) GDC 10, *Reactor Design*

(2) GDC 15, *Reactor Coolant System Design*

(3) GDC 26, *Reactivity Control System Redundancy and Capability*

The applicable SRP Section is 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)."

3.0 TECHNICAL EVALUATION

FANP has revised Section 5.6 of EMF-2310(P), Revision 0 in three areas. Each is discussed below.

3.1 The Dilution Front Model will be Used when the RHR System is in Operation

When one or more reactor coolant pumps are operating it is assumed that complete, instantaneous mixing of boron with the reactor coolant system (RCS) water occurs. Section 3.3 of this safety evaluation discusses this further. For modes where the RHR, or shutdown cooling system, is in operation, flow rates may not be sufficient to assure complete mixing of the reactor coolant system. Under these conditions the mixing front approach is applied.

The mixing front approach assumes that the diluent mixes with the RCS and results in reduced boron concentration at the mixing location. The dilution is then viewed as a series of dilution fronts progressing through the RCS. Dilution mixture transit time to the bottom of the core is based on the volume and the flow rates of both the diluent and RCS flows. The result is that dilution flows are fully mixed in the lower plenum prior to entrance into the core.

The NRC staff has reviewed the model as presented in EMF-2310(P), Revision 1, Section 5.6, and finds it acceptable. If operator action is required to terminate the transient, the time to dilution below the critical concentration must provide sufficient margin that the operator has the following times to take corrective action:

(a) During refueling: 30 minutes.

(b) During startup, cold shutdown, hot standby, and power operation: 15 minutes.

3.2 All Control Rods will be Assumed to be Inserted in Modes 4 and 5

Control rod insertion is permitted in Modes 4 and 5, but during refueling operations the analysis must assume withdrawal of all control rods. This is stated in SRP Section 15.4.6, Acceptance Criteria, parameter assumption (vi).

FANP has stated that if a plant has procedures that increase the shutdown boron requirements to compensate for a stuck rod, then the critical boron concentration is determined assuming that all rods are inserted for Modes 4 and 5. Otherwise, the critical boron concentration is determined using the assumption that the most reactive rod is stuck in the fully withdrawn position.

The NRC staff finds this consistent with the requirements of GDC 26 and guidance of SRP Section 15.4.6, Acceptance Criteria and, therefore, is acceptable.

3.3 Complete Mixing of the Fluid is Assumed Prior to Entry of the Diluted Fluid into the Core

Support of the complete mixing model is based on supporting calculations performed with the STAR-CD computational fluid dynamics (CFD) code for the International Standard Problem ISP-43. ISP-43 is a voluntary participation problem of a test performed at the University of

Maryland 2x4 Thermal-Hydraulic Loop. The test was performed by holding the vessel coolant at a constant temperature of 347K (165°F) while injecting water into one cold leg. Mixing was determined through thermocouple measurements. Boron was not injected in this test, but the measure of success in predicting the test is to predict the temperature distribution as measured by the exit of the downcomer.

Results of the STAR-CD simulation indicate very close agreement with the test data over most of the range of the test. The initial temperature, the end state temperature, and time of the end state temperature are predicted very accurately. There is a few percent difference in the slope of the temperature decay as the entering fluid mixes. The difference is not significant, however, and demonstrates that the complete mixing assumption is valid for the flow conditions in the test.

FANP, in Attachment A of Reference 1, has stated that "[t]he analysis of a boron dilution event depends on the rate of dilution and the plant design. The plant layout dictates whether the dilution can be treated symmetrically or asymmetrically....If the charging line for residual heat removal flow is not in the same cold leg as the dilution flow, or if the RHR flow is distributed across the other cold legs, the boron dilution event is asymmetrical." ~~Review of the specific application of the EMF-2310(P) methodology must be performed to ensure the situation warrants use of the complete mixing assumption. The complete mixing model cannot be used under asymmetric conditions.~~ Insert 1

4.0 CONDITIONS

The NRC staff notes that a generic TR describing a code such as S-RELAP5 cannot provide full justification for each specific individual plant application. When a license amendment is necessary in order to use the S-RELAP5 based methodology, the individual licensee or applicant must provide justification for the specific application of the code which is expected to include:

(1) Nodalization: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

(2) Chosen Parameters and Conservative Nature of Input Parameters: A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the TR approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation. Insert 2

(3) Calculated Results: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses of the reactor vessel peak pressure.

The parameters and assumptions used in the analytical model should be suitably conservative. The following values and assumptions are considered acceptable:

(1) For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent, or justified amount, to account for power-measurement uncertainty.

(2) The boron dilution is assumed to occur at the maximum possible rate.

(3) The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

(4) All fuel assemblies are installed in the core.

(5) A conservatively low value is assumed for the reactor coolant volume.

(6) For analyses during refueling, all control rods are withdrawn from the core.

(7) For analyses during power operation, the minimum shutdown margin allowed by the technical specifications is assumed to exist prior to the initiation of boron dilution.

(8) For each event analyzed, a conservatively high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

(9) Conservative scram characteristics are assumed, i.e., maximum delay time with the most reactive rod held out of the core.

5.0 CONCLUSIONS

~~The NRC staff concludes that the S-RELAP5 code is capable of addressing the thermal-hydraulic response of the boron dilution event in a conservative manner and is, therefore, approved for reference in licensing actions.~~

Insert 3

6.0 REFERENCES

(1) Letter from Framatome ANP to NRC, Requesting Review of EMF-2310(P) Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," August 12, 2003 (ADAMS Accession No. ML032460852).

(2) Letter from NRC to Framatome ANP, Acceptance for Referencing of Licensing Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology For Pressurized Water Reactors" (TAC No. MA7192)," May 11, 2001 (ADAMS Accession No. ML033580677).

(3) Title 10 of the *Code of Federal Regulations* Appendix A to Part 50, General Design Criteria for Nuclear Power Plants.

(4) NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, April 1996.

Principal Contributor: Ralph Landry

Date:

Attachment B

Insert 1

The STAR-CD analyses provide adequate support for the assumption of complete mixing of the fluid prior to the entry of the diluted fluid into the core under asymmetric conditions.

Insert 2

Framatome ANP should review the applicability of the complete mixing assumption for each specific application of the methodology.

Insert 3

The NRC has reviewed the boron dilution event analysis methodology as presented in EMF-2310(P) Revision 1 section 5.6 and finds it acceptable.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 11, 2001

Mr. James F. Mallay
Director, Regulatory Affairs
Framatome ANP, Richland, Inc.
2101 Horn Rapids Road
Richland, WA 99352

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT EMF-2310(P), REVISION 0, "SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR PRESSURIZED WATER REACTORS" (TAC NO. MA7192)

Dear Mr. Mallay:

The NRC staff has completed its review of Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" submitted by Framatome ANP Richland, Inc. (FRA-ANP, previously known as Siemens Power Corporation (SPC)) on November 22, 1999, and supplemented by letter dated January 26, 2001.

On the basis of our review, the staff finds the subject report to be acceptable for referencing in license applications to the extent specified, and under the limitations delineated in the report, and in the enclosed safety evaluation (SE). The SE defines the basis for NRC acceptance of the report.

Pursuant to 10 CFR 2.790, we have determined that the enclosed SE does not contain proprietary information. However, we will delay placing the SE in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.790.

The staff will not repeat its review and acceptance of the matters described in the report, when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with the procedures established in NUREG-0390, the NRC requests that FRA-ANP publish accepted versions of the report, including the safety evaluation, in the proprietary and non-proprietary forms within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an "-A" (designating accepted) following the report identification symbol. The accepted versions shall also incorporate all communications between FRA-ANP and the staff during this review.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, FRA-ANP and the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or to submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Stuart A. Richards, Director
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Safety Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT EMF-2310(P), REVISION 0

"SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR

PRESSURIZED WATER REACTORS"

PROJECT NO. 702

1.0 INTRODUCTION

Framatome ANP Richland Inc. (FRA-ANP), formerly known as Siemens Power Corporation (SPC) submitted Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" (Reference 1), on November 22, 1999 (Reference 2), for NRC review and approval for application of the S-RELAP5 thermal-hydraulic analysis computer code (Reference 3), to Chapter 15 non loss-of-coolant accident (LOCA) transients. The application to use S-RELAP5 is as a replacement for the NRC approved code ANF-RELAP (References 4 and 5). S-RELAP5 is an updated version of ANF-RELAP. The application of S-RELAP5 to the analysis of the small-break LOCA (SBLOCA) under the guidance of 10 CFR Part 50, Appendix K was previously approved by the staff (Reference 6).

The stated goal of FRA-ANP is to apply a single computer code to the analysis of both LOCA and non-LOCA transient events. The code of choice is to be one that has had wide industry acceptance and application. To achieve this goal the decision was made to modify the approved ANF-RELAP code in such a way as to bring it up to a standard that incorporates the thermal-hydraulic code RELAP5/MOD2 (Reference 7), the fuel design code RODEX2 (Reference 8), along with codes specifically needed for LOCA analysis into a single system calculation. In so doing the RELAP5/MOD2 code was modified to include selected models from the RELAP5/MOD3 code (Reference 9), improved numerics, and models necessary to satisfy the requirements of 10 CFR Part 50, Appendix K, though not necessary for the non-LOCA transient events.

The XCOBRA-IIIC code (Reference 10), will continue to be used to obtain the final predicted Minimum Departure from Nucleate Boiling Ratio (MDNBR) for each non-LOCA transient event. The core conditions calculated for the reactor coolant system (RCS) by S-RELAP5 will be used as input to the XCOBRA-IIIC core and subchannel methodology to predict the event-specific MDNBR.

2.0 CODE APPLICABILITY

During the course of review of S-RELAP5 for application to SBLOCA transients, extensive examination of the code numerics, two-fluid equations, heat transfer, point kinetics, and general assessment took place. At that time requests for additional information (RAIs) were developed relating to the code itself (Reference 11), and responded to by FRA-ANP (Reference 12). Meetings were held with the Advisory Committee on Reactor Safeguards (ACRS) and Thermal-Hydraulic Phenomena Subcommittee regarding the modeling within S-RELAP5. The meetings with the ACRS and the reviews conducted by its members and their consultants were considered in the preparation of the staff's RAIs.

The RELAP5 computer code is a light water reactor transient analysis code developed for the NRC for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for nuclear power plant analyses. RELAP5 is a general purpose code that, in addition to calculating the behavior of a RCS during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable gas, and solute. The RELAP5 code is based on a nonhomogeneous and nonequilibrium model for the two-phase system. Solution is by a partially implicit numerical scheme to permit economical calculation of system transients. The objective of the RELAP5 development effort was to produce a code that included important first-order effects necessary for accurate prediction of system transients but that was sufficiently simple and cost effective so that parametric or sensitivity studies were possible.

The code includes many generic component models from which general systems can be simulated. These component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control and trip system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, counter-current flow limit (CCFL), boron tracking, and noncondensable gas transport. The code also incorporates many user conveniences such as extensive input checking, free-form input, internal plot capability, restart, renodalization, and variable output edits.

The S-RELAP5 code evolved from FRA-ANP's ANF-RELAP code, a modified RELAP5/MOD2 version used by FRA-ANP for pressurized water reactor (PWR) plant licensing analyses that included the SBLOCA analysis, steam line break analysis, and PWR non-LOCA Updated Final Safety Analysis Report (UFSAR) Chapter 15 event analyses. During the modifications to permit realistic analyses, enhancements were made to incorporate the requirements of 10 CFR Part 50, Appendix K SBLOCA analysis. The code structure was also modified to be similar to that of RELAP5/MOD3. This included incorporation of the RELAP5/MOD3 reactor kinetics, control systems and trip systems models.

Some of the major modifications made to RELAP5/MOD2 and ANF-RELAP to produce the S-RELAP5 code include the following:

1. <u>Multi-Dimensional Capability</u> - Two-dimensional treatment has been added to the hydrodynamic field equations. This capability can handle the Cartesian and cylindrical coordinate systems.

2. Energy Equations - The energy equations were modified to conserve the energy transported into and out of a control volume, thus correcting the tendency of the RELAP5 codes to produce an energy error when a large pressure gradient exists between two adjacent control volumes.

3. Numerical Solution of Hydrodynamic Field Equations - Where the RELAP5 codes use a Gaussian elimination solver to reduce the hydrodynamic finite-difference equations to a pressure equation, S-RELAP5 uses algebraic manipulation.

4. State of Steam-Noncondensable Mixture - At very low steam quality, the ideal gas equation is used for both steam and noncondensable gas. This permits calculation of state relations for both steam and noncondensable gas at low steam quality and also the presence of pure noncondensable gas below the ice point.

5. Hydrodynamic Constitutive Models - Significant modifications were made to the RELAP5 interphase friction and interphase mass transfer models. Some of the flow regime (two-phase flow) transient criteria were modified to be consistent with published data. Transient flow regimes are introduced for smoothing the constituent models. Most of the RELAP5/MOD2 partition functions were only slightly modified if at all. A more accurate wall friction factor approximation replaces the Colebrook equation.

6. Heat Transfer Model - The RELAP5/MOD2 use of different heat transfer correlations in reflood was eliminated. The Dittus-Boelter single phase steam heat transfer correlation was replaced with the Sleicher-Rouse correlation which gives higher steam temperatures and has a smaller uncertainty range.

7. Choked Flow - The Moody critical flow model was implemented for 10 CFR Part 50, Appendix K calculations. The modification of ANF-RELAP to use an iterative scheme to compute the equation of state at the choked plane rather than using the previous time step information was also implemented.

8. Counter-Current Flow Limit - The Kutateladze type CCFL correlation of ANF-RELAP was replaced with the Bankoff form. This conforms with RELAP5/MOD3.

9. Component Models - The pump model includes the EPRI pump performance degradation data, and the pump head term in the fluid field equations was made more implicit. The ICECON containment code was incorporated to run concurrently with S-RELAP5. User guidelines were implemented to specify a replacement procedure for modeling the accumulator model.

10. Fuel Models - The RODEX2 fuel deformation and conductivity models were incorporated for SBLOCA applications. The flow diversion model of TOODEE2 was implemented to account for the effect of cladding rupture on heat transfer. The Baker-Just metal-water reaction model was implemented as required by 10 CFR Part 50, Appendix K.

11. Code Architecture - Modifications were made to bring the S-RELAP5 code into conformance with the description of the RELAP5/MOD3 code architecture. This

includes writing the code in FORTRAN 77 and maintaining a common source for all computer versions.

Many of the above noted modifications made in the development of S-RELAP5 are not applicable to non-LOCA transient analysis but were reviewed during the overall code review performed for the application to the SBLOCA. The non-LOCA transient events are described in Reference 13. Specific event application of S-RELAP5 is given in Table 1.

Table 1

Applicable SRP Chapter 15 Events

Event	SRP No.
15.1 - Increase in Heat Removal by the Secondary System	
Decrease in Feedwater Flow	15.1.1
Increase in Feedwater Flow	15.1.2
Increase in Steam Flow	15.1.3
Inadvertent Opening of Steam Generator Relief/Safety Valve	15.1.4
Steam System Piping Failures Inside and Outside Containment	15.1.5
15.2 - Decrease in Heat Removal by Secondary System	
Loss of Outside External Load (LOEL)	15.2.1
Turbine Trip	15.2.2
Loss of Condenser Vacuum	15.2.3
Closure of Main Steam Isolation Valve	15.2.4
Steam Pressure Regulator Failure	15.2.5
Loss of Non-Emergency AC Power to the Station Auxiliaries	15.2.6
Loss of Normal Feedwater (LONF) Flow	15.2.7
Feedwater System Piping Breaks Inside and Outside Containment	15.2.8
15.3 - Decrease in Reactor Coolant Flow Rate	
Loss of Forced Reactor Coolant Flow (LOCF)	15.3.1
Flow Controller Malfunctions	15.3.2
Reactor Coolant Pump (RCP) Rotor Seizure	15.3.3
RCP Shaft Break	15.3.4

Table 1
(continued)

15.4 - Reactivity and Power Distribution Anomalies	
Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal From a Subcritical or Low Power Startup Condition	15.4.1
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2
RCCA Misoperation	15.4.3
Dropped Rod/Bank	15.4.3.1
Single Rod Withdrawal	15.4.3.2
Statically Misaligned RCCA	15.4.3.3
Startup of an Inactive Loop at an Incorrect Temperature	15.4.4
Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease of Boron Concentration (Boron Dilution)	15.4.6
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (Misloaded Assembly)	15.4.7
Spectrum of Rod Ejection Accidents	15.4.8
15.5 - Increase In Reactor Coolant Inventory	
Inadvertent Operation of the Emergency Core Cooling System that Increases Reactor Coolant Inventory	15.5.1
CVCS Malfunction that Increases Reactor Coolant Inventory	15.5.2
15.6 - Decreases in Reactor Coolant Inventory	
Inadvertent Opening of a Pressurizer Pressure Relief Valve	15.6.1
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	15.6.2
Radiological Consequences of Steam Generator Tube Rupture (SGTR)	15.6.3

Application of S-RELAP5 is to be made to Combustion Engineering (CE) 2x4 plants and Westinghouse 3 and 4-loop plants.

3.0 STAFF APPROACH TO REVIEW

The staff performed an extensive review of the S-RELAP5 code during the review of the code for application to the SBLOCA events. Review for application to non-LOCA transient events focused on the results of assessment cases and comparison to calculations performed using the approved ANF-RELAP code.

4.0 CODE ASSESSMENT

The assessment of S-RELAP5 for application to Chapter 15 non-LOCA transient events consists of calculation of four Loss-of-Fluid Test (LOFT) program transient tests and comparison with ANF-RELAP calculations of various transients. The four LOFT tests that were calculated are:

- LOFT L6-1 Loss of Load
- LOFT L6-2 Loss of Primary Flow
- LOFT L6-3 Excessive Steam Load
- LOFT L6-5 Loss of Feedwater

The four LOFT tests represent SRP Category 15.1 - Increase in Heat Removal by the Secondary System; SRP Category 15.2 - Decrease in Heat Removal by Secondary System; and SRP Category 15.3 - Decrease in Reactor Coolant Flow Rate events.

The major parameters for each of the tests were predicted well in both trend, timing and magnitude. For the loss of load case, the steam generator and pressurizer levels were underpredicted (conservative) by both S-RELAP5 and ANF-RELAP, while both the hot leg and cold leg temperatures were conservatively overpredicted. For the loss of primary flow case, again the hot and cold leg temperatures were overpredicted. For the case of excessive steam load, the secondary and pressurizer levels were conservatively predicted and the reactor power prediction was within 2.5 percent of that measured at the peak. Again, the hot and cold leg temperatures were conservatively predicted. For the loss of feedwater case, the steam generator secondary side liquid level showed a small oscillation about the measured value, a low pressurizer pressure prediction and conservative predictions of hot and cold leg temperatures and steam generator steam flow.

Overall, the predictions of the LOFT transient tests show good agreement with the measured results and good comparison with the calculated results of ANF-RELAP. The simulations performed by the code included modeling of automatic control components and systems such as pressurizer sprays and heaters, feedwater control, pressure control, steam generator level control, and reactor power.

Chapter 15 transient events are to be analyzed considering the following:

- Timing of Loss-of-Offsite Power
- Mitigating Systems
- Operator Actions

- Single Failures

- Number of Loops Operating

- Axial and Radial Power Distributions

In performing the analyses, the analyst is expected to select values and equipment for the above in accordance with the guidelines provided in the appropriate Regulatory Guides, Standard Review Plan, and computer code user guides to ensure conservative calculations are performed. In addition, the analyst must ascertain where it is appropriate to use nominal or technical specification values for the initial core power level, initial reactor coolant flow rate, initial reactor coolant temperature, initial reactor pressure and pressurizer level, moderator temperature reactivity coefficient, Doppler reactivity coefficient, reactor protection system trip and equipment setpoints and delay times, and scram characteristics. The sample problems provided being of a generic nature, assume nominal values for these parameters for the given plant designs being analyzed.

The comparisons with event-specific ANF-RELAP calculations included both CE and Westinghouse design transients. The events analyzed fall into two major categories: Anticipated Operational Occurrences and Postulated Accidents. Specific sample problems provided in Reference 1, include the main steam line break (MSLB), both pre- and post-scram, LOEL, LONF, LOCF, the uncontrolled rod bank withdrawal at power, and SGTR.

The staff finds that the assessment performed in support of the S-RELAP5 application to Chapter 15 non-LOCA events is adequate in that it compares code results with ANF-RELAP results for the selected LOFT transients and for plant calculations. Specific plant applications may still require additional supporting assessment calculations should plant specific features or conditions be outside the range of the generic assessments.

5.0 EVALUATION OF S-RELAP5

The staff's SE on the ANF-RELAP code application to Chapter 15 non-LOCA events (Reference 5), identified six restrictions on use of the code. The staff notes the following regarding those restrictions:

- The stated application of the S-RELAP5 code is for the events listed above in Table 1. There are other computer codes and methodologies employed for evaluation of the events not listed in the table. For each licensing basis event analyzed, the applicant must, as always, justify the methodology used whether by reference to S-RELAP5 or whatever methodology has been used.

- Analysis of Chapter 15, Events 15.6.2 and 15.6.3, on radiological consequences is beyond the scope of the S-RELAP5 computer code. However, where primary and secondary mass and energy release are used as the principal source of the radiological components of the event, S-RELAP5 is capable of providing that information.

- The S-RELAP5 documentation provides support for use of the code in cases for which upper head voiding occurs and cases where the boron tracking model is used.

- The S-RELAP5 documentation provides sufficient support for use of the code in cases for which natural circulation cooldown is to be calculated.

- As is the case in reviewing all generic topical report applications, submittals for specific plants and events must include justification of the nodalization used, input parameters, options selected, and all of the parameters that influence the progression of the event and its mitigation.

- The S-RELAP5 code incorporates the RODEX2 fuel analysis code. Although the application of the S-RELAP5 code to Chapter 15 non-LOCA events uses a conservative input value of the fuel rod gap conductance, should it be necessary, or desirable, an analyst does have available the RODEX2 capability as an integral part of the S-RELAP5 code.

The staff finds that the modifications performed in developing the S-RELAP5 code satisfy the restrictions that had been placed on the ANF-RELAP code when applied to Chapter 15 non-LOCA events. The staff encourages and supports efforts to develop methodologies capable of analyzing a broad spectrum of events rather than separate methodologies for each event which must be analyzed. The staff also notes, however, that a generic topical report describing a code such as S-RELAP5 cannot provide full justification for each specific individual plant application. The individual applicant must still provide justification for the specific application of the code which is expected to include as a minimum, the nodalization, defense of the chosen parameters, any needed sensitivity studies, justification of the conservative nature of the input parameters, and calculated results.

The MSLB is one of the most challenging non-LOCA events for a PWR. The analysis performed with the S-RELAP5 code provides the thermal-hydraulic response of the RCS combined with the RELAP5 point kinetics model. A detailed fuel failure calculation is performed by using the thermal-hydraulic conditions predicted with detailed fuel and cladding calculations from XCOBRA-IIIC (Reference 10) and a neutronics code for the highest powered fuel assemblies. Since the report under review is intended to be generic in nature, an extensive evaluation is performed to determine the most important phenomena affecting the CE configuration as well as the Westinghouse configuration. The calculation is then broken down into two phases: pre-scram and post-scram. The calculation is further complicated in having to separate those plants which still have high boron concentration storage tanks from those which have removed the tanks. The neutronic response is significantly different for the cases with or without high concentration boron injection.

Although the remainder of the transients for which S-RELAP5 is to be applied are less severe than the MSLB, FRA-ANP evaluates the important phenomena for the non-LOCA events to determine that the code is capable of predicting the phenomenological response for the plant being analyzed. Although this is not a phenomena identification and ranking table in the strictest sense, the goal is achieved in showing that the important phenomena are adequately represented.

The staff concludes that the S-RELAP5 code, with conservative input modeling assumptions, is capable of addressing the thermal-hydraulic response of the target non-LOCA events in a conservative manner in keeping with the staff's SRP guidance and is, therefore, an acceptable replacement for the ANF-RELAP code.

6.0 CONCLUSIONS

The staff supports the efforts of applicants to integrate codes for analysis of accidents and transients rather than manual transfer of information between the codes. Integrating the thermal-hydraulic, fuel rod performance, and other codes, permits a smoother and more accurate prediction of the performance of the system under accident conditions.

The staff finds that the modifications performed in developing the S-RELAP5 code satisfy the restrictions that had been placed on the ANF-RELAP code when applied to Chapter 15 non-LOCA events. The staff encourages and supports efforts to develop methodologies capable of analyzing a broad spectrum of events rather than separate methodologies for each event which must be analyzed. The staff also notes, however, that a generic topical report describing a code such as S-RELAP5 cannot provide full justification for each specific individual plant application. The individual applicant must still provide justification for the specific application of the code which is expected to include as a minimum, the nodalization, defense of the chosen parameters, any needed sensitivity studies, justification of the conservative nature of the input parameters, and calculated results.

The staff finds that the assessment performed in support of the S-RELAP5 application to Chapter 15 non-LOCA events is adequate in that it compares code results with ANF-RELAP results for the selected LOFT transients and for plant calculations. Specific plant applications may still require additional supporting assessment calculations should plant specific features or conditions be outside the range of the generic assessments.

The staff concludes that the S-RELAP5 code is capable of addressing the thermal-hydraulic response of the target non-LOCA events in a conservative manner and is, therefore, an acceptable replacement for the ANF-RELAP code.

7.0 REFERENCES

1. EMF-2310(P), Rev. 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Siemens Power Corporation, November 1999.

2. Letter, SPC to NRC, dated November 22, 1999.

3. EMF-2100(P), Rev. 2, "S-RELAP5 Models and Correlations Code Manual," Siemens Power Corporation, January 2000.

4. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992.

5. Letter, NRC to Siemens Nuclear Power Corporation, "Acceptance for Referencing of Topical Report ANF-89-151(P), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," dated March 16, 1992.

6. Letter, NRC to Framatome ANP, Richland, Inc., "Acceptance for Referencing of Licensing Topical Report EMF-2328(P), Revision 0, 'PWR Small Break LOCA Evaluation Model, S-RELAP5 Based'," dated March 15, 2001.

7. NUREG/CR-4312, EGG-2396, Rev. 1, "RELAP5/MOD2 Code Manual," March 1987.

8. XN-NF-81-58(P)(A), Rev. 2, Supplements 1 and 2, "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984.

9. NUREG/CR-5535, INEL-95/0174, "RELAP5/MOD3 Code Manual," August 1995.

10. XN-75-21(P)(A), Rev. 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Exxon Nuclear Company, January 1986.

11. Letter NRC to SPC dated December 11, 2000.

12. Letter SPC to NRC dated January 26, 2001.

13. NUREG-0800, "USNRC Standard Review Plan," U. S. Nuclear Regulatory Commission, Washington, DC 20555, July 1981.

Principal Contributor: R. R. Landry

Date: May 11, 2001

SIEMENS

November 22, 1999
NRC:99:048

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Request for Review of EMF-2310(P) Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"

Fifteen proprietary and 12 nonproprietary copies of topical report EMF-2310(P) Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" are being submitted to the NRC for review and acceptance for referencing in licensing actions. (NOTE: Three proprietary copies and one nonproprietary copy have been sent directly to Mr. N. Kalyanam).

The topical report describes a revised SPC Pressurized Water Reactor non-LOCA transient analysis methodology that incorporates S-RELAP5 as the systems analysis code in place of ANF-RELAP. The methodology applies to all PWR non-LOCA transients, including Main Steamline Break.

The objective in using S-RELAP5 is to apply a single advanced version of an industry recognized code for all analyses, including LOCA and non-LOCA events. Using a single code that has had extensive review permits the development of one base input deck for the analysis of all events for a particular application. The benefits of using a single code include ease of use by engineers, reduced maintenance requirements, improved quality of both code and applications, and reduction of resources for the NRC review of associated methodology.

It is requested that the NRC approve this report by September 30, 2000, to support plant analyses performed by SPC for its PWR customers.

Siemens Power Corporation considers some of the information contained in the enclosure to this letter to be proprietary. As required by 10 CFR 2.790(b), an affidavit is enclosed to support the withholding of this information from public disclosure.

Very truly yours,

James F. Mallay

James F. Mallay, Director
Regulatory Affairs

Enclosures

cc: Mr. R. Caruso
Mr. N. Kalyanam (w/Enclosures)
Mr. J. L. Wermiel
Project No. 702 (w/Enclosures)

Siemens Power Corporation

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AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

I, Jerald. S. Holm, being duly sworn, hereby say and depose:

1. I am Manager, Product Licensing, for Siemens Power Corporation ("SPC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with SPC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the SPC information included in report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," referred to as "Document" transmitted by letter NRC:99:048. Information contained in this Document has been classified by SPC as proprietary in accordance with the control system and policies established by SPC for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by SPC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document will not be disclosed or divulged.

6. This Document contains information which is vital to a competitive advantage of SPC and would be helpful to competitors of SPC when competing with SPC.

7. The information contained in the Document is considered to be proprietary by SPC because it reveals certain distinguishing aspects of SPC licensing methodology which secure competitive advantage to SPC for product optimization and marketability, and includes information utilized by SPC in its business which affords SPC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in this Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it valuable insights into SPC licensing methodology and would result in substantial harm to the competitive position of SPC.

9. This Document contains proprietary information which is held in confidence by SPC and is not available in public sources.

10. In accordance with SPC's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside SPC only as required and under suitable agreement providing for nondisclosure and limited use of the information.

11. SPC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. Information in this Document provides insight into licensing methodology developed by SPC. SPC has invested significant resources in developing the methodology as well as the strategy for this application. Assuming a competitor had available the same

background data and incentives as SPC, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as SPC.

13. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Gerald S Holm

SUBSCRIBED before me this 22nd day of November, 1999.

SUE M. GALPIN
COMMISSION EXPIRES
NOTARY PUBLIC
FEB 27, 2000
STATE OF WASHINGTON

Sue M. Galpin

Sue M. Galpin
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 02/27/00

SIEMENS

February 3, 2000
NRC:00:009

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

NRC Review of Siemens Power Corporation Topical Reports

Ref.: 1. References are listed on page 2.

Siemens Power Corporation submitted two topical reports for NRC review and approval in References 1 and 2. The methodologies described in the topical reports provided in these two references utilize the code S-RELAP5 to perform transient and safety analyses for PWRs. In support of these two submittals, SPC is providing copies of the S-RELAP5 theory manual (Reference 3), programmer's manual (Reference 4), and user's manual (Reference 5). Eight copies of References 3, 4, and 5 have been provided directly to the NRC project manager for SPC, Mr. N. Kalyanam. SPC is also providing with this letter a CD containing the code S-RELAP5 and specific test cases. The CD also contains codes and files necessary to execute the test cases. The Attachment to this letter provides a description of the contents of the CD and how to extract the information from the CD. The CD has been provided directly to N. Kalyanam.

The version of S-RELAP5 described in References 3, 4, and 5, and contained on the CD, is that used for the analyses presented in the report transmitted by Reference 2 (PWR SBLOCA Methodology). The code version used for the analyses in the report accompanying Reference 1 (PWR Non-LOCA Methodology) was different in one respect. The difference between the two S-RELAP5 versions is in the implementation of the RODEX2 models. Since the RODEX2 model is not utilized for the PWR Non-LOCA methodology, the results are not affected by the difference.

Siemens Power Corporation considers the attachment and the enclosure to this letter to be proprietary. As required by 10 CFR 2.790(b), an affidavit is enclosed to support the withholding of this information from public disclosure.

Very truly yours,

James F. Mallay

James F. Mallay, Director
Regulatory Affairs

Enclosure/Attachment

cc: R. Caruso
N. Kalyanam (w/Enclosures)
R. R. Landry
Project No. 702

Siemens Power Corporation

2101 Horn Rapids Road
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Tel: (509) 375-8100
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Ref.: 1. Letter, J. F. Mallay (SPC) to Document Control Desk (NRC), "Request for Review of EMF-2310(P) Revision 0, *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*," NRC:99:048, November 22, 1999.

Ref.: 2. Letter, J. F. Mallay (SPC) to Document Control Desk (NRC), "Request for Review of EMF-2328(P) Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*," NRC:00:002, January 10, 2000.

Ref.: 3. EMF-2100(P) Revision 2, *S-RELAP5 Models and Correlations Code Manual*, Siemens Power Corporation, January 2000.

Ref.: 4. EMF-2101(P) Revision 1, *S-RELAP5 Programmers Guide*, Siemens Power Corporation, December 1999.

Ref.: 5. EMF-CC-097(P) Revision 4, *S-RELAP5 Input Data Requirements*, Siemens Power Corporation, December 1999.

Contents of CD ROM for S-RELAP5

INTRODUCTION

This 'README' contains instructions for creating an executable version of S-RELAP5, instructions for creating 'xmgr' and 'r2dmx' executables for plotting purposes, and instructions for testing the installation. Also included are descriptions of the contents of the transmittal.

GETTING STARTED

To get started, place the file `'transmittal.Z'` in a suitable subdirectory on the target workstation and uncompress (type in: `uncompress transmittal`). The file 'transmittal' is a 'tar' file, so the next step is to untar the file (type in: `tar -xvf transmittal` at the command prompt). The directory contents should be similar to the following:

```
drwxr-x---  16 t3923     eng           1024 Feb  2 13:50 codes
drwxr-x---   4 t3923     eng             96 Jan 31 16:33 sample_problem
-rw-r-----   1 t3923     eng       33976320 Feb  2 16:40 transmittal
```

BUILDING S-RELAP5

Change directory to `'codes'`. The build script `'build_sr5'` has been tested under both c-shell and korn-shell on both HPUX-9.0 and 11.0 operating systems. If the machine being used operates under the bourne-shell, then switch to c-shell by typing in: `csh`.

Next, type in: `build_sr5`

This command builds S-RELAP5 and executes three sample problems interactively. The output can be redirected to local files by using the following commands:

`build_sr5 >&build_sr5.log &` (for c-shell)
or
`nohup build_sr5 &`
`mv nohup.out build_sr5.log` (for korn-shell)

The executable files will reside in subdirectory `bin`, and the sample problem output will be located in subdirectory `sample`.

BUILDING XMGR

These plotting utilities are included due to incompatibility of S-RELAP5 restart-plot files with the NRC's versions of xmgr and r2dmx. The script creates the plot utilities `xmgr` and `r2dmx` and stores them in the subdirectory `bin`. This script runs under c-shell exclusively. Therefore, to execute `build_xmgr`, type in the following commands:

```
csh
build_xmgr
```

This script is highly dependent on system and local libraries installed on the workstation being used and may abort without creating the necessary executables. If this happens, assistance from the system administrator will be required. An alternative approach would be to try using the `xmgr` and `r2dmx` executable files located in the subdirectory `executables`. The files in this subdirectory will execute on machines using HPUX-9.0 and HPUX-11.0 operating systems.

SAMPLE PROBLEMS

In addition to the check problems used in the build step, a small break LOCA deck and a non-LOCA deck are included with this transmittal. Change directory to `sample_problem` and there are two subdirectories and a script:

```
drwxr-x---   2 t3923      eng           1024 Feb  2 16:34 nonloca
-rwxr-x---   1 t3923      eng            130 Jan 27 10:50 run-sbloca-cases
drwxr-x---   2 t3923      eng           1024 Feb  2 16:37 sbloca
```

The script, which is provided for installation check-out purposes, runs a RODEX2 transient, an S-RELAP5 steady state, and a 100 second S-RELAP5 transient. Both directories contain input files, run scripts, and sample output files. The non-LOCA problem is the turbine trip example from EMF-2310(P), *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*. The SBLOCA sample problem is the one presented in EMF-2328(P), *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based.*

Non-LOCA

There are three steady state input decks and one transient deck. The script `runssi` runs the steady state decks and generates a restart file used to start the transient. The `runner` script runs the transient using `trn_in` as the input deck. The `'*.out'` files are provided for checking the installation. There will be small, but insignificant differences between the calculations. These differences are due to compiler differences between operating system versions.

```
-rwxr-x---   1 t3923      eng            360 Jan 27 11:31 runner
-rwxr-x---   1 t3923      eng            605 Jan 27 11:30 runssi
-rw-r-----   1 t3923      eng         638595 Jan 27 11:20 ss_in_1
-rw-r-----   1 t3923      eng           1244 Jan 27 11:21 ss_in_2
-rw-r-----   1 t3923      eng            212 Jan 27 11:21 ss_in_3
-rw-r-----   1 t3923      eng        3349545 Feb  2 15:17 test_ss1.out
-rw-r-----   1 t3923      eng        2844318 Feb  2 15:25 test_ss2.out
-rw-r-----   1 t3923      eng         666004 Feb  2 15:25 test_ss3.out
-rw-r-----   1 t3923      eng        1463598 Feb  2 15:32 test_trn.out
-rw-r-----   1 t3923      eng           7441 Jan 27 11:23 trn_in
```

SBLOCA

The SBLOCA methodology requires a binary file of rod data from RODEX2. This file is made by executing `rdx2lse` (see run script `rdx2.job`), the hp version of RODEX2 that writes the S-RELAP5 readable binary file `rodex2d`. This file must reside in the same directory as the

S-RELAP5 input file for steady state calculations. The `rodex2d` file is not needed for transient calculations.

S-RELAP5 uses the default RELAP5 format for file management; input resides on INPUT, printed output is written to OUTPUT, and the restart-plot file is written to RSTPLT. The script `runssi` shows the job flow for steady state. The script `runner2` shows the job flow for transient calculations. The input used for the transient has an end time of 100 seconds.

The end time for the sample problem is 3500 seconds (see script `runner-3500` and input deck `trn-3500.in`), which takes several hours to execute and is left for the analyst to run overnight.

```
-rw-r-----   1 t3923      eng         137608 Jan 21 16:03 new-base
-rw-r-----   1 t3923      eng          43971 Dec 15 10:33 rdx-eoc
-rwxr-x---   1 t3923      eng             92 Jan 27 10:51 rdx2.job
-rw-r-----   1 t3923      eng          24384 Feb  2 14:33 rodex2d
-rwxr-x---   1 t3923      eng            387 Jan 27 10:53 runner-3500
-rwxr-x---   1 t3923      eng            380 Jan 27 10:52 runner2
-rwxr-x---   1 t3923      eng            361 Jan 27 10:52 runssi
-rw-r-----   1 t3923      eng        1070710 Feb  2 14:33 test-rdx2.out
-rw-r-----   1 t3923      eng        2023398 Feb  2 15:25 test-ssi.out
-rw-r-----   1 t3923      eng         901293 Feb  2 15:46 test-tran.out
-rw-r-----   1 t3923      eng           1140 Jan 21 16:22 trn-3500.in
-rw-r-----   1 t3923      eng           1204 Jan 21 15:34 trn.in
```

EXECUTABLES

The subdirectory `executables`, contains executables made using HPUX-9.0 and are included as back-up in the event that the load steps fail. These files should execute on hp-workstations using HPUX-9.0 and above operating systems. They have been tested on a HPUX-11.0 machine. The files are:

```
-rw-r-----   1 t3923      eng         161872 Feb  2 14:34 STH2XT
-rwxr-x---   1 t3923      eng         107592 Feb  1 09:30 r2dmx
-rwxr-x---   1 t3923      eng         352304 Feb  2 14:34 rdx2lse
-rwxr-x---   1 t3923      eng        1461992 Feb  2 14:34 relap5
-rwxr-x---   1 t3923      eng          28672 Feb  2 14:34 select
-rwxr-x---   1 t3923      eng         385024 Feb  2 14:34 sth2xg
-rwxr-x---   1 t3923      eng        4594340 Feb  1 09:30 xmgr
```

SUB-DIRECTORY DESCRIPTIONS

R2DMX	r2dmx source code
XMGR	xmgr source code
bin	executable files from the build steps
build_sr5	S-RELAP5 build script
build_xmgr	xmgr build script
clean-up	script to remove '*.o' files
envr	source code for the environmental library used by S-RELAP5
eumod1	source code for the eumod1 library used by S-RELAP5

executables	executable files made on a hp-workstation using HPUX-9.0
icecon	source code for the icecon library used by S-RELAP5
libcalls	source code for the SPC library used by S-RELAP5, RODEX2, and environmental library
relap5	source code for S-RELAP5
rodex2	source code for RODEX2
sample	sample problems to test the S-RELAP5 build step
steam	source for water property routines
utils	source for selectx

```
drwxr-x---   2 t3923      eng            1024 Feb  1 09:36 R2DMX
drwxr-x---   5 t3923      eng            6144 Feb  1 09:36 XMGR
drwxr-x---   2 t3923      eng            1024 Feb  2 16:38 bin
-rwxr-x---   1 t3923      eng             651 Feb  2 13:50 build_sr5
-rwxr-x---   1 t3923      eng             292 Jan 27 11:34 build_xmgr
-rwxr-x---   1 t3923      eng             394 Feb  2 16:39 clean-up
drwxr-x---   4 t3923      eng              96 Jan 31 16:30 envr
drwxr-x---   4 t3923      eng              96 Jan 31 16:30 eumod1
drwxr-x---   2 t3923      eng            1024 Feb  1 09:30 executables
drwxr-x---   4 t3923      eng              96 Jan 31 16:30 icecon
drwxr-x---   2 t3923      eng            1024 Feb  2 16:38 lib
drwxr-x---   2 t3923      eng            2048 Feb  2 16:38 libcalls
drwxr-x---   4 t3923      eng              96 Jan 31 16:30 relap5
drwxr-x---   2 t3923      eng            2048 Feb  2 16:38 rodex2
drwxr-x---   2 t3923      eng            1024 Feb  2 14:31 sample
drwxr-x---   4 t3923      eng            1024 Feb  2 14:21 steam
drwxr-x---   2 t3923      eng              96 Feb  2 14:07 utils
```

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for Siemens Power Corporation ("SPC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by SPC to determine whether certain SPC information is proprietary. I am familiar with the policies established by SPC to ensure the proper application of these criteria.

3. I am familiar with the SPC information transmitted by letter NRC:00:010 and referred to herein as "Document." Information contained in this Document has been classified by SPC as proprietary in accordance with the policies established by SPC for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by SPC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in the Document be withheld from public disclosure.

6. The following criteria are customarily applied by SPC to determine whether information should be classified as proprietary:

(a) The information reveals details of SPC's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for SPC.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for SPC in product optimization or marketability.

(e) The information is vital to a competitive advantage held by SPC, would be helpful to competitors to SPC, and would likely cause substantial harm to the competitive position of SPC.

7. In accordance with SPC's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside SPC only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. SPC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 3rd day of February, 2000.

AMY R. NIXON
NOTARY PUBLIC
STATE OF WASHINGTON
COMMISSION EXPIRES
DECEMBER 6, 2003

Amy R. Nixon
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 12/06/03

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 11, 2000

Mr. James F. Mallay
Director, Regulatory Nuclear Affairs
Siemens Power Corporation
2101 Horn Rapids Road
Richland, WA 99352

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - SIEMENS POWER CORPORATION TOPICAL REPORTS, EMF-2310(P), REVISION 0, "SRP CHAPTER 15 NON-LOCA METHODOLOGY FOR PRESSURIZED WATER REACTORS," (TAC NO. MA7192) AND EMF-2328(P), REVISION 0, "PWR SMALL BREAK LOCA EVALUATION MODEL, S-RELAP5 BASED" (TAC NO. MA8022)

Dear Mr. Mallay:

By letter dated November 22, 1999, Siemens Power Corporation (SPC) submitted Topical Report EMF-2310(P), Revision 0, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," and by letter dated January 10, 2000, Topical Report EMF-2328(P), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based" for staff review. The primary document describing the S-RELAP5 code is EMF-2100(P), Rev. 2, "S-RELAP5 Models and Correlations Code Manual," dated January 2000. Review of the two topical reports involves reviewing the use of the S-RELAP5 code for application to the small break loss-of-coolant (LOCA) and non-LOCA transients.

In order for the staff to complete its review of the topical reports, the enclosed information pertaining to the S-RELAP5 computer code is required.

A mutually agreeable target date of within 30 days of the date of this letter for your response has been established. If circumstances result in the need to revise the target date, please call me at the earliest opportunity at 301-415-1480.

Sincerely,

N. Kalyanam, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 702

Enclosure: Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

SIEMENS POWER CORPORATION

TOPICAL REPORTS EMF-2310(P), REVISION 0, "SRP CHAPTER 15 NON-LOCA

METHODOLOGY FOR PRESSURIZED WATER REACTORS," AND

EMF-2328(P), REVISION 0, "PWR SMALL BREAK LOCA EVALUATION MODEL,

S-RELAP5 BASED"

PROJECT NO. 702

The following questions are in regard to Topical Report EMF-2100(P), Revision 2.

Comments/Editorials:

G.1 In several places including the first sentence on page 2-1, you stated that the S-RELAP5 code solves two-phase, two-fluid six equations plus one continuity equation for noncondensable gas and a boron tracking equation. S-RELAP5 actually includes a two-fluid model for a two-phase system. The sentence in your report implies that the code models two phases for two different fluids. This is not accurate.

Chapter 1: Introduction

1.1 On page 1-2 you stated that you have applied 2-D modeling to the downcomer, core, and upper plenum. Please explain why 2-D modeling of the lower plenum and lower head has not been applied.

1.2 On page 1-2 you stated that the modification made to the energy equations are more appropriate for analyses involving a containment volume. In Information Notice 92-02 the staff stated that codes in the RELAP5 series are not intended to be used as containment analysis codes. Containment analysis specific codes exist for that purpose. The primary purpose of the RELAP5 codes is analysis of the response of the nuclear steam supply system (NSSS) to accident and transient conditions. Please clarify the intent of your statement in light of the statement in the Information Notice.

1.3 On page 1-6 you stated that the steady-state option does not perform convergence tests and that users are required to set up the conditions for determining whether a steady-state is obtained. Please discuss the guidance provided to the users to aid them in doing this and identify where such guidance has been included.

Chapter 2: Fluid Field Equations and Numerical Solutions

2.1 Please provide a description of the major differences between S-RELAP5 and RELAP5/MOD2 pertaining to the Semi-Implicit Numerical Solution Scheme.

2.2 In the second paragraph on page 2-29, Section 2.6, it is stated that RELAP5/MOD2 was extended to include a two-dimensional flow solution scheme in S-RELAP5. Was this new scheme benchmarked or validated to ensure correct implementation and correctness of the scheme? Please discuss the benchmarking.

2.3 On page 2-54, the subject of time-step control is discussed. How does the time-step calculation in S-RELAP5 differ from that used in RELAP5/MOD2? In particular, discuss any differences in the way the error is measured within the two methods.

2.4 The energy equations presented do not include energy dissipations due to wall friction and pump effects. Please derive your energy equations to show how these terms are eliminated and/or justify the exclusion of these terms. Please justify your simplifying assumption included as Equation 2.13 in your report.

2.5 The energy equations presented assume that the enthalpy in the wall vaporization term ($\Gamma_w h$) is the saturation enthalpy. Please justify this assumption.

2.6 On page 2-6 you stated that under most circumstances, assessment calculations indicate that there are essentially no differences in the results of key loss of coolant accident (LOCA) parameters between the RELAP5/MOD2 energy equations and the energy equations provided in S-RELAP5. Please provide a discussion of the assessment calculations performed including a discussion of the key LOCA parameters that were assessed. In addition, please provide a discussion of the circumstances where differences were identified and justify your methodology in light of those differences. Also, provide similar discussions related to the other transients that you are proposing to analyze with the code.

2.7 Please provide a discussion of the heat transfer at the noncondensable gas-liquid interface and the effect of this on the energy equations. Please explain how this is modeled in your proposed methodology.

2.8 Under Section 2.4, State Relationships, you assume that the interface temperature is the saturation temperature. Please justify this assumption.

2.9 Please derive Equation 2.42 and justify your assumption that the extrapolated κ is just the saturation value for both the superheated liquid and the subcooled steam.

2.10 Your statement that substitution of Equations 2.45 and 2.47 into Equation 2.48 yields Equation 2.50 does not appear correct. Please show how Equation 2.50 was obtained. Note that this error continues in later derivations.

2.11 Regarding Equations 2.101 and 2.102, why is the velocity at j+1 evaluated at time n+1 while being multiplied by the density and void fraction at j+1 from time n? Note that the velocity at j is evaluated at time n and multiplied by the density and void fraction at j from time n. Also, compare with Equations 2.103 through 2.105, wherein the velocity at j+1 is evaluated at time n+1 but multiplied by the density and void fraction at j+1 from time n. But velocity at j is evaluated at time n+1 and multiplied by density and void fraction at j from time n.

2.12 How are areas for the momentum flux terms in the 2-D components calculated? How is this conveyed to the user?

2.13 How are the variables (a_g) and (a_f) in Equation 2.116 defined?

2.14 Given the fact the rθ is treated as r when using the (z,θ) form of the 2-dimensional momentum equations, as opposed to the (z,r) form, how is the "r" defined?

2.15 Has the effect of violations of the material Courant limit been evaluated? What is the recommended value for $\Delta t_c(i)$ in Equation 2.212?

Chapter 3: Hydrodynamic Constitutive Models

Editorial:

3.1 Page 3-6, first paragraph states that Wallis asserted that $j_{g^*} \approx 0.9$. The star appears incorrectly placed. Consistent with the remainder of the text it appears that the star should be a superscript to j instead of g.

Technical:

3.2 On page 3-1, end of the second paragraph, it is stated that code-data comparisons for the key parameters are to be used for assessing the applicability of the interphase constitutive models. Earlier in the same paragraph it was stated that the key parameters are phasic temperatures, phasic velocities, phasic densities, mass flow rates, and void fractions. Please explain how the key parameters were identified and provide the assessments that were performed to confirm the applicability of the interphase constitutive models.

3.3 [This question is related to large break LOCA (LBLOCA) only and may be responded to at the time of the BE LBLOCA submittal].

On Page 3-1, last paragraph, it is stated that the Electric Power Research Institute (EPRI) drift-flux correlations used in RELAP5/MOD3 are tuned mostly to the steady-state data with regular flow profiles and that there is little evidence that these fix-profile correlations produce good results in simulating LBLOCA transients which are highly irregular and chaotic in nature. It is also stated that the EPRI correlations do not cover the entire range of two-phase flow conditions. Based on this information, it was stated that S-RELAP5 did not adopt the same approach as used in RELAP5/MOD3 but that assessment examples are presented to show that the S-RELAP5 two-fluid formulation produces code-data comparisons that are as good as those obtained by RELAP5/MOD3 for steady-state and nearly steady-state cases. Since the concern stated with the EPRI drift-flux correlations was with the modeling of the LBLOCA transients which are highly irregular and chaotic in nature, please provide the assessments that were performed to ensure that the correlations used in S-RELAP5 are adequate for highly irregular and chaotic transient cases.

3.4 In Equation 3.7, you limited α_L to a minimum value of 0.1 and used $(D^*/19)^8$. Which experiments form the basis for choosing these values? Please justify the use of these values.

3.5 Please describe the tests used in the assessment and provide the assessments performed to validate the use of Equation 3.11 and the limits provided in the text that follows the equation on pages 3-6 and 3-7 in relation to the $\alpha_{S\text{-}A}$ criteria.

3.6 On page 3-7, end of the first paragraph, it is stated that introduction of transition regions may reduce the chances of occurrence and magnitude of discontinuities in interphase interaction terms, but it cannot completely eliminate the discontinuities. Please describe known discontinuities that still remain and how these are dealt with in the coding of S-RELAP5.

3.7 Please describe the information used to confirm the validity of the interpolation in Equation 3.15.

3.8 Under the vertical stratification section starting on page 3-8, there appear to be no flow/velocity criteria established for when vertical stratification may occur. Please explain how vertical stratification is detected.

3.9 Please describe how the mixture level model described under the vertical stratification section was validated.

3.10 Please describe the assessment performed to justify the method used for the transition region between the stratified and non-stratified flow (i.e., Equation 3.26 and associated restrictions and criteria).

3.11 Justify the choice of 0.9 for j_g^* for the boundary between slug and annular mist flow (Equation 3.28) in light of the wide range of 0.25 to 1.0 suggested by Wallis. What are the sensitivities of the results of the analyses of interest to the value of j_g^* and why is 0.9 appropriate in light of these sensitivities? What is the range of hydraulic diameters that this criterion is valid for? Please describe the assessment performed to cover the sensitivity to hydraulic diameter. Provide a comparison to applicable experimental data.

3.12 Please describe how the effect of condensation at the ECCS injection point is handled in S-RELAP5.

3.13 Please show how Equation 3.23 is derived from the material in the reference. Also, it appears in Equation 3.23 that the α_g is a subscript to β. Please confirm or correct this.

3.14 On page 3-48, it is stated that various assessment calculations indicate that Equations 3.98 and 3.99 function well. Please identify and discuss the tests that were used in the assessment calculations and the results of the assessment calculations.

3.15 Section 3.4.8 discusses the equilibrium option that exists in S-RELAP5. Please provide a table showing when (i.e., in what transient analyses) this option would be allowed and when it would not be allowed. Also, please provide a reference to the section in the user's manual that directs the user to follow these restrictions. If allowed in any of the licensing analyses, please justify the values selected.

3.16 Section 3.4.9 discusses the effect of noncondensables on condensation rate. Please justify your use of Equations 3.169 and 3.165 in S-RELAP5 to handle the reduction of condensation rate in the presence of noncondensables. Please provide a description and results of assessment calculations that justify the use of these equations.

3.17 For time smoothing, it is stated on page 3-68 that the scheme implemented in S-RELAP5 is empirical and that various assessment calculations indicate that it works satisfactorily. Please describe the assessment calculations performed for confirming the time smoothing scheme. In addition, show how the assessment calculations provide a test for the scheme.

3.18 In Section 3.4.10, in relation to mass error, it is stated that S-RELAP5 implements a strategy which forces only condensation to take place when the amount of liquid in a volume is small and subcooled and the vapor is superheated. In addition, this strategy forces only evaporation to take place when the amount of vapor in a volume is small and subcooled and the liquid is superheated. It is stated that these limits have no significant effects on physical results as one would expect from such a diminishing amount of liquid or vapor and that these limits reduce mass error substantially. Please justify your strategy for dealing with the mass error. In your justification, please discuss any assessments that were performed, the tests used in the assessments, and the results.

3.19 In Section 3.4.10, in relation to subcooled nucleate boiling, it is stated that S-RELAP5 implements a strategy which lowers the interphase heat transfer coefficients in order to eliminate situations where the total mass transfer rate, Γ_g, becomes negative. Please justify your strategy for dealing with this situation. In your justification, please discuss any assessments that were performed, the tests used in the assessments, and the results. In addition, the last paragraph on page 3-70 states that there is no guarantee that the final solution at the end of each time step meets all the conditions or limits described in the section. Please explain what is meant by this statement and explain and justify what is done in S-RELAP5 when the conditions or limits are not met.

3.20 Please provide a list of the figures of merit and important phenomena in relation to each of the transients and accidents to be analyzed with S-RELAP5. Please also describe how these figures of merit and important phenomena were designated as important for the relevant analyses.

3.21 In Sections 3.4.1 through 3.4.7, heat transfer correlations, limits on these correlations, and transition equations are presented for different flow regimes. However, no justifications are provided. Please provide justifications for the material presented in these sections and provide a discussion of assessments performed to confirm the adequacy of correlations used in S-RELAP5.

3.22 Page 3-11, last paragraph, it is stated that "...some calculations with RELAP5/MOD2 indicated that the range of stratified flow is too small. Kukita et al suggested that the vapor velocity on the left side of Equation 3.22 be replaced by the relative velocity (v_g-v_f).

This approach along with an additional constraint to exclude high mass flux conditions was implemented in the previous S-RELAP5 code versions. Recent experience with small break test cases and plant calculations indicated that the new approach might increase code variability. Therefore, the approach of replacing the vapor velocity with relative velocity is abandoned."

Since the approach was abandoned, what was done to address the concern that the range of stratified flow was too small and how was that justified? Please provide comparisons of your approach to data to justify the adequacy of your approach.

3.23 [This question is related to LBLOCA only and may be responded to at the time of the BE LBLOCA submittal].

For dry-wall flow regimes, please justify your use of 0.1 for the α_{IA-IS} criterion in light of the information provided in the text preceeding Equation 3.13 that indicates that the transformation of the three wet-wall flow regimes into inverted annular, inverted slug, and mist flow regimes should be used.

Chapter 4: Heat Transfer Models

4.1 In reviewing Section 4, Heat Transfer Models, it is apparent that this section is totally different to any comparable heat transfer section in RELAP5/MOD2. Contributions from various known sources constitute the basis for this heat transfer model. Please provide qualitative (and quantitative) justification for the formulation of this particular heat transfer model (i.e., assumptions, mass flow rates, pressure, enthalpy, etc.).

4.2 On page 4-2 of the S-RELAP-5 Models and Correlations Code Manual, the last sentence of the last paragraph discusses the issue of reflood being turned off and on. Who decides when or where the option is turned on or off at the appropriate time?

4.3 Please provide an explanation of the difference between the data and the calculational results in Figure 4.3

4.4 How does RELAP-5/MOD2 or MOD3 compare to the same data as that presented in RAI 4.3 above? A comparison of S-RELAP5 and RELAP5/MOD2 against the data and on the same page would help.

Chapter 11: Point Kinetics Model

11.1 On page 11-16, the last equation has a term missing. The term " $-V_{01}$ " is missing. Compare with Equation 7.6-21 in NUREG/CR-5535, V1.

The following question is in regard to Topical Report EMF-2328(P), Revision 0.

SB.1 Please justify use of 0 percent fuel clad preoxidation in the SBLOCA analysis.

SIEMENS

January 26, 2001
NRC:01:007

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Request for Additional Information – Siemens Power Corporation Topical Reports, EMF-2310(P) Revision 0, (TAC No. MA7192) and EMF-2328(P) Revision 0, (TAC No. MA8022)

Ref.: References are listed on page 2.

In Reference 1, the NRC requested additional information to facilitate the completion of its review of the SPC topical report on the PWR Non-LOCA methodology (Reference 2) and the topical report on the PWR SBLOCA evaluation model (Reference 3). Responses to this request are provided in two attachments: one proprietary and one nonproprietary.

In Reference 4, SPC provided supporting information for the review of these two topical reports. Due to NRC comments regarding typographical errors in the supporting information, SPC is providing revised copies (References 5 and 6). (NOTE: Eight copies of these reports have been provided directly to N. Kalyanam.)

Siemens Power Corporation considers some of the information contained in the attachments and enclosures to this letter to be proprietary. The affidavits provided with the original submittals of the reference reports (References 2, 3, and 4) satisfy the requirements of 10 CFR 2.790(b) to support the withholding of this information from public disclosure.

Very truly yours,

James F. Mallay, Director
Regulatory Affairs

/arn

Attachments - 2/Enclosures – 2

cc: N. Kalyanam (w/Att. & Enc.)
Project No. 702 (w/Att.)

Siemens Power Corporation

2101 Horn Rapids Road
Richland, WA 99352

Tel: (509) 375-8100
Fax: (509) 375-8402

Ref.: 1. Letter, N. Kalyanam (NRC) to J. F. Mallay (SPC), "Request for Additional Information – Siemens Power Corporation Topical Reports, EMF-2310(P), Revision 0, 'SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,' (TAC NO. MA7192) and EMF-2328(P), Revision 0, 'PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,' (TAC NO. MA8022)," December 11, 2000.

Ref.: 2. Letter, J. F. Mallay (SPC) to Document Control Desk (NRC), "Request for Review of EMF-2310(P) Revision 0, *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*," NRC:99:048, November 22, 1999.

Ref.: 3. Letter, J. F. Mallay (SPC) to Document Control Desk (NRC), "Request for Review of EMF-2328(P) Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*," NRC:00:002, January 10, 2000.

Ref.: 4. Letter, J. F. Mallay (SPC) to Document Control Desk (NRC), "NRC Review of Siemens Power Corporation Topical Reports," NRC:00:009, February 3, 2000.

Ref.: 5. EMF-2100(P) Revision 3, *S-RELAP5 Models and Correlations Code Manual*, Siemens Power Corporation, January 2001.

Ref.: 6. EMF-2101(P) Revision 2, *S-RELAP5 Programmers Guide*, Siemens Power Corporation, January 2001.

S-RELAP5 Request for Additional Information (RAI)

The following are in regard to EMF-2100(P) Rev. 2:

Comments/Editorials:

G.1 *In several places including the first sentence on Page 2-1, you stated that the S-RELAP5 code solves two-phase, two-fluid six equations plus one continuity equation for noncondensable gas and a boron tracking equation. S-RELAP5 actually includes a two-fluid model for a two-phase system. The sentence in your report implies that the code models two phases for two different fluids. This is not accurate.*

The S-RELAP code solves two-fluid six equations plus one continuity equation of noncondensable gas and a boron tracking equation for flow of a two-phase steam-water mixture which can contain a noncondensable in the vapor phase and a soluble in the liquid phase.

Chapter 1: Introduction

1.1 *On Page 1-2 you stated that you have applied 2-D modeling to the downcomer, core, and upper plenum. Please explain why 2-D modeling of the lower plenum and lower head has not been applied.*

The S-RELAP5 2-D component is flexible and can be applied to any selected component through input, and the 2-D modeling has been successfully applied to various RCS components, including the lower head and plenum. Use of the 2-D model adds considerably to the complexity of the system input and running time of the analysis model. Therefore, SPC methodologies will invoke the use of the 2-D model only for regions in applications where significant multi-dimensional effects are expected. Thus, the use of the S-RELAP5 2-D model will be different depending on the licensing application.

For SBLOCA applications, 2-D modeling is applied in the core and downcomer regions. Significant multi-dimensional effects which would require 2-D modeling in the lower plenum and lower head are not expected for SBLOCA. For non-LOCA transients, the 2-D capabilities are not required. The methodology topical reports for each S-RELAP5 application describe the use of the 2-D modeling for that specific application.

1.2 *On Page 1-2 you stated that the modification made to the energy equations are more appropriate for analyses involving a containment volume. In Information Notice 92-02 the staff stated that codes in the RELAP5 series are not intended to be used as containment analysis codes. Containment analysis specific codes exist for that purpose. The primary purpose of the RELAP5 codes is analysis of the response of the NSSS to accident and transient conditions. Please clarify the intent of your statement in light of the statement in the Information Notice.*

During a PWR LBLOCA, a coupling exists between reflood heat transfer and containment back pressure. Calculation of this coupling requires that accurate mass and energy release data be provided to the containment code calculation which then feeds back the appropriate back pressure for the reactor system calculation. To correct the problem associated with the Information Notice, changes were made to the S-RELAP5 code to provide energy conservation for all conditions. In addition, changes were made to incorporate the ICECON containment code into S-RELAP5, and to interface the containment code calculation so that the containment calculation is performed as part of S-RELAP5 in parallel with the NSSS transient calculation.

The energy equation changes were made to directly address the problem identified by Northeast Utilities which resulted in Information Notice 92-02. It was found that the base RELAP5 code did not conserve energy when critical flow was calculated with a large pressure drop between volumes such as from the NSSS to the containment during a LBLOCA event. This means that the mass and energy release to the containment calculated by the then existing versions of the RELAP5 code could be erroneous and results from these code versions should not be used as the source terms for containment analysis performed with either RELAP5 or a containment analysis code.

The energy equations in the base RELAP5 code are formulated in terms of thermal energy. With this formulation, P-V work terms are not calculated accurately. For the large pressure drop conditions, this results in an energy conservation error. The S-RELAP5 energy equations are formulated in terms of total energy which conserves energy over all pressure drop conditions.

In the coupled NSSS and containment calculation, the mass and energy release to a time dependent volume is calculated by S-RELAP5 for one time step. This information is then passed to the ICECON (CONTEMPT) portion of S-RELAP5 where the updated containment back pressure is calculated. Back pressure is then passed back to the time dependent volume and applied as a boundary condition on the NSSS calculation for the next time step. Since energy is conserved using the S-RELAP5 code, and this code now contains the ICECON containment module, the containment pressure can be determined using S-RELAP5.

It should be noted that the energy equation changes in the S-RELAP5 documentation have little effect on S-RELAP5 calculations for SBLOCA or non-LOCA transients, and that containment pressure is not calculated for these methodologies. The changes are necessary and important for the planned submittal of the realistic LOCA methodology, and the applications described apply only to that methodology. This change also would be important if mass and energy release are calculated for use in a containment analysis code such as GOTHIC.

1.3 *On Page 1-6 you stated that the steady-state option does not perform convergence tests and that users are required to set up the conditions for determining whether a steady-state is obtained. Please discuss the guidance provided to the users to aid them in doing this and identify where such guidance has been included.*

SPC develops user guidelines for each event analysis and a guideline for input deck generation. Those guidelines include specific requirements for developing steady- state controllers, as well as guidelines for establishing criteria for acceptable steady-state conditions. Currently, those guidelines are specific to using ANF-RELAP for the thermal hydraulic portion of the transient.

Upon acceptance of the proposed methodologies, the guidelines will be updated to reflect the differences between the use of ANF-RELAP and S-RELAP5. However, both the SBLOCA and non-LOCA analyses will use the current ANF-RELAP guidelines for establishing steady-state acceptance criteria.

The criteria for establishing steady-state calculation acceptance for any of the events are as follows:

The calculated results from the null transient using the steady-state option are examined closely to ensure that a true steady-state condition has been established. This is achieved by examining specific parameters (listed below) and comparing them against the desired steady-state plant conditions. Reasonable stability and comparison of these parameters with known steady-state values would indicate an acceptable steady-state condition has been achieved. Current guidelines recommend that plots of the key parameters be included in the calculation notebook, so the attainment of a steady-state can be visually verified.

The following are parameters are recommended for inspection to assure steady conditions have been reached:

- Reactor power
- Primary pressure
- Loop pressure drop
- Loop flow rate
- Core bypass and leakage path flow rates
- Vessel upper head temperature
- Cold leg temperature
- Hot leg temperature
- SG secondary pressure
- SG secondary mass inventory
- SG secondary void profile
- SG feedwater and steam flow rates
- SG recirculation ratio
- Mass flow rates in the SG boiler region
- Pressurizer collapsed liquid level
- Core collapsed liquid level
- Hot channel wall temperatures
- Core mass flow

Chapter 2: Fluid Field Equations and Numerical Solutions

2.1 *Please provide a description of the major differences between S-RELAP5 and RELAP5/MOD2 pertaining to the Semi-Implicit Numerical Solution Scheme.*

[

] The detailed algebraic manipulation is shown in Equations (2.131) to (2.195). The Gaussian solver without pivoting may lose significant accuracy under some circumstances (e.g., when the matrix is nearly singular); therefore, the RELAP5/MOD2 method is not used. Another difference is the more implicit treatment of the pump junction velocities, which is described on Pages 2-40 to 2-41 [Equations (2.118 to (2.124)].

2.2 *In the second paragraph on Page 2-29, Section 2.6, it is stated that RELAP5/MOD2 was extended to include a two-dimensional flow solution scheme in S-RELAP5. Was this new scheme bench-marked or validated to ensure correct implementation and correctness of the scheme? Please discuss the bench-marking.*

The S-RELAP5 two-dimensional flow scheme was verified and validated. Two types of benchmark cases were used to verify/validate the 2-D model: cases with known solutions and comparisons to multi-dimensional flow data. Calculations of cases with known solutions, such as 2-D symmetrical fill problems, validate correct implementation of the 2-D model. Comparisons with measured data show the validity of the model. A symmetric fill problem was set up for the (z,θ)-type 2-D component to check if correct velocities and flow symmetry are calculated in the 2-D model. The 2-D nodalization scheme is similar to that used for modeling the reactor vessel downcomer. The calculation shows that the liquid advances with the same velocity as the injection (time-dependent junction) velocity in all vertical directions and flow symmetry is maintained throughout the entire period, including the period after the 2-D component completely fills. This verifies that the 2-D momentum flux terms are correctly treated. A similar exercise was performed on the (z,x)-type 2-D component, producing correct results. Since the plant steady-state conditions such as flow rates, velocities, and flow patterns are known, the plant steady-state calculations can also be used to check the correctness of the 2-D model implementation.

The purpose of a comparison with test data using the 2-D component is to validate its applicability for modeling multi-dimensional flow problems. Two-dimensional flow test comparisons performed specifically to validate the S-RELAP5 2-D modeling are given in section 5.1 of the SBLOCA topical EMF-2328(P). Section 5.5.2 of EMF-2100(P) also discusses results from a UPTF simulation where the (z,θ)-type 2-D component was used to model a downcomer. The calculated results shown in Figure 5.17 on Page 5-60 of EMF-2100(P) demonstrate that a proper velocity profile was obtained in that simulation.

2.3 *On Page 2-54, the subject of time-step control is discussed. How does the time-step calculation in S-RELAP5 differ from that used in RELAP5/MOD2? In particular, discuss any differences in the way the error is measured within the two methods.*

In S-RELAP5 the time step control is performed through four criteria: (1) material Courant limit {Equation (2.211)}, (2) consistency check on the mass solution {Equation (2.213)}, (3) consistency check on the energy solution {Equation (2.214)}, and (4) Failure of equation of state. For the Courant limit, RELAP5/MOD2 implements a partial violation of the Courant limit. The partial violation scheme is present in the S-RELAP5 code, but is not used, i.e., no partial violation of Courant limit is allowed. RELAP5/MOD2 does not have item (3) and adds a measure of overall system mass differences in item (2). The criteria for the mass consistency check are 1×10^{-3} (repeat) and 1×10^{-4} (double) [see description below Equation (2.213)] in the S-RELAP5 Theory Manual, and are 2×10^{-3} and 2×10^{-4} in RELAP5/MOD2. Both S-RELAP5 and RELAP5/MOD2 check the mass conservation by computing the accumulated mass generation (or destruction) in the system, which is shown on the major edit as mass error. This system mass error is not used in time-step control in S-RELAP5 and the RELAP5 codes.

2.4 *The energy equations presented do not include energy dissipations due to wall friction and pump effects. Please derive your energy equations to show how these terms are eliminated and/or justify the exclusion of these terms. Please justify your simplifying assumption included as Equation 2.13 in your report.*

The energy equations in S-RELAP5 are expressed in the total energy form. The terms in Equations (2.4) and (2.5) plus Equation (2.13) can be identified from the following general statement of the law of conservation of energy for the fluid in a control volume:

$$\left\{\begin{matrix}\text{rate of}\\ \text{accumulation}\\ \text{of internal}\\ \text{and kinetic}\\ \text{energy}\end{matrix}\right\} = \left\{\begin{matrix}\text{rate of}\\ \text{internal and}\\ \text{kinetic energy}\\ \text{in}\\ \text{by convection}\end{matrix}\right\} - \left\{\begin{matrix}\text{rate of}\\ \text{internal and}\\ \text{kinetic energy}\\ \text{out}\\ \text{by convection}\end{matrix}\right\}$$

$$+\left\{\begin{matrix}\text{net rate of}\\ \text{heat addition}\\ \text{by conduction}\end{matrix}\right\} - \left\{\begin{matrix}\text{net rate of work}\\ \text{done by system}\\ \text{on surroundings}\end{matrix}\right\}$$

(see Page 311 of Transport Phenomena by R. B. Bird, W. E. Stewart, and E. N. Lightfoot, 1960.)

[

]

2.5 *The energy equations presented assume that the enthalpy in the wall vaporization term ('wh) is the saturation enthalpy. Please justify this assumption.*

The product $\left(\Gamma_w\, h_k^s\right)$ represents the energy transfer for phase *k* (either addition or subtraction) associated with the mass transfer due to the "wall vapor generation" term. In subcooled boiling, Γ_w is positive and the energy transferred to the vapor within the control volume is $\left(\Gamma_w\, h_g^s\right)$ as it should be. The implication is that the generated vapor appears at the saturation temperature corresponding to the local pressure. The energy removed from the liquid phase within the control volume is then $\left(\Gamma_w\, h_f^s\right)$. As the liquid phase is subcooled, there appears to be an energy imbalance with the magnitude $\left[\Gamma_w\left(h_f^s - h_l^s\right)\right]$ corresponding to the liquid sensible heat that must be added to bring the subcooled liquid up to the saturation temperature. This energy imbalance does not exist because this sensible heat requirement has already been accounted for through the determination of the fraction of the wall heat flux that causes vapor generation (see Equation (4.27) of Section 4.3.2) as discussed below.

S-RELAP5 uses the Lahey subcooled boiling model. The wall heat flux is first divided into two parts: one for sensible heat transfer and one that is "available" for vapor generation (denoted as q''_{wv} in the manual). This heat flux that is available for vapor generation is then further partitioned into a fraction that actually causes vapor generation $\left(q''_{evap}\right)$ and that corresponding to the sensible heat transfer needed to bring the bulk liquid up to the saturation temperature based on an equal volume exchange $\left(q''_{pump}\right)$. Thus, the sensible heat transfer due to this "pumping" term accounts for the energy transfer needed to bring the mass of subcooled liquid that is being evaporated up to the saturation temperature.

2.6 *On Page 2-6 you stated that under most circumstances, assessment calculations indicate that there are essentially no differences in the results of key loss of coolant accident (LOCA) parameters between the RELAP5/MOD2 energy equations and the energy equations provided in S-RELAP5. Please provide a discussion of the assessment calculations performed including a discussion of the key LOCA parameters that were assessed. In addition, please provide a discussion of the circumstances where differences were identifies and justify your methodology in light of those differences. Also, provide similar discussions related to the other transients that you are proposing to analyze with the code.*

The referenced assessment calculations were from undocumented developmental assessment results using LOFT L2-5, LOFT L2-6, CCTF Run 54, and FLECHT-SEASET Test 31504. Those calculations were made at the time of the energy equation modification. The stated differences were from comparing the previous results without the model changes with results having the model changes implemented. The parameters compared were cladding temperatures, steam temperatures, void fractions and pressures. The model had essentially no effect on the calculated result, as expected, since the system models did not include containment modeling (e.g., a large pressure drop across a choke plane).

The non-LOCA sample problems show comparisons between ANF-RELAP, which uses the same energy equations as RELAP5/MOD2, and S-RELAP5 calculated results. Those comparisons show that S-RELAP5 is essentially equivalent to ANF-RELAP for the modeling of non-LOCA transients Page 2-1, EMF-2310(P).

The SBLOCA methodology does not include containment modeling, therefore there are no expected differences in the results.

The Realistic LBLOCA model simulates the interaction between primary system and the containment response to blowdown. In this situation, the correct energy transfer to the containment model is necessary.

A demonstration calculation can be made to show the energy error when using the S-RELAP5 energy equations compared to the RELAP5/MOD2 energy equations. Consider a closed system where potential and kinetic energies are negligible and consisting of a small diameter pipe (1 m) at high pressure (150 bar) blowing down into a large diameter pipe (10 m) at low pressure (1 bar) through an orifice. Since there is no change in total internal energy in a perfect system, a comparison of initial internal energy to the transient internal energy during the blowdown should indicate net internal energy error.

A calculation of this type was made with both S-RELAP5 and ANF-RELAP (ANF-RELAP uses RELAP5/MOD2 energy equations). The results in Figure 1 show that energy is conserved to within 0.04% by S-RELAP5 while ANF-RELAP shows an error of approximately -2%. These results imply that there will be a much smaller energy error when transferring energy out of a system (i.e., coupled primary and containment calculation) using S-RELAP5 compared to a code using the RELAP5/MOD2 energy equations.

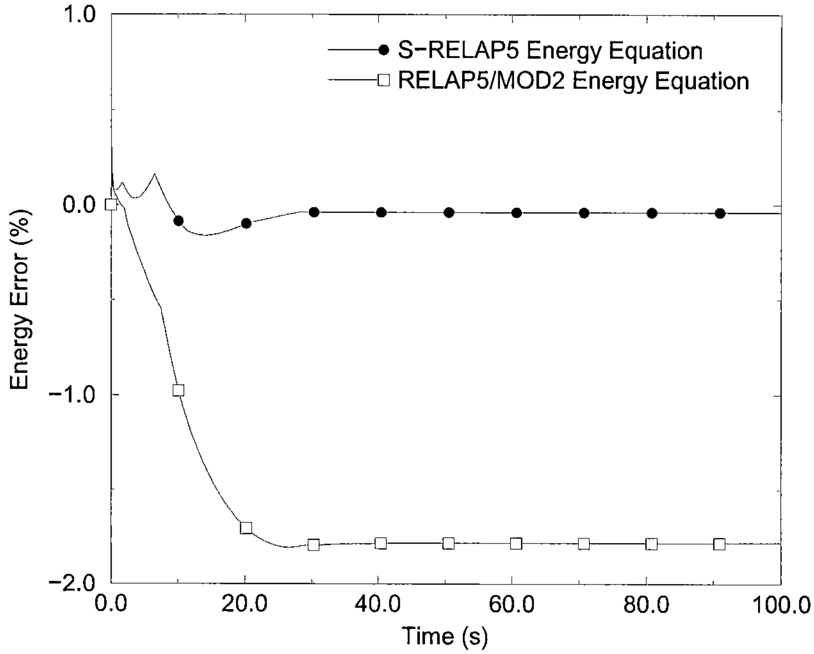

Figure 1. Comparison of energy error between S-RELAP5 and RELAP5/MOD2 energy equations

2.7 *Please provide a discussion of the heat transfer at the noncondensable gas-liquid interface and the effect of this on the energy equations. Please explain how this is modeled in your proposed methodology.*

The noncondensable interphase heat transfer is described in Section 3.4.9 {pp. 3-65 - 3-66 of EMF-2100(P)}. The effect of the model on the energy equations is handled through the interphase heat transfer terms in the energy equations (see Equations (2.5), (2.6) and the discussion on Pages 2-3 to 2-9). For SBLOCA and non-LOCA events, the noncondensable does not leave the accumulators; therefore, the noncondensable interphase heat transfer model has no effect. For LBLOCA, the entering of the noncondensable into the cold legs after the accumulators are emptied of water reduces the steam condensation rate, and thus, increases the cold leg pressures. This in turn causes a surge of ECC water into the core and provides additional cooling for a short period. It has a weak to moderate effect on the clad temperatures during the reflood phase of a LBLOCA.

2.8 *Under Section 2.4, State Relationships, you assume that the interface temperature is the saturation temperature. Please justify this assumption.*

The interface temperature is assumed to be at saturation for the modeling of the interphase heat transfer. The state relationship provides a computation of derivatives at the saturation temperature so that the interphase heat transfer terms can be linearized and treated implicitly.

It is a standard approach to use the saturation temperature as the reference temperature for formulating the interphase heat transfer model. The net effect of the interphase heat transfer model is to compute the amount of mass exchanged between the two phases. That is, the heat transfer from a phase to the saturation interface is just an intermediate step and the significant quantity is the heat transfer between the phases. At the equilibrium state, both phases are saturated. Setting the reference (interface) temperature to saturation provides a convenient measure of the deviation of a phase from equilibrium and simplifies the interphase heat transfer model.

2.9 *Please derive Equation 2.42 and justify your assumption that the extrapolated κ is just the saturation value for both the superheated liquid and the subcooled steam.*

Equation (2.42) has a typographical error. The corrected form, and the derivation is provided in the following text.

Equation (2.42) will be corrected in the next revision of the models and correlations document EMF-2100(P).

2.10 *Your statement that substitution of Equations 2.45 and 2.47 into Equation 2.48 yields Equation 2.50 does not appear correct. Please show how Equation 2.50 was obtained. Note that this error continues in later derivations.*

There is a typographical error in Equation (2.47): the "+" should be "=". That is, Equation (2.47) should be

$$V_g = X_n V_n = (1-X_n)V_v$$

The above equation is a direct consequence of the Gibbs-Dalton assumption that all gases (i.e., steam and noncondensables) occupy the same space. This equation will be corrected in the next revision to the models and correlations document, EMF-2100(P).

2.11 *Regarding Equations 2.101 and 2.102, why is the velocity at j+1 evaluated at time n+1 while being multiplied by the density and void fraction at j+1 from time n? Note that the velocity at j is evaluated at time n and multiplied by the density and void fraction at j from time n. Also, compare with Equations 2.103 through 2.105, wherein the velocity at j+1 is evaluated at time n+1 but multiplied by the density and void fraction at j+1 from time n. But velocity at j is evaluated at time n+1 and multiplied by density and void fraction at j from time n.*

There are typographical errors in Equations (2.101) and (2.102). The velocities at junction j should be superscripted with n+1. These changes will be made to the next revision of EMF-2100(P).

The time level difference between the velocity and mass, energy, or quality parameter is from the assumptions used in developing the semi-implicit numerical scheme. In the discussion in Section 2.6 of EMF-2100 (P), Rev 2., a reference is made to implicit terms formulated to be linear in the dependent variables at new time. The mass, energy, and noncondensable quality fluxes are those terms. Note that the momentum flux terms in Equations (2.109) and (2.110) consist of old time, or time level n, velocities. This allows the momentum equations to be reduced to Equation (2.116), the velocity at time level n+1. These new time velocities can be substituted into Equations (2.101) through (2.105) and yield expressions for mass, energy, and noncondensable quality in terms of ΔP. With appropriate substitutions, those equations can be combined into a single expression in terms of ΔP. The process is discussed in detail starting on Page 2-43.

2.12 *How are areas for the momentum flux terms in the 2-D components calculated? How is this conveyed to the user?*

The areas appear in the 2-D (and 1-D) momentum flux terms only indirectly through the volume average velocities, which are defined in Equations (2.98) - (2.100). The user usually provides the lengths and volumes of the 2-D nodes through input and the code calculates the areas by: area = volume/length. The user may also have to provide the junction areas according to the actual geometry. The S-RELAP5 Input Data Requirements section of the S-RELAP5 users' manual, EMF-CC-097(P), has a section for the 2-D component input prescription. Additional procedures will be discussed in the methodology guidelines.

2.13 *How are the variables (a_g) and (a_f) in Equation 2.116 defined?*

2.14 *Given the fact the $r\theta$ is treated as r when using the (z, θ) form of the 2-dimensional momentum equations, as opposed to the (z,r) form, how is the "r" defined?*

In the cylindrical (z,θ) 2-D system, r is measured from the origin and $r\Delta\theta$ is the length of the arc for the angle $\Delta\theta$. Since r is constant, $r\Delta\theta = \Delta(r\theta)$= arc length of an azimuthal sector. In the (z,r) system, Δr is the nodal length in the r-direction, which is the distance between two radial rings.

2.15 *Has the effect of violations of the material Courant limit been evaluated? What is the recommended value for $\Delta t_c(i)$ in Equation 2.212?*

Violation of the material Courant limit often leads to unstable solutions in the semi-implicit scheme. In the earlier years of RELAP5 development, partial violation of the Courant limit was considered to be acceptable if the solution was stable. However, its effect is difficult to quantify. Therefore, partial violation of the Courant limit is no longer used in S-RELAP5 applications. As stated in the paragraph below Equation (2.212), i=1 is used, i.e., no partial violation of the Courant limit.

Chapter 3: Hydrodynamic Constitutive Models

Editorial:

3.1 *Page 3-6, first paragraph states that Wallis asserted that $j_{g^*} \approx 0.9$. The star appears incorrectly placed. Consistent with the remainder of the text it appears that the star should be a superscript to j instead of g.*

Concur. The typo will be corrected.

Technical:

3.2 *On Page 3-1, end of the second paragraph, it is stated that code-data comparisons for the key parameters are to be used for assessing the applicability of the interphase constitutive models. Earlier in the same paragraph it was stated that the key parameters are phasic temperatures, phasic velocities, phasic densities, mass flow rates, and void fractions. Please explain how the key parameters were identified and provide the assessments that were performed to confirm the applicability of the interphase constitutive models.*

The key parameters were identified from analysis of the interphase constitutive models and their usage in the mass, energy, and momentum conservation equations. The constitutive models have an effect on mass fractions, temperatures, and slip. The parameters characterizing those phenomena are then void fraction, phasic densities, phasic temperatures, and phasic velocities. Flow rate is consequence of those preceding parameters.

Based on past experience and informal peer reviews, an informal PIRT was developed (see response to RAI 3.20). In the PIRT, processes and phenomena were ranked as having high importance, medium importance, and low importance during the five periods of a SBLOCA transient. Those processes which were ranked as having high importance established a basis for which of the S-RELAP5 models received rigorous assessment and the experimental data sets that were used for the assessment. Additionally, periods of two-phase flow could be identified in the PIRT. The experiments identified in the PIRT included the S-RELAP5 standard test set (STS), four SBLOCA specific tests, and a 2-Dimensional flow test. The STS consists of a wide range of experiments that are used to validate code performance and are exercised for each code version created for production use. The additional SBLOCA specific experiments are used primarily as phenomenological assessments in addition to model assessments. The 2-Dimensional test was used to validate the S-RELAP5 2-Dimentional capability.

The interphase constitutive models, interphase drag and interphase mass transfer, were assessed in the context of best possible performance under all conditions, as well as specific to SBLOCA transients.

In EMF-2100(P), results from several of the tests that make up the STS are presented. Listed below are those experiments with brief descriptions of the key parameters with references to their location in EMF-2100(P):

- <u>GE Level Swell</u> – The test assesses the level model, interphase friction, and interphase mass transfer. Key parameters are void fraction and liquid level. Discussion of results begins on Page 3-42 and void profiles are shown in Figures 3.5 and 3.6 on Page 3-43.

- <u>THTF Tests 3.09.10j, 3.09.10m, and 3.09.10dd</u> – The tests are steady boiling tests with level swell and are representative of the core boiling process during SBLOCA. They are used specifically for interphase friction and subcooled boiling assessments. The key parameter is void fraction. The discussion of results begins at the bottom of Page 3-43 and void profiles are shown in Figures 3.7 through 3.9 on Pages 3-44 through 3-45.

- <u>Bennett Heated Tube Tests 5358 & 5379</u> – Tests used to validate transient CHF. These are not applicable to SBLOCA. The key parameter is wall temperature. The discussion of results begins on Page 4-31 and wall temperature comparisons are shown in Figures 4.2 and 4.3 on Page 4-32 in EMF-2100(P).

- <u>FLECHT-SEASET Test 33056</u> – Test 33056 is used to assess the Sleicher-Rouse heat transfer coefficient to vapor. The key parameters are void fraction, mass inventory, steam temperatures, differential pressure, heat transfer coefficients, and wall temperatures. The discussion of results begins on Page 4-32 and wall temperature comparisons are shown in Figures 4.4 and 4.5 on Page 4-33.

- Marviken Tests 22 & 24 – The tests assess the S-RELAP5 critical flow model. Since Moody is used for Appendix K analysis, these tests are not applicable to SBLOCA. The key parameters are pressure, fluid temperature, and mass flow (break). The discussion of results begins in Section 5.1.3.2 and comparisons with data are shown in Figures 5.6 through 5.10 on Pages 5-28 through 5.32.

- UPTF Tests 6 & 7 – These tests were designed to quantify downcomer ECC bypass during the blowdown phase (accumulator injection phase) of a LBLOCA. The tests are also used to show 2-Dimensional effects in the downcomer inlet annulus region. The key parameters are differential pressure and mass in lower plenum. The discussion of results begin in Section 5.5.2 and gas velocity profiles are shown in Figure 5.17 on Page 5-60.

- UPTF Test 11 – The test assesses hot leg CCFL at the steam generator inlet. The test is run under SBLOCA conditions and the phenomena are applicable to SBLOCA. The key parameters are mass flow rate and CCFL. The discussion of results begin in Section 5.5.3 and comparison with data is shown in Figure 5.19 on Page 5-64.

The experiments listed below are the additional tests used specifically for SBLOCA. Included are lists of key parameters assessed. The tests are documented in EMF-2328(P):

- 2-Dimensional Flow Problems – A set of three steady state flow problems in a bundle test section. The flow was partially blocked in one of the two bundles, providing 2-Dimensional flow data for assessing 2-Dimensional codes. This problem is used to assess the S-RELAP5 2-Dimensional model. The key parameters are pressure drops and velocities. The results are discussed in Section 5.1.

- Semiscale Test S-UT-8 – This is a small scale test that investigated the effects of downcomer to upper plenum bypass on SBLOCA. The significant phenomena observed was a deep, long core level depression and subsequent heat-up prior to loop seal clearing. The portion of the transient used for assessment was the period of core heat-up prior to loop seal clearing and CCFL. The key parameters are cladding temperatures, pressure histories, mass flows, and liquid levels. The results are discussed in Section 5.2.

- LOFT LP-SB-3 – A SBLOCA test with a nuclear core. HPSI was not activated in order to instigate a core heat-up. Upon reaching designated cladding temperatures, a 'feed and bleed' process was activated in the steam generators to bring the system pressure down to accumulator injection pressure, thus terminating the experiment. The heat-up portion of this test was used to assess the dryout wall heat transfer, level model, and the 2-Dimensional model. The key parameters are cladding temperatures, pressure histories, and liquid levels. The results are discussed in Section 5.3.

- UPTF Loop Seal Clearing Test – A separate effects test to show loop seal clearing behavior under typical SBLOCA conditions. This test was used to assess loop seal clearing and horizontal stratified flow. The key parameters are pressure drops and liquid levels. Results are discussed in Section 5.4.

- BETHSY Test 9.1.b – A small scale (1/100 volume, full height) SBLOCA test with 3 loops. HPSI was not instigated to cause core heat-up. Upon reaching designated temperature, steam generators were blown down to atmospheric conditions to bring primary pressure down to accumulator injection pressure. The accumulator injection quenches the core. The experiment was continued past core quenching to show a second loop seal clearing. This test was used to assess relevant SBLOCA phenomena, including loop seal clearing (including second clearing), core heat-up, core quenching, and CCFL. The key parameters are cladding temperatures, pressure histories, pressure drops, mass flows, and liquid levels. The results are discussed in Section 5.5.

The following tests are used to assess the non-LOCA capability and are discussed in EMF-2310(P), Sections 4.2 through 4.5:

- LOFT L6-1 – Loss of load
- LOFT L6-2 – Loss of primary flow.
- LOFT L6-3 – Excessive steam load.
- LOFT L6-5 – Loss of feedwater.

Non-LOCA transients are integral tests that are event focused rather than S-RELAP5 constitutive model focused. The assessments therefore identified that the general system behavior in the simulation was physical (e.g. in a heatup transient, does the coolant expand and the pressurizer level rise? Does the power in the reactor core decrease? etc.). That being the case, the following information was considered important in the LOFT non-LOCA simulations:

- SG Level
- Pressurizer Level
- Pressurizer Pressure
- SG Pressure
- Reactor Power
- Hot Leg Temperature
- Cold Leg Temperature
- SG Steam Flow Rate
- FW Flow Rate (L6-3)
- RCS Flow Rate (L6-2)
- RCP Speed (L6-2)

At a fundamental level these few key parameters characterize the mass and energy in the system.

The additional tests listed below are part of the STS, but were not documented in EMF-2100(P). They are used for S-RELAP5 model assessment. The tests are listed with brief descriptions and the key parameters are identified:

- MIT Pressurizer – The test is used to validate the level model. The key parameters are pressure and liquid level.

- FLECHT-SEASET Tests 31504 – Test 31504 is used to assess dry-wall interphase drag and reflood wall heat transfer. The key parameters are void fraction (or differential pressure), steam temperatures, mass inventory, heat transfer coefficients and wall temperatures. Figures 2 to 8 show some examples of code-data comparisons. The results of the time-step and nodalization study depicted in Figures 6 to 8 are important for validating the flow regime transition regions and criteria. The main purpose of flow regime classification is to provide smooth transitions between different sets of correlations. The physical phenomena are mainly determined by the constitutive correlations used. Step-changes in interphase interaction terms often produce oscillations and distort the solution. Correlations are of little value if a relatively smooth solution can not be obtained. For a system code such as S-RELAP5, the applicability of the flow regime classification is primarily measured by how harmoniously different correlations work together. Therefore, the most important factors in determining the transition criteria and the extent of the transition region are appearance of smooth solutions, number of repeated time steps, time-step and nodalization sensitivities, and mass error. The interphase heat transfer correlation of Equation (3.134) is mainly responsible for the good comparison between measured and calculated steam temperatures shown in Figure 2. With respect to Figure 3, the interphase friction correlation for the inverted-slug flow sets the amount of liquid in the quench front region. The calculated differential pressure indicates that more liquid is present in neighborhood of the quench front region. This is consistent with the lower wall temperatures before quench, as shown in Figures 6 and 7. Due to numerical diffusion inherent with the donor scheme, it is difficult to spread out the liquid in a longer range, which may occur in the experiment. By keeping more liquid in the inverted slug region, a lower amount of liquid is in the upper elevations. This results in good code-data comparison of wall temperatures in the temperature-rise period. As PCT occurs in the temperature-rise period, it is significant that the code has the capability to properly calculate the thermal-hydraulic responses far above the quench front. Figure 8 shows that the calculated maximum temperature points are distributed in the outer envelope of the data points and that the spread due to time step and nodalization sensitivity is much smaller than the spread of data. The interphase friction package is responsible for the bundle mass displayed in Figure 4.

Figure 2. Steam Temperatures Calculated at 6.3 feet and Measured at 6 feet for FLECHT-SEASET Test 31504

Figure 3. FLECHT-SEASET Test 31504 Calculated and Measured Differential Pressures Between 6 and 7 feet.

Figure 4. FLECHT-SEASET Test 31504 Total Mass in the Bundle

Figure 5. FLECHT-SEASET Test 31504 Heat Transfer Coefficients

Figure 6. Calculated Rod Surface Temperatures at 6.6 feet for 20 Volume Core Cases with Various Time Step Sizes for FLECHT-SEASET Test 31504. The Solid Curve is the Data

Figure 7. Calculated Rod Surface Temperatures at 6.6 feet for 40 Volume Core Cases with Various Time Step Sizes for FLECHT-SEASET Test 31504. The Solid Curve is the Data

Figure 8. Code-Data Comparison of Maximum Clad Temperatures vs. Axial Elevation for FLECHT-SEASET Test 31504

- LOFT Tests L2-5 & L2-6 – These tests are used to assess the LBLOCA capability of S-RELAP5. Several phenomena that occur during these tests can be used to assess various models that are also used in SBLOCA. These include phenomena associated with ECC injection (subcooled water injected into superheated steam), horizontal stratification, and interphase condensation. The key parameters are cladding temperatures, pressure histories, mass flows, density, fluid temperatures, and liquid levels.

  Examples of code-data comparisons of key parameters for LOFT 2-6 and L2-5 are shown in Figures 9 to 26. The agreements are in general good. The calculated results are plotted as a solid line and the measured data are plotted as a dashed line.

Figure 9. LOFT L2-6 Broken Hot Leg Mass Flow Rate

Figure 10. LOFT L2-6 Intact Loop Cold Leg Mass Flow Rate

Figure 11. LOFT L2-6 Broken Cold Leg Density

Figure 12. LOFT L2-6 Upper Plenum Fluid Temperatures

Figure 13. LOFT L2-6 Lower Plenum Fluid Temperatures

Figure 14. LOFT L2-6 Intact Loop Hot Leg Fluid Temperatures.

Figure 15. LOFT L2-6 Pressurizer Collapsed Liquid Level

Figure 16. LOFT L2-6 Primary and Secondary Pressures

Figure 17. LOFT L2-6 Central Bundle Cladding Temperatures (Solid Pellet) at 27.5 in.

Figure 18. LOFT L2-5 Broken Loop Hot Leg Mass Flow Rate

Figure 19. LOFT L2-5 Broken Loop Cold Leg Mass Flow Rate

Figure 20. LOFT L2-5 Broken Cold Leg Density

Figure 21. LOFT L2-5 Upper Plenum Fluid Temperatures

Figure 22. LOFT L2-5 Lower Plenum Fluid Temperatures

Figure 23. LOFT L2-5 Intact Loop Hot Leg Fluid Temperatures

Figure 24. LOFT L2-5 Pressurizer Collapsed Liquid Level

Figure 25. LOFT L2-5 Primary and Secondary Pressures

Figure 26. LOFT L2-5 Central Bundle Cladding Temperatures (Solid Pellet) at 27.5 in.

- 2-D Symmetric Fill – This is a simple model with an analytic solution that can be determined visually (by inspection of printed or plotted velocities) for assessing the 2-Dimensional model (see response to Question 2.2). The key parameters are velocities and liquid levels.

- CCTF – Run 54 – This is an integral test to show the LBLOCA capability of S-RELAP5. Several phenomena that occur during these tests can be used to assess various models that are also used in SBLOCA. These include phenomena associated with ECC injection (subcooled water injected into superheated steam), horizontal stratification, interphase condensation, and core heat-up. The key parameters are cladding temperatures, pressure histories, mass flows, mass inventory, differential pressures, void fraction, and liquid levels. Examples of code-data comparisons for some key parameters are shown in Figures 27 to 32. The calculated results are generally in good agreement with the data. Note particularly that the condensation in the cold leg during the ECC injection period is well calculated, as shown in Figure 29. During the short period of accumulator injection, both calculated results and measured data indicate that the cold leg is almost full of liquid (part from ECC injection and part from condensation of steam). During the LPCI injection period, the amount of liquid is too small to be measured accurately by the instrument.

Figure 27. CCTF Test Run 54 Pump-Side Break Mass Flow Rate

Figure 28. CCTF Test Run 54 Intact Loop Hot Leg Mass Flow Rates

Figure 29. CCTF Test Run 54 Intact Loop Cold Leg Void Fraction

Figure 30. CCTF Test Run 54 Downcomer Differential Pressure

Figure 31. CCTF Test Run 54 Core Differential Pressure

Figure 32. CCTF Test Run 54 Heater Rod Surface Temperatures around the Mid-Plane for High Power Bundles

3.3 *(This question is related to large break LOCA (LBLOCA) only and may be responded to at the time of the BE LBLOCA submittal.)*

On Page 3-1, last paragraph, it is stated that the Electric Power Research Institute (EPRI) drift-flux correlations used in RELAP5/MOD3 are tuned mostly to the steady-state data with regular flow profiles and that there is little evidence that these fix-profile correlations produce good results in simulating LBLOCA transients which are highly irregular and chaotic in nature. It is also stated that the EPRI correlations do not cover the entire range of two-phase flow conditions. Based on this information, it was stated that S-RELAP5 did not adopt the same approach as used in RELAP5/MOD3 but that assessment examples are presented to show that the S-RELAP5 two-fluid formulation produces code-data comparisons that are as good as those obtained by RELAP5/MOD3 for steady-state and nearly steady-state cases. Since the concern stated with the EPRI drift-flux correlations was with the modeling of the LBLOCA transients which are highly irregular and chaotic in nature, please provide the assessments that were performed to ensure that the correlations used in S-RELAP5 are adequate for highly irregular and chaotic transient cases.

This question will be responded to as part of the NRC review of the SPC Realistic PWR LBLOCA model (to be submitted).

3.4 *In Equation 3.7, you limited α_L to a minimum value of 0.1 and used $(D^*/19)^8$. Which experiments form the basis for choosing these values? Please justify the use of these values.*

As explained in the paragraph after Equation (3.7), $(D^*/19)^8$ is a way to convert a discontinuous transition criterion of Equation (3.3) into a mathematically continuous formulation. In reactor applications, D* is either much greater than 19 or much smaller than 19; therefore, there is no practical difference between Equation (3.3) and $(D^*/19)^8$. The smaller diameter criterion of Equation (3.3) is mainly applicable to the core in the reactor systems. The core hydraulic diameter is sufficiently small to preclude the presence of the bubbly flow regime. For computational reasons, a narrow region of bubbly flow is required to provide a smooth transition between single phase liquid and slug flow. Historically, values such as 0.02, 0.05 and 0.1 have been used to define a small region of bubbly flow. There are no apparent ill effects from using any one of the values mentioned above. The value of 0.1 is chosen to provide consistency in the transformation of bubbly flow to inverted annular flow (see RAI question 3-23) since a reactor core is the only component where the dry-wall flow regimes may be of significance. Assessments of ORNL THTF Level Swell Tests, LOFT L2-5, L2-6, FLECHT-SEASET 31504 and CCTF Run 54 validate the use of these values (see data comparisons shown in response to Question 3.2).

3.5 *Please describe the tests used in the assessment and provide the assessments performed to validate the use of Equation 3.11 and the limits provided in the text that follows the equation on Page 3-6 and 3-7 in relation to the $\alpha_{S\text{-}A}$ criteria.*

Equation (3.11) is an empirical relation based on theoretical consideration and experimental observation. The justifications for using the relationship of Equation (3.11) are discussed on Pages 3-5 and 3-6. Jones and Zuber (Reference 3.12) experimentally determined that the transition between slug flow and annular flow occurs around a void fraction of 0.8. The separate-effects tests that may be sensitive to this flow regime transition criterion and, therefore, indirectly validate the criterion are: GE 1ft Level Swell Test 1004-3, UPTF Test 11, and Marviken Critical Flow Tests (co-current down flow). Assessment results of tests such as LOFT L2-5 and L2-6, Semiscale Test S-UT-8, UPTF Loop Seal Clearing Test, and Bethsy Test 9.1b also depend on the transition criterion (see data comparisons shown in response to Question 3.2).

3.6 *On Page 3-7, end of the first paragraph, it is stated that introduction of transition regions may reduce the chances of occurrence and magnitude of discontinuities in interphase interaction terms, but it can not completely eliminate the discontinuities. Please describe known discontinuities that still remain and how these are dealt with in the coding of S-RELAP5.*

By incorporating transition regions, there are no mathematical discontinuities between flow regimes. The statement was referring to the evaluation of an interphase interaction terms at successive time-steps where flow conditions are such that different flow regimes are calculated to occur. The resulting values from the interphase interaction terms may differ greatly, appearing to be computationally discontinuous. The effect of these large differences may reduce the quality of the data comparison or cause oscillations of undetermined magnitude. In general, decreasing the time-step size reduces the computational difference between successive time-step interphase interaction terms. However, reducing the time-step size does not guarantee that the computed differences will be sufficiently small so to not affect the quality of the comparison or reduce oscillations to negligible magnitudes.

The last sentence in the paragraph will be rephrased as follows to clarify its meaning:
It should be cautioned that introduction of transition regions may reduce the chances of occurrence of step-changes in magnitude of interphase interaction terms, but it cannot completely eliminate them.

3.7 *Please describe the information used to confirm the validity of the interpolation in Equation 3.15.*

The intent of Equation (3.15) is to bridge two different sets of constitutive equations. The proof of its effectiveness is mainly measured by sensitivities in time-step and nodalization sizes. The FLECHT-SEASET Test 31504, CCTF Run 54, and THTF Level Swell tests (see response to Question 3.2) were used specifically for assessing Equation (3.15). The parameters used for determination of acceptable performance were void fraction and transition to dryout.

3.8 *Under the vertical stratification section starting on Page 3-8, there appear to be no flow/velocity criteria established for when vertical stratification may occur. Please explain how vertical stratification is detected.*

The detection logic for vertical stratification is described on Pages 3-8 to 3-10. The essential point is that there is a sharp void fraction increase in a consecutive three vertical volume stack. Such a condition usually can not be established under high flow conditions. Therefore, it is redundant to include velocity/flow criteria. Nevertheless, for computational efficiency, the detection of vertical stratification is not performed for mass fluxes greater than 1500 kg/m^{2}s. This is simply a filter to exclude the circumstances where Equations (3.16) and (3.17) are nearly impossible to be satisfied.

3.9 *Please describe how the mixture level model described under the vertical stratification section was validated.*

The mixture level model is most critical for handling a condensation process. Under condensation conditions, the mixture level usually becomes a liquid level. The model is validated by 1-D and 2-D fill problems (see response to Question 2.2), the MIT pressurizer problem (qualitatively), and the LOFT non-LOCA Tests (pressurizer behavior). For flashing or boiling conditions, the mixture level provides only a small enhancement on phase separation. For flashing cases with insignificant wall-to-fluid heat transfer, the rapid decrease of interphase friction with increasing void fraction is sufficient, by itself, to produce a sharp mixture level. The assessment of the GE 1ft Level Swell Test validates the mixture level under flashing conditions. Within the PWR applications, the mixture level for the boiling cases is dominated by the transition from pre-CHF to post-CHF heat transfer. The sharp gradient in void fraction is produced by the transition from slug flow to mist (dispersed) flow. The model under such circumstances is validated by ORNL THTF Level Swell Tests, FLECHT-SEASET Test 31504, CCTF Run 54, LOFT L2-5 and LOFT L2-6 (see data comparisons shown in response to Question 3.2).

3.10 *Please describe the assessment performed to justify the method used for the transition region between the stratified and non-stratified flow (i.e., Equation 3.26 and associated restrictions and criteria).*

The primary test used for developing the transition region criteria was the UPTF Loop Seal Clearing test (see response to Question 3.2). Time-step size sensitivities were used to introduce perturbations due to apparent discontinuities between the interfacial drag for horizontal stratified and bubbly/slug flow (see response to Question 3.6). Since the vapor flow exceeded the stable flow criteria and was in the transition region, this process is an acceptable method determining transition region criteria. The acceptance criteria for determining the transition region was consistent liquid levels in the horizontal section when using time step sizes of 5 milliseconds to 100 milliseconds.

3.11 *Justify the choice of 0.9 for j_g* for the boundary between slug and annular mist flow (Equation 3.28) in light of the wide range of 0.25 to 1.0 suggested by Wallis. What are the sensitivities of the results of the analyses of interest to the value of j_g* and why is 0.9 appropriate in light of these sensitivities? What is the range of hydraulic diameters that this criterion is valid for? Please describe the assessment performed to cover the sensitivity to hydraulic diameter. Provide a comparison to applicable experimental data.*

The flow regime classification is an intermediate model necessary and convenient for providing a reasonable approximation of evaluating the interphase friction and interphase heat transfer. High precision of flow regime transition criteria is not warranted since the uncertainty of interphase friction is large. The inclusion of large region of transition before the annular flow boundary further diminishes the importance of the transition line criterion.

For the US PWR plants and their related test facilities, the main horizontal components are hot legs and cold legs. In the case of SBLOCA, the cold legs and hot legs are in bubbly flow during the early period. The flow regime then changes to and stays in the horizontal stratification flow since the vapor velocity is low. All other horizontal flow regimes play no role; therefore, the precision of the annular flow transition criterion is immaterial. For LBLOCA, the annular flow can appear in the hot legs for a short duration (about 2 sec) during the very early period of blowdown when the void fraction is higher than 0.8 and the pressure is still rather high. Under such circumstances, the limit value of 0.8 overwrites the J_g^* criterion. As soon as the pressure decreases and the density ratio of liquid to vapor increases to around 500, the flow regimes of both cold and hot legs become horizontally stratified. At liquid-vapor density ratio of 500, void fraction of about 0.8 and typical hot leg diameter of about 0.75 m, Equation (3.23) yields a critical vapor velocity of $v_{HS} \approx 40 \mathrm{m/s}$. The interpolation scheme used in the transition region [see Equation (3.65)] suggests that the horizontal stratification may be dominant at least up to half of the transition region, i.e., up to vapor velocity of about 70 m/s. Considering the vapor velocity is about 50 m/s during the refill period and about 30 m/s during the reflood period, the horizontal stratified flow is still the most important flow regime in the horizontal components for LBLOCA. Thus, for LOCA, the annular flow in the horizontal component either plays no role or is insignificant; therefore, there is no need to consider the dependency of hydraulic diameter or to determine an accurate value of J_g^*. It should be pointed out, however, that the high value of 0.9 is more appropriate for J_g^*. Since the annular flow can only be present at very high vapor velocity, a low value of J_g^* will yield a void fraction too low to be considered as annular flow.

There are no appropriate data for a direct assessment of flow regime criteria. The assessment can only be performed on the whole constitutive package, not individual pieces. The horizontal constitutive package is validated through examining mass flow rate, fluid density, fluid temperature and void fraction in cold legs and hot legs for LOFT L2-5, LOFT L2-6, CCTF Run 54, and UPTF Test 11 (see data comparisons shown in response to Question 3.2).

3.12 *Please describe how the effect of condensation at the ECCS injection point is handled in S-RELAP5.*

The effect of condensation at the ECCS injection point is generically treated by the condensation mass transfer model, including Equations such as (3.115), (3.116), (3.123), (3.142) and (3.148). There is no special ECCS component or model.

3.13 *Please show how Equation 3.23 is derived from the material in the reference. Also, it appears in Equation 3.23 that the α_g is a subscript to β. Please confirm or correct this.*

3.14 *On Page 3-48, it is stated that various assessment calculations indicate that Equations 3.98 and 3.99 function well. Please identify and discuss the tests that were used in the assessment calculations and the results of the assessment calculations.*

The purpose of Equations (3.98) and (3.99) is to bring any metastable state to as close to the saturation state as possible to prevent unforeseeable numerical difficulties caused by large departure of superheated liquid or subcooled steam state from the saturation state. The term "function well" simply means that the purpose is achieved. All metastate temperatures are close to the saturation and there are no state failures in any assessment calculation. This is a numerical necessity, as explained on Page 3-47. Except for Marviken Critical Flow Tests, there are no experimental data exhibiting effects caused by highly superheated liquid or highly subcooled vapor and the code does not calculate any of them. As for the Marviken Tests, the break flow data show an extremely short period of sudden drop and rise of break flow [see Figure 5.6 on Page 5-28 of EMF-2100(P)] due to the presence of highly superheated liquid right after the break is initiated. The code does not calculate such a sharp drop and rise in break mass flow rate, but the period is too short to be of any significance.

3.15 *Section 3.4.8 discusses the equilibrium option that exists in S-RELAP5. Please provide a table showing when (i.e., in what transient analyses) this option would be allowed and when it would not be allowed. Also, please provide a reference to the section in the user's manual that directs the user to follow these restrictions. If allowed in any of the licensing analyses, please justify the values selected.*

The equilibrium option is not and has not been used in SPC assessment and licensing analysis calculations. The need for guidelines has not been necessary since the code will not run with the option turned on.

3.16 *Section 3.4.9 discusses the effect of noncondensables on condensation rate. Please justify your use of Equations 3.169 and 3.165 in S-RELAP5 to handle the reduction of condensation rate in the presence of noncondensables. Please provide a description and results of assessment calculations that justify the use of these equations.*

The effects of noncondensables on interphase condensation appear in LBLOCA. The tests used for assessment are LOFT tests L2-5 and L2-6. In those tests, subcooled safety injection initiates in the approximate time frame as the accumulator empties of liquid and injects nitrogen into the system. Thus, subcooled liquid is injected into a two-phase mixture with noncondensables present.

The safety injection is delivered to the primary system from a constant head pump which makes the flow dependent on downstream pressure. Under the system conditions with subcooled liquid injected into superheated steam, condensation would occur, causing a slight pressure decrease which would further increase the injection rate. The reduction in condensation due to nitrogen injection from the accumulator increases the downstream pressure and thus reduces the injection rate. Therefore, LPSI flow rate is a key parameter for assessing the effects of condensation with noncondensables present.

From Figure 33, the measured LPSI flow shows a short period of decreased flow indicating that the pressure had increased during that period. The reason for the short period of increased pressure/decreased flow was the decrease in condensation due to the presence of noncondensables.

In S-RELAP5, the LOFT L2-6 LPSI is modeled with a time dependent junction specifying flow as a function of downstream pressure (simulating a constant head pump). As shown in Figure 33, the calculated and measured LPSI initially agree well. Subsequently the calculated LPSI flow rate decreases for a short period when an increasing flow is expected, following the trends measured during the experiment. This comparison shows the effects of Equations (3.165) and

(3.169). The comparison also shows the reduction in condensation is underestimated. This assessment case is used primarily for LBLOCA validation.

Figure 33. Comparison of S-RELAP5 LPSI Flow with Measured Data from LOFT Test L2-6

3.17 *For time smoothing, it is stated on Page 3-68 that the scheme implemented in S-RELAP5 is empirical and that various assessment calculations indicate that it works satisfactorily. Please describe the assessment calculations performed for confirming the time smoothing scheme. In addition, show how the assessment calculations provide a test for the scheme.*

Any test where mass transfer effects dominate the calculated results can be used to demonstrate the effectiveness of the smoothing algorithm. Upon completion of model development, the GE Level Swell Test [see Page 3-43 of EMF-2100(P)] was used to study the effects of mass transfer time smoothing, Equations (3.171) through (3.174). The criteria used for determining the constant in Equation (3.172) was the assumption that fewer repeated time-steps implies a smoother transient.

3.18 *In Section 3.4.10, in relation to mass error, it is stated that S-RELAP5 implements a strategy which forces only condensation to take place when the amount of liquid in a volume is small and subcooled and the vapor is superheated. In addition, this strategy forces only evaporation to take place when the amount of vapor in a volume is small and subcooled and the liquid is superheated. It is stated that these limits have no significant effects on physical results as one would expect from such a diminishing amount of liquid or vapor and that these limits reduce mass error substantially. Please justify your strategy for dealing with the mass error. In your justification, please discuss any assessments that were performed, the tests used in the assessments, and the results.*

3.19 *In Section 3.4.10, in relation to subcooled nucleate boiling, it is stated that S-RELAP5 implements a strategy which lowers the interphase heat transfer coefficients in order to eliminate situations where the total mass transfer rate, Γ_g, becomes negative. Please justify your strategy for dealing with this situation. In your justification, please discuss any assessments that were performed, the tests used in the assessments, and the results. In addition, the last paragraph on Page 3-70 states that there is no guarantee that the final solution at the end of each time step meets all the conditions or limits described in the section. Please explain what is meant by this statement and explain and justify what is done in S-RELAP5 when the conditions or limits are not met.*

The rationale and method for the special treatment of vapor generation under subcooled boiling conditions are discussed on Pages 3-69 to 3-70. In general, the sum of bulk mass transfer (condensing) and wall vapor generation is positive (.i.e., vaporizing) when the wall temperature is above the net-vapor-generation point and the scheme is not applied. However, mismatched conditions may be calculated at times. Mismatched conditions may be ignored or corrected. The treatment used for correcting the model inconsistency in subcooled nucleate boiling is designed to improve the quality of the numerical solutions, such as smoothness in space/time, reducing the number of repeated time steps, and reducing the mass error. The effect on the liquid temperature due to the adjustment of bulk condensation rate to be smaller than the wall vaporization rate is extremely insignificant. The scheme is intended to enhance the numerical performance of the code without affecting significantly the overall physical results. Therefore, its only validation is that the code is numerically performing well on all calculations; i.e., extremely rare code failures, no excessive number of repeated time steps, no appreciable mass error, etc. This is the case. Also, there is no code problem caused by condensation in a subcooled nucleate boiling volume, as it used to be years ago. The table shown in the response to Question 3.18 confirms that this scheme (strategy) together with other special numerical treatments for the mass transfer model produces very good mass conservation in S-RELAP5.

All special treatments discussed in this section are based on old time (i.e., at the beginning of the current time step) information. As shown in Equation (2.197), the new time (i.e., at the end of the time step) vapor generation rate is obtained from the old time vapor generation rate by including the contributions from changes of pressure, liquid energy, vapor energy, and noncondensable quality within the time step. The new time vapor generation rate (part of the final solution) may not satisfy all the conditions or limits imposed by the special treatments. As no check is made on whether any inconsistency is still present at the end of time step, there is no guarantee that the final solution meets all the conditions or limits. However, it is expected that even if some conditions are not met, the discrepancy is not significant enough to cause appreciable solution truncation error. In any case, the final solution is checked against the time-step control criteria described in Section 2.6.7 to ensure solution convergence.

3.20 *Please provide a list of the figures of merit and important phenomena in relation to each of the transients and accidents to be analyzed with S-RELAP5. Please also describe how these figures of merit and important phenomena were designated as important for the relevant analyses.*

In addition to 10.CFR 50.46 requirements of PCT and maximum cladding oxidized, the time histories of the following parameters are reported with a SBLOCA analysis:

- Primary and secondary pressure
- Reactor power
- Core level
- Core collapsed liquid level
- Total primary system mass
- Break mass flow rate
- Void fraction at the break junction
- Combined delivered SI flow
- Combined accumulator flow
- PCT node vapor temperature
- PCT node clad surface temperature
- Rupture node clad surface temperature (only if rupture occurs)
- Steam generator liquid level
- Void fraction in the last node of the loop seal (RCP side)
- Steam velocity in the loop seal
- Metal-water reaction information

A review of the behavior of the above parameters as a function of time is performed to assure that the analysis produces expected results. The choice of those parameters was confirmed by an informal PIRT that was developed to identify important phenomena with respect to SBLOCA transient period. A summary of the results of the informal PIRT for the SBLOCA event is shown in Table 2 below.

3.21 *In Sections 3.4.1 through 3.4.7 heat transfer correlations, limits on these correlations, and transition equations are presented for different flow regimes. However, no justifications are provided. Please provide justifications for the material presented in these sections and provide discussion of assessments performed to confirm the adequacy of correlations used in S-RELAP5.*

The limits on the interpolation parameters for smoothing are Equations (3.103), (3.114), (3.119), (3.124), (3.136) and (3.150). They define the transition region between two correlations of different valid ranges, for example, a subcooled correlation and a superheated correlation. In the transition region, they have the values between 0 and 1. They usually can and do take the value of either 0 or 1 to select one of the correlations. Many of the limits are simply the maximum of two correlations. They include Equations (3.101), (3.123), (3.146), (3.151), and (3.159). The approach is standard. The rest are limits placed on the phasic velocities. These are in Equations (3.125), (3.142), (3.144) and (3.148). They are numerical necessities to filter out fluctuations in code-calculated phasic velocities. They were put in to improve reliability of the code calculations.

The limits and the correlations work together as an integral package. Some of the assessments that justify/validate the mass transfer constitutive package are LOFT L2-5, LOFT L2-6, CCTF Run 54, ORNL THTF Level Swell Tests and FLECHT-SEASET 31504 (see data comparisons shown in response to Question 3.2). From LOFT L2-5 and L2-6 assessments, code data comparisons are performed on fluid temperatures at various locations, and density comparisons in cold and hot legs. In CCTF Run 54, the cold leg void fraction is a good test for the condensation model. The ORNL THTF Level Swell Tests assess the subcooled nucleate boiling model. Code-data comparisons of steam temperatures for the FLECHT-SEASET test validates the vaporization model for superheated steam. Also, the depressurization rates in blowdown calculations such as the LOFT tests and Marviken Critical Flow Tests are affected by the vaporization model of superheated liquid.

3.22 *Page 3-11, last paragraph, it is stated that "...some calculations with RELAP5/MOD2 indicated that the range of stratified flow is too small. Kukita et al suggested that the vapor velocity on the left side of Equation 3.22 be replaced by the relative velocity (v_g-v_f). This approach along with an additional constraint to exclude high mass flux conditions was implemented in the previous S-RELAP5 code versions. Recent experience with small break test cases and plant calculations indicated that the new approach might increase code variability. Therefore, the approach of replacing the vapor velocity with relative velocity is abandoned."*

Since the approach was abandoned, what was done to address the concern that the range of stratified flow was too small and how was that justified? Please provide comparisons of your approach to data to justify the adequacy of your approach.

The concern needs to be addressed because RELAP5/MOD3 uses similar approach (i.e., relative velocity and a mass flux criterion). The information is useful for the code developers so that they know the approach was tried once. Actually, the range of stratified flow defined by Equation (3.23) is not small at all for the diameter size of typical PWR hot and cold legs. This can be seen from Fig. 6 of Reference 3.3 (Taitel's paper). The region of stratified flow expands substantially with increasing diameter. The response to Question 3.11 also shows that the range of stratified flow is rather large under typical LBLOCA conditions of hot and cold legs. For PWR SBLOCA, with Equation (3.23) the flow regime in the cold/hot legs stays always in the stratified flow, but not so with the approach using relative velocity plus an additional constraint. The assessments of LOFT L2-5 and L2-6, CCTF Run 54 and UPTF Test 11 show that Equation (3.23) is applicable to both large and small diameters (see data comparisons shown in response to Question 3.2).

3.22 (This question is related to LBLOCA only and may be responded to at the time of the BE LBLOCA submittal.)

For dry-wall flow regimes, please justify your use of 0.1 for the $\alpha_{IA\text{-}IS}$ criterion in light of the information provided in the text preceding Equation 3.13 that indicates that the transformation of the three wet-wall flow regimes into inverted annular, inverted slug, and mist flow regimes should be used.

In US reactor applications, the classification of dry-wall flow regimes is really required only in the core. As discussed in response to Question 3.4, the bubbly flow boundary for the core is set at void fraction of 0.1 to be consistent with the $\alpha_{IA\text{-}IS}$ criterion.

Chapter 4: Heat Transfer Models

4.1 *In reviewing Section 4, Heat Transfer Models, it is apparent that this section is totally different to any comparable heat transfer section in RELAP5/MOD2. Contributions from various known sources constitute the basis for this heat transfer model.* Please provide qualitative (and quantitative) *justification for the formulation of this particular heat transfer model. (i.e., assumptions, mass flow rates, pressure, enthalpy, etc.).*

Most heat transfer correlations in S-RELAP5 are inherited from RELAP5/MOD2 with or without minor modifications. In the code manual, the RELAP5/MOD2 heat transfer equations are written for the heat transfer rates into hydro volumes, while the S-RELAP5 heat transfer equations are expressed in terms of the heat flux and heat transfer coefficient. The boundary conditions for the conduction solution scheme are expressed in terms of heat transfer coefficients and heat fluxes in both RELAP5/MOD2 and S-RELAP5. The selection logic for heat transfer regimes is somewhat simplified in S-RELAP5, but the regimes are essentially the same in both codes.

Equation (4.1) is a general expression for total heat in the RELAP5 series of codes, including RELAP5/MOD2 and S-RELAP5. The same is true for the heat transfer coefficients, Equation (4.2). Note that not all of the terms in Equations (4.1) and (4.2) may be present for a given heat transfer regime, as explained on Page 4-1. For example, the subcooled nucleate boiling heat transfer is described in S-RELAP5 by Equation (4.15):

$$q'' = h_{mac}(T_w - T_f) + h_{mic}(T_w - T_{sat}) .$$

The heat transfer to the vapor phase is not present, i.e., h_{cg} of Equation (4.1) is zero. The same heat transfer equation is documented in RELAP5/MOD2 (RELAP5/MOD2 Code Manual Volume 1: Code Structure, Systems Models, and Solution Methods, NUREG/CR-4312, Rev. 1, March 1987, Page 109) as

$$Q_{wf} = \left[h_{mic}\,\Delta T_{sat} + h_{mac}\,(T_w - T_f)\right] A_{wf} / V$$
$$Q_{wg} = 0$$

The terms inside the square brackets of the above equation are the same as those on the right side of Equation (4.15) of S-RELAP5. Also the correlations for h_{mic} and h_{mac} are the same for both codes. In general, there are no differences between RELAP5/MOD2 and S-RELAP5 in heat transfer modeling schemes and principles.

4.2 *On Page 4-2 of the S-RELAP-5 Models and Correlations Code Manual, the last sentence of the last paragraph discusses the issue of reflood being turned off and on. Who decides when or where the option is turned on or off at the appropriate time?*

The reflood model is an input option, which can be selected by the user for some particular heat structures. If the option is selected, the user also has the option to set the time to start the model. The users' manual, RELAP5 Input Data Requirements, EMF-CC-097(P), Revision 4, describes the general recommendations for setting the starting time of the reflood model, but the specific procedures will be stated in the methodology guidelines. For SBLOCA and non-LOCA transients, the reflood model is not used. For LBLOCA applications, the user must follow the LBLOCA methodology guidelines.

4.3 *Please provide an explanation of the difference between the data and the calculational results in Figure 4.3.*

The discrepancy is explained on Page 4-31 of EMF-2100(P). It should be pointed out that this particular case is outside the range of reactor accident applications because the mass flux in the post-CHF regimes under accident conditions will never reach such a high value of 3797.4 kg/m^2-s.

4.4 *How does RELAP-5/MOD2 or MOD3 compare to the same data as that presented in RAI 4.3 above? A comparison of S-RELAP5 and RELAP5/MOD2 against the data and on the same page would help.*

Figure 34 shows measured data and the calculated results from S-RELAP5, ANF-RELAP and RELAP5/MOD3.2. "5379_Calc" is the same as shown in Figure 4.3 in EMF-2100(P) for S-RELAP5, "5379_ANFR" is from ANF-RELAP, which should yield the same result as RELAP5/MOD2, and "5379_MOD3.2" is from RELAP5/MOD3.2. The ANF-RELAP code produces the best post-CHF results because the under-prediction of vapor convective heat transfer is compensated by the use of the modified Bromley correlation at high void fraction (higher elevations). {Note: on Page 113 of the RELAP5/MOD2 manual, it indicates that Dougall-Rohsenow is used. This is incorrect. In all of the released versions of RELAP5/MOD2 and MOD3, the factor $(1-\alpha_g)v_f$ is not included in the vapor phase convective heat transfer computation.} As discussed on Pages 4-16 to 4-18 of EMF-2100(P), two correlations (Forslund-Rohsenow dispersed film boiling and modified Bromley) are used for the film boiling heat transfer in S-RELAP5. This yields much lower film boiling heat transfer than RELAP5/MOD2 at higher elevations where the void fraction is high. In RELAP5/MOD3, a multiplication factor is applied on the modified Bromley correlation to reduce the heat transfer coefficient to liquid at the high void fractions. Therefore, the temperature trend at higher elevations is very similar for S-RELAP5 and RELAP5/MOD3.2.

Figure 34. Comparison of RELAP5 Versions

Chapter 11: Point Kinetics Model

11.1 *On Page 11-16, the last equation has a term missing. The term "* $\mathbf{-V}_{01}$ *" is missing. Compare with Equation 7.6-21 in NUREG/CR-5535, V1.*

Concur. The code manual EMF-2100(P) will be corrected.

The following question is in regard to Topical Report EMF-2328(P) Revision 0:

SB.1 *Please justify use of 0 percent fuel clad preoxidation in the SBLOCA analysis.*

The SPC methodology described in EMF-2328(P), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," results in a conservative calculation of peak local oxidation for comparison to the 17% oxidation criteria of 10 CFR 50.46. The methodology assumes that the pre-accident cladding oxidation is zero in order to maximize the rate and extent of oxidation during a LOCA. This assumption results in higher peak cladding temperatures and higher peak local oxidation than assuming a non-zero pre-accident oxidation value.

Cladding oxidation from two sources is considered: (1) pre-accident or pre-transient oxidation due to corrosion at operating conditions, and (2) transient oxidation which occurs at high temperature during the LOCA. Pre-transient oxidation is determined by a fuel performance calculation and is a function of burnup. Over the burnup range that the fuel rod is at high power and can approach technical specification peaking limits, the pre-transient oxidation is small; however, at high burnups, pre-transient oxidation can become significant.

Transient oxidation is calculated as part of the LOCA analyses. By rule, this oxidation must be computed using the Baker-Just reaction rate equation. Using this equation, the calculated reaction rate decreases in direct proportion to the increase in thickness of the layer oxidized and increases exponentially with absolute temperature. Therefore, the transient oxidation is maximized by minimizing the initial oxidation layer which yields the highest reaction rate. The increased reaction rate produces higher temperatures which further increases the reaction rate, thus compounding the effect.

The reason that the assumption of zero pre-accident oxidation value results in a conservative calculation of peak cladding temperature and total peak local oxidation is that SPC's calculations show that a non-zero pre-accident oxidation assumption reduces the transient oxidation by an amount greater than the pre-accident oxidation. Therefore, the maximum oxidation; i.e., the sum of both pre-transient and transient oxidation is greatest when zero pre-transient oxidation is assumed. These results apply for conditions where the transient oxidation is the dominant contributor to the total oxidation, which is the case for calculated PCTs in excess of 2000°F and for burnups at which peaking can approach the technical specification limits. These are the most limiting cases for both LBLOCA and SBLOCA.

SPC also recognizes that conditions exist where the total oxidation is dominated by the pre-transient oxidation. This situation occurs when lower PCTs are calculated and at high burnups. For cases with low PCTs, the pre-accident oxidation becomes dominant because the transient oxidation is substantially reduced or effectively eliminated due to the low absolute temperature. For high burnups, the transient oxidation is reduced or effectively eliminated due to the inherent low power and associated low transient temperatures, and is further reduced by the presence of a significant initial oxide layer. For these cases, the maximum total oxidation is essentially equal to the initial pre-accident oxidation value. This oxidation value can exceed the value calculated using a zero initial pre–accident oxidation for these conditions; however, the total oxidation is precluded from approaching or exceeding the 17% value by the design limit on pre-accident oxidation. [

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EMF-2310(NP)
Revision 1

SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors

May 2003

Nature of Changes

Item	Page(s)	Description and Justification
1.	All	The company name is changed to Framatome ANP throughout the document.
2.	5-30 to 5-36	Section 5.6, titled "CVCS Malfunction That Results in a Decrease in the Boron Dilution Concentration in the Reactor Coolant (Boron Dilution)" is revised to clarify the applicable model and assumptions when the plant is operating on the SDCS in modes 4 (Hot Shutdown), 5 (Cold Shutdown), and 6 (Refueling). The margin is marked with a vertical bar on lines where the discussion is changed.

Contents

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Nomenclature

Acronym	Definition
2-D	two-dimensional
3-D	three dimensional
ADVs	atmospheric steam dump valves
AFW	auxiliary feedwater
ANF	Advanced Nuclear Fuels Corporation
AOOs	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
BOC	beginning-of-cycle
CE	Combustion Engineering
CEA	control element assembly
CFR	Code of Federal Regulations
CHF	critical-heat-flux
CRGT	control rod guide tube
CVCS	chemical and volume control system
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EAB	exclusion area boundary
ECC	emergency core coolant
ECCS	emergency core cooling system
ENC	Exxon Nuclear Company
EOC	end-of-cycle
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ESFAS	Engineered Safeguard Feature Actuation System
FCM	fuel centerline melt
FSAR	final safety analysis report
HFP	hot full power
HPSI	high pressure safety injection
HTP	High Thermal Performance
HZP	hot zero power
LBLOCA	large break loss-of-coolant accident
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LOCF	Loss of Forced Reactor Coolant Flow
LOEL	Loss of External Load
LOFT	Loss of Fluid Test Facility
LONF	Loss of Normal Feedwater

LPD	local power density
LPZ	low population zone
MDNBR	minimum departure from nucleate boiling ratio
MFW	main feedwater
MSFCV	main steam flow control valve
MSIS	main steam isolation signal
MSIVs	main steam isolation valves
MSSVs	main steam safety valves
MSLB	main steamline break
MTC	moderator temperature coefficient
NI	nuclear instrumentation
non-LOCA	non-loss-of-coolant accident
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OD	outside diameter
PAs	Postulated Accidents
PORV	power operated relief valve
PWR	pressurized water reactor
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RWST	refueling water storage tank
SAFDLs	specified acceptable fuel design limits
SCS	secondary coolant system
SDCS	shutdown cooling system
SER	Safety Evaluation Report
SGs	steam generators
SGTR	steam generator tube rupture
SIS	safety injection system
SRP	Standard Review Plan
SRV	safety relief valve
TM/LP	thermal margin/low pressure
TT	turbine trip
UCBW	uncontrolled bank withdrawal
UPTF	Upper Plenum Test Facility
VHP	variable high power

Abstract

A revised Pressurized Water Reactor (PWR) non-Loss-of-Coolant Accident (non-LOCA) transient analysis methodology is presented that incorporates S-RELAP5 as the systems analysis code in place of ANF-RELAP. The methodology applies to all PWR non-LOCA transients, including Main Steamline Break (MSLB). The methodology retains the previously approved XCOBRA-IIIC methodology for predicting the event-specific minimum departure from nucleate boiling ratio (MDNBR). The methodology is robust, providing assurance that the event-specific acceptance criteria specified in the United States Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) are met.

Loss of Fluid Test Facility (LOFT) calculations demonstrate that S-RELAP5 is capable of modeling the non-LOCA transients for which a systems analysis is required. Sample problem calculations for a PWR demonstrate how the methodology can be applied to analyze events from the major categories of SRP Chapter 15.

1.0 Introduction

Framatome ANP, Inc. (FRA-ANP) plans to use the S-RELAP5 (Reference 1) code for analysis of all events in the SRP (Reference 2) for PWRs that require a system analysis. The NRC has reviewed and accepted the FRA-ANP methodology using the ANF-RELAP code for analyzing non-LOCA transients (Reference 3), the MSLB event (Reference 4), and small break loss-of-coolant accident (LOCA) event (Reference 5) for PWRs. S-RELAP5 is an updated version of ANF-RELAP.

The purpose of this report is to demonstrate the adequacy of the revised FRA-ANP non-LOCA methodology, including the replacement of ANF-RELAP by S-RELAP5. In addition, References 3 and 4 have been combined into a single non-LOCA transients analysis methodology document in a manner that removes all restrictions placed on Reference 3. The report also incorporates events that do not require a systems analysis. This non-LOCA transient analysis methodology will be applied to all PWR plant types designed by Combustion Engineering (CE) and Westinghouse.

The objective in using S-RELAP5 is to apply a single advanced, industry recognized code for all analyses, including LOCA and non-LOCA events. Using a single code that has had extensive review permits the development of one base input deck for the analysis of all events for a particular application. The benefits of using a single code include ease of use by engineers, reduced maintenance requirements on developers, improved quality of both code and applications, and reduction of resources for the NRC review of associated methodology.

S-RELAP5 is a modification of ANF-RELAP. The modifications were made primarily to accommodate large and small break LOCA modeling. S-RELAP5 remains essentially equivalent to ANF-RELAP for non-LOCA applications.

The XCOBRA-IIIC code (Reference 6) will continue to be used to obtain the final predicted MDNBR for each transient event. That is, the core conditions from the S-RELAP5 reactor coolant system (RCS) calculations will be used as input to the existing XCOBRA-IIIC core and subchannel methodology to predict the event-specific MDNBR.

This report describes:

- Transient modeling
- LOFT non-LOCA transient calculations
- Event-specific application methodology
- Sample SRP events.

2.0 Summary of Results

The non-LOCA transient analysis methodology was developed to apply to all of the SRP Chapter 15 events listed in Table 2.1. The methodology is robust, providing assurance that the event-specific acceptance criteria specified in the NRC SRP are met. The Disposition of SRP Chapter 15 Events (Disposition of Events) provides a rigorous assessment of a reactor's existing Chapter 15 analyses of record to determine which analyses must be updated to support a new reload. The strategy for biasing of parameters, consistent with the SRP, provides realistic event simulation while maintaining sufficient conservatism in the calculated results.

Non-LOCA event-specific LOFT calculations are described in Section 4.0 to demonstrate that S-RELAP5 is capable of capturing the essential features of modeling non-LOCA transients for those SRP Chapter 15 events which require a system analysis. Key parameters, such as reactor power, primary and secondary system pressures, mass flow rates in the primary and secondary systems, and levels in the pressurizer and steam generator (SG), all compare well with the results from ANF-RELAP (the currently approved code) and with data from LOFT.

Sample problems for selected events are provided for a CE 2x4 plant. These events include Main Steamline Break (MSLB), Loss of External Load (LOEL), Loss of Forced Reactor Coolant Flow (LOCF), Loss of Normal Feedwater (LONF), Uncontrolled Bank Withdrawal at Power, and Steam Generator Tube Rupture (SGTR). These sample problems demonstrate the application of the methodology to events representing the major categories of Chapter 15 except for category 15.5, which are normally bounded by the other Condition II events.

Differences in key parameters for the events are summarized in Table 2.2. In both the LOFT calculations and the sample problem calculations, the S-RELAP5 predictions are compared to those of ANF-RELAP. They show that S-RELAP5 is essentially equivalent to ANF-RELAP for the modeling of non-LOCA transients.

This report includes additional information on the following subjects:

- Inclusion of upper head heat structures in system model nodalizations (Figure 4.1, Figure 6.1, and Figure 6.4);
- MSLB event-specific methodology (Section 5.4) and associated sample problems (Section 6.1 and 6.2);
- SGTR event-specific methodology (Section 5.5) and sample problem (Section 6.7);

- Boron dilution event-specific methodology (Section 5.6);
- Misloaded assembly event-specific methodology (Section 5.7);
- Control rod ejection event-specific methodology (Section 5.8);
- Radiological Consequences of the Failure of Small Lines Carrying Reactor Coolant Outside Containment (Section 5.9); and
- Fuel rod modeling for fast and slow transients (Section 3.2).

Table 2.1 Applicable SRP Chapter 15 Events

Event	SRP No.	Typical Condition
CATEGORY 15.1 – Increase in Heat Removal by the Secondary System		
Decrease in Feedwater Temperature	15.1.1	II
Increase in Feedwater Flow	15.1.2	II
Increase in Steam Flow	15.1.3	II
Inadvertent Opening of Steam Generator (SG) Relief/Safety Valve[a]	15.1.4	II
Steam System Piping Failures Inside and Outside Containment[a]	15.1.5	IV
CATEGORY 15.2 – Decrease in Heat Removal by Secondary System		
Loss of Outside External Load (LOEL)	15.2.1	II
Turbine Trip (TT)	15.2.2	II
Loss of Condenser Vacuum	15.2.3	II
Closure of Main Steam Isolation Valve (MSIV)	15.2.4	II
Steam Pressure Regulator Failure	15.2.5	II
Loss of Non-Emergency AC Power to the Station Auxiliaries	15.2.6	II
Loss of Normal Feedwater (LONF) Flow	15.2.7	II
Feedwater System Pipe Breaks Inside and Outside Containment	15.2.8	IV
CATEGORY 15.3 – Decrease in Reactor Coolant Flow Rate		
Loss of Forced Reactor Coolant Flow (LOCF)	15.3.1	II
Flow Controller Malfunctions	15.3.2	II
Reactor Coolant Pump (RCP) Rotor Seizure	15.3.3	IV
RCP Shaft Break	15.3.4	IV

[a] This event is analyzed with the Steam Line Break methodology described in Section 5.4.

Table 2.1 Applicable SRP Chapter 15 Events (*Continued*)

Event	SRP No.	Typical Condition
CATEGORY 15.4 – Reactivity and Power Distribution Anomalies		
Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal From a Subcritical or Low Power Startup Condition	15.4.1	II
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2	II
RCCA Misoperation	15.4.3	
Dropped Rod/Bank	15.4.3.1	II
Single Rod Withdrawal	15.4.3.2	III
Statically Misaligned RCCA	15.4.3.3	II
Startup of an Inactive Loop at an Incorrect Temperature	15.4.4	II
Chemical and Volume Control System (CVCS) Malfunction That Results in a Decrease of Boron Concentration (Boron Dilution)	15.4.6	II
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (Misloaded Assembly)	15.4.7	III
Spectrum of Rod Ejection Accidents	15.4.8	IV
CATEGORY 15.5 – Increase in Reactor Coolant Inventory		
Inadvertent Operation of the Emergency Core Cooling System (ECCS) That Increases Reactor Coolant Inventory	15.5.1	II
CVCS Malfunction That Increases Reactor Coolant Inventory	15.5.2	II
CATEGORY 15.6 – Decreases in Reactor Coolant Inventory		
Inadvertent Opening of a Pressurizer Pressure Relief Valve	15.6.1	II
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	15.6.2	II
Radiological Consequences of Steam Generator (SG) Tube Failure	15.6.3	IV

Table 2.2 Summary of Key Parameters

Event	Parameter	S-RELAP5	ANF-RELAP	Difference
Sample Problem				
Pre-Scram MSLB	Peak Power (%)[a]	137.6	137.6	< 0.1
Post-Scram MSLB	Peak Power (%)[a]	8.8	8.9	0.1
LOEL	Peak Pressure (psia)	2691.4	2692.2	0.8
LONF	Minimum SG Mass (%)[b]	20.1	27.4	7.3
LOCF	MDNBR	1.58	1.54	0.04
UCBW at Power	Peak Power (%)[a]	112.2	112.2	< 0.1
SGTR	Affected SG Release (lb_m)	100,800	100,200	600
LOFT				
L6-1 LOEL (LOFT: 15.86 MPa)[c]	Peak Pressure (MPa)	16.02	16.01	0.01
L6-2 LOCF (LOFT: 23.0 s)	Natural Circulation Established (s)	22.6	22.4	0.2
L6-3 Excess Steam Flow (LOFT: 42.6 MW)	Peak Fission Power (MW)	41.6	40.3	0.3
L6-5 LONF (LOFT: 2.14 m)	SG Level (m)	2.05	1.83	0.22

[a] Relative to rated power

[b] Relative to initial mass

[c] LOFT measurement is for pressurizer; calculated values are for bottom of reactor vessel

3.0 S-RELAP5 Modeling

3.1 *System Modeling*

The system analysis is performed with S-RELAP5. A description of S-RELAP5 is provided in Reference 1. The reactor vessel nodalization provides modeling of the key components in the reactor vessel using junctions, volumes, and heat structures. The secondary side includes the tube bundles, feedwater system, separators, steamlines, and turbine simulator.

Nodalizations for specific plant analyses are necessarily different from each other to capture unique design and hardware features (for example, the CE design of the reactor system piping is different than that for Westinghouse). These plant differences are reflected in the nodalization. When the nodalization needs to be revised, consideration is made for realistic modeling of significant phenomena and the need for conservatism. Also, existing nodalization studies are evaluated to determine if additional studies are warranted. Sample nodalizations are provided for LOFT (Figure 4.1) and a sample problem based on a CE 2x4 plant (Figure 6.1 to Figure 6.5).

A complete reactor point kinetics model simulates the production of nuclear power in the core. The model computes both the immediate fission power and the power generated from decay of fission fragments and actinides. The model provides capability to include feedback due to moderator density and fuel temperature changes.

The control systems coincide with the model's nodalization so that the transient initiation requires a minimum amount of user input. The typical main control systems modeled for these events are:

- Automatic rod control
- RPS control
- Pressurizer heater and spray control
- Steam flow and turbine control
- SG liquid level and feedwater control
- Primary and secondary relief valve control
- Safety injection

The RPS controller trip functions include the trip function and the uncertainties and time delay associated with it. The RPS trips incorporated into the model (for a typical CE plant) are:

- Low SG pressure
- Low pressurizer pressure
- High pressurizer pressure
- Thermal Margin/Low Pressure (TM/LP)
- VHP
- Low RCS flow
- SG low level
- Engineered Safeguard Feature Actuation System (ESFAS) signals

The control logic is versatile enough to establish and maintain steady-state operating conditions. It will also initiate and analyze transients. The transient events typically initiate as restart runs from the established steady-state run.

3.2 *Fuel Modeling*

The approach to fuel modeling is changed from that described in Reference 3. It is still based on RODEX2 (References 7 and 8), however.

Fuel modeling contributes to the determination of the power and heat flux for the core. The heat flux determines the core coolant heating rate and, ultimately, the temperature response of the RCS to power changes. The power is affected by changes in fuel temperature, that determines the Doppler feedback, and by the change in the core coolant temperature, that determines the moderator feedback. Studies of the relevant fuel parameters using FRA-ANP's fuel design code, RODEX2 (Reference 7) are described below.

The reactor core is typically modeled as a single hydraulic channel with axial volumes, each of which is coupled to a heat structure. These heat structures model axial segments of the fuel. The heat structures represent the fuel, cladding, and fuel-to-cladding gap by a series of concentric cylinders. The radial mesh points of the heat structures are characterized by radial locations of the boundaries, relative power in the volume, heat capacity, and thermal conductivity. Most of the radial mesh points represent the fuel pellet stack. The gap is represented by one radial mesh point, and the cladding by two or more.

Fuel Design Studies

The average core behavior and hot rod behavior for PWR fuel designs were evaluated using RODEX2. The effects of gap conductance, heat capacity, thermal conductivity, porosity, and exposure were assessed and are discussed here.

The gap conductance varies significantly throughout the cycle. However, it is not strongly dependent on the fuel design. All four fuel designs show similar variations in gap conductance with power (fuel temperature) and burnup. Near the beginning-of-cycle (BOC), the average value of gap conductance at full power is about 1200 Btu/hr-ft^2-°F, independent of the fuel design. It increases until the end-of-cycle (EOC), where it is greater than 5000 Btu/hr-ft^2-°F. At higher powers, the fuel temperature increases, and both the contact pressure between the pellet-cladding and the gap conductance also increase.

Heat capacity and thermal conductivity models from RODEX2 were evaluated over a range of temperatures. The heat capacity for a fuel pellet varies with temperature and, to a very small extent, pellet densification. The thermal conductivity of a fuel pellet varies with temperature and with porosity. Pellet porosity is radially distributed and varies throughout the cycle. The RODEX2 evaluation shows that pellet porosity and its changes throughout the cycle are relatively independent of the fuel design. Burnup adjustments can be incorporated in both uranium dioxide (UO_2) and (when appropriate) Gd_2O_3 properties to conservatively represent the time in cycle.

The model applies to PWR fuel designs. Significant changes in fuel design, that fall outside the study, require reassessment to determine if the model remains applicable. The characteristics which determine when re-evaluation is required are free volume to fuel ratio, cladding type (creepdown behavior), porosity, a change in certain RODEX2 properties (pellet resintering, pellet cracking, porosity), cycle exposure, and/or a change in the fuel rod code (use of a code other than RODEX2).

The impact of fuel modeling on transient heat flux and fuel temperature is discussed below. Slow transients and fast transients are considered separately.

Slow Transients

Most of the transients to be evaluated using this methodology are considered to be slow transients. In almost all cases, a power transient is associated with the event. During a complete operating cycle, the effective time constant ranges from about 5 seconds at BOC to about 3 seconds at EOC. For slow transients, this range of responses results in a relatively minor, but noticeable, difference in the heat flux from the core. For the average core, FRA-ANP uses average fuel rod properties based on the fuel design studies, which represent the appropriate time in the cycle. By so doing, FRA-ANP's transient analysis captures the transient response of the fuel in the core and results in heat fluxes that are consistent with the fuel rod thermal properties in the fuel design code.

The challenge to the fuel centerline melt (FCM) limit is generally evaluated statically for these slower transients, although it may be evaluated using the more mechanistic hot spot model described below. In the static evaluation, the maximum effective linear heat generation rate (LHGR) (based on rod surface heat flux rather than neutron power) for a UO_2 pellet is compared to a bounding melt limit to determine whether FCM has occurred.

Fast Transients

Fast transients, such as Control Rod Ejection or Uncontrolled Bank Withdrawal from Hot Zero Power (HZP), are characterized by rapidly changing power levels. The power changes so rapidly that the rod surface heat flux bears little resemblance to the power. The modeling of the transient response of the fuel can result in significantly different peak heat flux and fuel temperature. This transient response depends on the mass, heat capacity, and thermal conductivity of the fuel and on the gap conductance; therefore, determining appropriate values for these parameters requires more care than for slow transients.

In a fast transient, the average core response will depend on the average fuel properties. The heat capacity of the average fuel rod depends almost entirely on the fuel temperature. The effect of densification on the heat capacity is small (less than 2 percent) and can be ignored. Thermal conductivity depends on the fuel porosity and the fuel temperature. It can change by about 8 to 10 percent over the range of porosities experienced by regions of a fuel pellet during operation. Fuel porosity varies radially in the pellet and changes with burnup. The thermal conductivity is adjusted to account for the porosity distribution of the average core at the burnup of interest.

Gap conductance varies with burnup and fuel temperature. When the event is initiated from HZP, the gap conductance will range from several hundred Btu/hr-ft^2-°F at the beginning of the transient to several thousand Btu/hr-ft^2-°F at the time of the peak fuel temperature. The power transient for such an event will be arrested by negative Doppler. [

]

Hot Spot Model

To demonstrate protection against FCM dynamically, an additional heat structure is added to the model. This heat structure, called the hot spot, represents the axial segment of the hot rod that has the maximum power. The physical modeling, except for the length and the total power, is the same as the average fuel rod. It is attached to the uppermost volume of the average core, to obtain the highest coolant temperature.

The thermal properties used for the hot spot heat structure are consistent with the exposure of the hot rod. For fuel designs in which the rods with burnable poison (Gd_2O_3) can reach a power close to that of the limiting UO_2 rod in the core, the properties of the Gd_2O_3 rod are used. The thermal conductivity and heat capacity, which are obtained using the models in RODEX2, are significantly different for Gd_2O_3.

The hot spot model provides a conservative calculation of the fuel centerline temperature, which is then compared to the fuel melt temperature.

3.3 *Summary of Code Differences Between S-RELAP5 and ANF-RELAP*

The S-RELAP5 code evolved from FRA-ANP's ANF-RELAP code, a modified RELAP5/MOD2 version, used at FRA-ANP for performing PWR plant licensing analyses including small break LOCA analysis, steamline break analysis, and PWR non-LOCA Chapter 15 event analyses. The code structure for S-RELAP5 was modified to be essentially the same as that for RELAP5/MOD3, with the similar code portability features. The coding for reactor kinetics, control systems and trip systems were replaced with those of RELAP5/MOD3. Most of the modifications to S-RELAP5 were undertaken to improve its applicability for the realistic calculation of Large Break LOCA (LBLOCA) and are irrelevant to the analysis of PWR Non-LOCA events.

In Section 1.1 of the S-RELAP5 Models and Correlations Manual (Reference 1) there is a list summarizing major modifications and improvements in S-RELAP5 that is reproduced below:

(1) Multi-Dimensional Capability

Full two-dimensional treatment was added to the hydrodynamic field equations. The 2-D capability can accommodate the Cartesian, and the cylindrical (z,r) and (z,θ) coordinate systems and can be applied anywhere in the reactor system. Thus far FRA-ANP has applied 2-D modeling to the downcomer, core, and upper plenum. Some improvements were also made to the RELAP5/MOD2 cross flow modeling. If necessary, 3-D calculations can be approximated by 2-D plus one direction of cross flow. The application of a 2-D component in the downcomer is essential for simulating the asymmetric ECC water delivery observed in the Upper Plenum Test Facility (UPTF) downcomer penetration tests. Note that the 2-D component was not used in the Non-LOCA event analyses.

(2) Energy Equations

The energy equations of RELAP5/MOD2 and RELAP5/MOD3 have a strong tendency to produce energy error when a sizable pressure gradient exits between two adjacent cells (or control volumes). This deficiency is a direct consequence of neglecting some energy terms which are difficult to handle numerically. Therefore, the energy equations were modified to conserve the energies transported into and out of a cell (control volume). For LOCA calculations, no significant differences were calculated in the key parameters such as clad surface temperature, break mass flow rate, void fraction, and others between the two formulations of the energy equations. For analyses involving a containment volume, the new approach is more appropriate. This code improvement had only a minor effect on the Non-LOCA event analyses. Specifically, a small effect on critical flow for steamline breaks was observed.

(3) Numerical Solution of Hydrodynamic Field Equations

The reduction of the hydrodynamic finite-difference equations to a pressure equation is obtained analytically by algebraic manipulations in S-RELAP5, but is obtained numerically by using a Gaussian elimination system solver in RELAP5/MOD2 and MOD3. This improvement aids computational efficiency and helps to minimize effects due to machine truncation errors.

(4) State of Steam-Noncondensible Mixture

Computation of state relations for the steam-noncondensible mixture at very low steam quality (i.e., the ratio of steam mass to total gas phase mass) was modified to allow the presence of a pure noncondensible gas below the ice point (0°C). The ideal gas approximation is used for both steam and noncondensible gas at very low steam quality. This modification is required to correctly simulate the accumulator depressurization and to prevent code failures during the period of accumulator ECC water injection.

(5) Hydrodynamic Constitutive Models

Significant modifications and enhancements were made to the RELAP5/MOD2 interphase friction and interphase mass transfer models. The constitutive models are flow regime dependent and are constructed from the correlations for the basic elements of flow patterns such as bubbles, droplets, vapor slugs (i.e., large bubbles), liquid slugs (i.e., large liquid drops), liquid film and vapor film. Some flow regime transition criteria of RELAP5/MOD2 were modified to make them consistent with published data. When possible and applicable, literature correlations are used as published. A constitutive formulation for a particular flow regime may be composed of two different correlations. Transition flow regimes are introduced for smoothing the constitutive models. Partition functions for combining different correlations and for transitions between two flow regimes are developed based on physical reasoning and code-data comparisons. Most of the existing RELAP5/MOD2 partition functions were not modified or only slightly modified. The vertical stratification model implemented in ANF-RELAP was further improved. Also, the RELAP5/MOD2 approximation to the Colebrook equation of wall friction factor is known to be inaccurate and is, therefore, replaced by an accurate explicit approximate formulation of Jain.

No major effects on non-LOCA event analyses resulted from these code improvements. However, as discussed in the LOFT benchmark and PWR sample problem analyses, some effects are noticeable due to the effects of wall drag on SG tube pressure drop and due to the effects of interfacial friction on SG secondary side liquid distribution.

(6) Heat Transfer Model

The use of a different set of heat transfer correlations for the reflood model in RELAP5/MOD2 was eliminated. Some minor modifications were made to the selection logic for heat transfer modes (or regimes), single phase liquid natural convection and condensation heat transfer. The Lahey correlations for vapor generation in the subcooled nucleate boiling region were implemented. No changes are made to the RELAP5/MOD2 CHF correlations.

(7) Choked Flow

The computation of the equation of state at the choked plane was modified. Instead of using the previous time step information to determine the state at the break, an iterative scheme is used. This modification was also implemented in ANF-RELAP. Some minor modifications were also made to the under-relaxation scheme to smooth the transition between subcooled single phase critical flow and two-phase critical flow. Moody critical flow model is also implemented, but not used for the realistic LBLOCA calculations though it is used for the Appendix K analyses.

(8) Counter-Current Flow Limiting

A Kutateladze type CCFL correlation was implemented in ANF-RELAP. This was replaced in S-RELAP5 by the Bankoff form, which can be reduced to either a Wallis type or a Kutateladze type CCFL correlation. RELAP5/MOD3 also uses the Bankoff correlation form.

(9) Component Models

The EPRI pump performance degradation data was included in the S-RELAP5 pump model. The computation of pump head in the fluid field equations was modified to be more implicit. A containment model was added. With this model, the containment pressure boundary conditions are provided by the approved EXEM/PWR evaluation model code, ICECON, which is run concurrently with S-RELAP5 using realistic values for parameter input. The accumulator model was eliminated because of its well-known problems. With S-RELAP5, the accumulator is to be modeled as a pipe with nitrogen or air as noncondensible gas. The ICECON containment model is not used in non-LOCA transient analyses.

(10) Fuel Model

Initial fuel conditions are supplied by the FRA-ANP realistic fuel performance code, RODEX3. The fuel deformation and conductivity models of RODEX3 were included in S-RELAP5. The plastic strain and metal-water reaction models were taken from RELAP5/MOD3 with minor modifications. RODEX2 has also been incorporated into S-RELAP5. The internal RODEX2 and RODEX3 models are not used for non-LOCA transients.

Despite this extensive list of modifications, only three model changes were discerned to be responsible for the relatively minor differences observed in the S-RELAP5 and ANF-RELAP calculations for the PWR Non-LOCA event sample problems. Specifically, it was found that changes to the single-phase wall drag, the interfacial shear package, and the energy equation affected the results. Nevertheless, no significant differences in the parameters that directly affect the specified acceptable fuel design limits (SAFDLs) were observed in these sample problems. Each of the LOFT calculations and PWR sample problems are discussed in Sections 4.0 and 6.0 and differences in the code-to-code predictions are examined with respect to these code modifications.

4.0 LOFT Non-LOCA Transient Calculation

This section presents the S-RELAP5 simulation of the LOFT L6-1, L6-2, L6-3, and L6-5 experiments. The S-RELAP5 calculations are compared to ANF-RELAP calculations to demonstrate the similarity of the two codes for non-LOCA analysis. ANF-RELAP was evaluated against the experimental data in Reference 3. LOFT measured data are provided also. ANF-RELAP results are provided for comparison, but unless specifically stated, the discussion of results is based on S-RELAP5 predictions.

The LOFT integral test facility was designed to simulate the major components of a four-loop, commercial PWR, thereby producing data on the hydraulic, thermal, nuclear, and structural processes expected to occur during anticipated or postulated accidents in a PWR. A general description of the LOFT facility and tests is given in Reference 9. References 10 and 11 provide detailed descriptions of test configurations, instrumentation, experimental procedures and results for L6-1, L6-2, L6-3, and L6-5. The information for simulating the LOFT facility and L6 experimental conditions was obtained from References 12, 13, and 14 and the electronic data received from the NRC databank.

4.1 *LOFT S-RELAP5 Model Description*

The LOFT control system, similar to a large PWR, contains many active subsystems, such as the feedwater control system, High Pressure Safety Injection (HPSI), pressurizer pressure control system, main steam flow control valve (MSFCV) control system, and reactor scram controls.

This section outlines the structure of the S-RELAP5 base deck that was used in the LOFT L6 series calculation. The schematic of the S-RELAP5 model displaying the thermal-hydraulic components and heat structures for the LOFT L6 experimental series is shown in Figure 4.1. The numbering scheme for components in this model is:

Major Component	Numbering
loop one	100 - 199
reactor	200 - 299
loop two	300 - 399
pressurizer	400 - 499

secondary side of SG	500 - 599
ECC system	600 - 699
containment	800 - 899

Heat Structures are included in the model also. The systems with heat structures include the reactor, SG, piping, and pressurizer.

The steady-state control section contains a reactor coolant flow rate control system, pressurizer pressure control system, pressurizer liquid level control system, SG temperature control system, and SG liquid level control system. These systems systematically adjust the RCS flow rate, pressurizer pressure and liquid level, SG temperature and liquid level, and reactor power to the specified initial conditions for a given experiment. The transient control system consists of a RCS pump control, reactor control, pressurizer control, SG control, and emergency core cooling system (ECCS) control systems.

Figure 4.1 S-RELAP5 Nodalization Schematic for LOFT Experiments

4.2 *LOFT L6-1 Loss of Load*

Event Description

The objectives of Experiment L6-1 are:

- To investigate plant response to a transient in which the heat removal capabilities to the secondary system are significantly reduced.
- To evaluate the automatic recovery methods in bringing the plant to a hot-standby condition.
- To provide data to evaluate computer code capabilities to predict secondary initiated events.

This event challenges both primary and secondary system over pressurization limits. The loss of heat removal capability creates a mismatch between heat removal and heat generation, leading to increases in temperature in both the primary RCS and secondary of the SG. The expansion of the primary coolant leads to a pressurizer insurge. Pressurizer sprays are activated to control the pressure. The reactor is tripped on an RCS high pressure signal.

In LOFT, the role of the MSFCV changes after event initiation and closure. The valve then behaves like a secondary side PORV, to relieve the pressure in the SG. When the SG pressure reaches the high pressure setpoint, the MSFCV opens to relieve pressure. Opening the valve increases heat removal from the secondary (and primary), driving the system pressure down. When the low pressure setpoint is reached, the valve closes. Stored heat and decay heat slowly increase the pressure until the MSFCV cycles again.

Analysis Results

Experiment L6-1 was initiated by the closure of the MSFCV. As soon as the MSFCV starts to close, the heat transfer from the RCS to the SCS decreases forcing the RCS temperature and pressure to increase. The temperature increase changes the density of the coolant causing an insurge into the pressurizer. The pressurizer spray was initiated at 8.8 seconds. Because the pressurizer spray was much cooler than the cold leg coolant temperature, the pressure rise is momentarily reversed. The pressure of the RCS increases until it reaches the high pressure scram setpoint.

As the primary system temperature increases, the reactor power decreases due to the moderator temperature and Doppler feedbacks. The reactor scrams at 22.3 seconds on high primary system pressure. Maximum pressure is reached at 23.0 seconds.

The pressure in the SG reaches the high pressure setpoint at 20.2 seconds causing the MSFCV to begin to open. The removal of energy from the secondary, in conjunction with decreased energy production in the primary following reactor scram drives the pressure downward. The pressurizer level reaches its maximum at 26.7 seconds. The decrease in pressure turns the spray off at 26.8 seconds. The depressurization continues causing an outsurge from the pressurizer as the primary volume shrinks. The backup pressurizer heaters come on at 27.8 seconds. The MSFCV is closed at 43.0 seconds and the system begins to stabilize. Because of stored heat in the system structures and decay heat from the reactor, the pressure in the SG slowly creeps upward until the high pressure setpoint is again reached. The MSFCV opens at 104 seconds and closes again at 117 seconds. The simulation is terminated at 200 seconds.

The transient event sequence is shown in Table 4.1. Key system parameters are plotted in Figure 4.2 to Figure 4.9. This table and these figures include calculation results from ANF-RELAP for comparison. The LOFT measured results are provided also.

This event challenges both primary and secondary over-pressurization limits. The peak pressures, for both the primary and secondary, are very close for the S-RELAP5 and ANF-RELAP calculations. Nevertheless, there is a noticeable difference in the event chronology for the calculations of the two codes. Specifically, the S-RELAP5 code predicts the reactor trip on high pressurizer pressure to occur at 22.3 seconds whereas it occurs 3.3 seconds earlier in the ANF-RELAP calculation. After this time, differences between the two codes are affected by this timing difference and so are not examined below.

Examining Figures 4.3 and 4.4 shows that the peak pressurizer pressure occurs earlier in the ANF-RELAP calculation and occurs at a lower pressurizer level. The difference between the two calculations is then the result of a difference in the effect of the pressurizer spray. The pressurizer spray is initiated by a high pressure setpoint (15.25 MPa) at about 8.8 seconds in both calculations. This leads to an initially rapid pressure decrease before the pressurizer insurge overwhelms this effect and the pressure resumes rising towards its peak value.

In the ANF-RELAP calculation, the pressure decrease due to condensation upon the spray causes the pressure to fall slightly below a low pressure setpoint (14.90 MPa) at about 11.4 seconds that shuts off the spray. The pressurizer spray remains off until the pressure once again rises above the high pressure setpoint at about 13.1 seconds. In the S-RELAP5 calculation, however, the pressurizer pressure reaches an initial minimum slightly above (~0.015 MPa) the low pressure setpoint, so that the spray remains on. This interval without the pressurizer spray is responsible for the early reactor trip in the ANF-RELAP calculation. In turn, the more rapid decrease in pressure upon spray initiation is due to a difference in the interfacial heat transfer package between the two codes for the annular/mist regime.

Conclusions

In summary, the S-RELAP5 code compares very well against ANF-RELAP calculated results and provides a satisfactory calculation of the LOFT L6-1 experiment. S-RELAP5 adequately captures the effects of pressurizer insurges, outsurges, condensation due to pressurizer spray, expansion and contraction of the reactor coolant, primary-to-secondary heat transfer, core heat generation, SG pressure, and steam flow. S-RELAP5 is essentially equivalent to ANF-RELAP in modeling this event.

Table 4.1 LOFT L6-1 Event Sequence

Figure 4.2 LOFT L6-1 Steam Generator Level

Figure 4.3 LOFT L6-1 Pressurizer Liquid Level

Figure 4.4 LOFT L6-1 Pressurizer Pressure

Figure 4.5 LOFT L6-1 Steam Generator Secondary Pressure

Figure 4.6 LOFT L6-1 Reactor Power

Figure 4.7 LOFT L6-1 Hot Leg Temperature

Figure 4.8 LOFT L6-1 Cold Leg Temperature

Figure 4.9 LOFT L6-1 Steam Generator Steam Flow

4.3 *LOFT L6-2 Loss of Primary Flow*

Event Description

The objectives of Experiment L6-2 are:

- To investigate plant response to a transient in which forced reactor coolant flow is lost.
- To obtain additional data on the natural circulation mode of cooling.
- To evaluate the automatic recovery methods in bringing the plant to a hot-standby condition, without the RCPs.
- To provide data to assess computer code capabilities to predict primary initiated events.

The LOCF event was simulated in LOFT by tripping the power to both RCPs. A flow coastdown begins. When low flow is detected, the reactor is scrammed and the turbine tripped. The loss of flow diminishes heat transfer from the primary to the secondary, increasing primary temperature and pressure. A pressurizer insurge occurs. Following scram, the mismatch between heat generated in the core and heat removed by the secondary causes a cooldown, resulting in coolant shrinkage. A pressurizer outsurge occurs. Following closure of the MSFCV, and completion of coastdown, natural circulation is initiated and the system reaches a stable state.

The initial power in the LOFT L6-2 test was reduced to approximately 75% of full power so as to not challenge fuel integrity. The purpose of the experiment was to investigate system response to a LOCF event.

Analysis Results

The S-RELAP5 simulation of the L6-2 transient was initiated by tripping the power to both RCPs and allowing them to coastdown. The experiment was scrammed by the detection of a low flow rate in the RCS. No measured low flow scram signal (value of flow rate) was given in the available documents but instead the time of scram was given as 2 seconds into the transient. Therefore, the scram was modeled to occur 2 seconds after the RCS pumps were tripped in the L6-2 simulation.

With reactor trip, the MSFCV began to close at 2.0 seconds, and was fully closed at 14.6 seconds.

The flow rate begins to decrease rapidly. Following an initial slight rise in temperature and, pressure, the pressure begins to drop rapidly, due to continued heat removal from the secondary. The low pressure setpoint is reached at 4.5 seconds, causing the backup heaters to come on, to mitigate the pressure decrease. The feedwater control valve begins to close at the time of reactor trip and becomes fully closed at 4.1 seconds.

At 18.2 seconds, the RCPs decouple from the motor sets and complete the coastdown. An increasing difference in temperature between the upper and lower plenum is first detected at 22.6 seconds as significant nature circulation cooling begins. The decrease in heat removal from the primary, in conjunction with decay heat and stored heat, terminates primary coolant shrinkage at 28.3 seconds, when the minimum level in the pressurizer is reached. The pressure gradually recovers and the pressurizer backup heaters turn off at 48.6 seconds.

The calculation is terminated at 200 seconds with the system in a stable state. The transient event sequence is summarized in Table 4.2. Key system parameters are plotted in Figure 4.10 through Figure 4.19. This table and these figures include calculation results from ANF-RELAP for comparison. The LOFT measurements are provided also.

The calculated transient response of the S-RELAP5 and ANF-RELAP codes is nearly identical for this event. Only minor differences are observed for a transient that included periods of pressurizer insurge, pressurizer outsurge, and loop natural circulation. The one apparently significant difference is the timing for the shutoff of the pressurizer backup heaters as the RCS pressure is recovered. S-RELAP5 calculates that the RCS pressure will reach this setpoint (15.07 MPa) at about 98.6 seconds and ANF-RELAP calculates it to occur at about 117 seconds. At the time the S-RELAP5 calculation reaches this setpoint, the difference in RCS pressure between the two codes is less than 0.038 MPa (5.5 psia). The event timing difference results because the RCS pressurization rate is very slow (~1.8×10^{-3} MPa/s). The observed differences in the calculations for this event are so small that no effect of code modeling differences can be discerned.

Conclusions

In summary, S-RELAP5 compares very well against ANF-RELAP calculated results and provides a satisfactory calculation of the LOFT L6-2 experiment. S-RELAP5 adequately captures RCP coastdown behavior and natural circulation flow rate. In addition, the calculations show overall good agreement with experimental data.

Table 4.2 LOFT L6-2 Event Sequence

Figure 4.10 LOFT L6-2 Reactor Coolant Mass Flow Rate

Figure 4.11 LOFT L6-2 Steam Generator Liquid Level

Figure 4.12 LOFT L6-2 Pressurizer Liquid Level

Figure 4.13 LOFT L6-2 Pressurizer Pressure

Figure 4.14 LOFT L6-2 Steam Generator Secondary Pressure

Figure 4.15 LOFT L6-2 Reactor Power

Figure 4.16 LOFT L6-2 Hot Leg Temperature

Figure 4.17 LOFT L6-2 Cold Leg Temperature

Figure 4.18 LOFT L6-2 Steam Generator Steam Flow Rate

Figure 4.19 LOFT L6-2 RCP Speed

4.4 ***LOFT L6-3 Excessive Steam Load***

Event Description

The objectives of Experiment L6-3 are:

- To investigate plant response to a transient in which the heat removal capability of the secondary system is significantly increased.
- To evaluate the automatic recovery methods.
- To provide data to assess computer code capabilities to predict secondary system initiated events.

An excess load is simulated by opening the MSFCV to its full open position. In response to the increased steam demand, feedwater flow increases. The increased energy removal rate cools the RCS inducing positive reactivity and the power begins to rise to match the increased demand. The cooldown of the RCS causes shrinkage of the coolant and outsurge from the pressurizer, the RCS pressure falls. When the pressure drops to the low pressure setpoint, the reactor scrams. Following scram, the MSFCV begins to close. The cooldown continues and the pressure drops further. HPSI pumps are automatically activated to increase coolant volume and keep the pressurizer from emptying. In conjunction with closing the MSFCV, the pressure and level reach a minimum and begin to recover. As the level and pressure recover, the HPSI is terminated and the system reaches a stable state.

Analysis Results

Experiment L6-3 was initiated by ramping the MSFCV to the fully open position from its steady-state operating position. The increased steam demand was followed by an increase in the main feedwater flow rate at 2.2 seconds. Within 6 seconds, the temperature in the core begins to drop, inducing positive reactivity and an increase in power. The cooling causes coolant shrinkage and decreasing pressure in the RCS system. At 10.4 seconds, the pressurizer backup heaters are activated in response to the decreasing pressure.

The pressure continues to drop and the low pressure setpoint of 14.12 MPs is reached at 18.3 seconds, causing the reactor to scram. Feedwater is tripped at scram. The MSFCV starts to close at 20.5 seconds. The pressurizer liquid level and the RCS pressure continues to fall. HPSI pumps are activated at 26.7 seconds. At this point, steam demand is decreasing because the MSFCV is being ramped closed. Minimum RCS pressure is reached at 34.6 seconds and minimum pressurizer level is reached at 37.3 seconds. The MSFCV is fully closed at 40.6 seconds. The pressure and level begin to recover and the HPSI flow is terminated at 53.0 seconds. As the level continues to recover, the pressurizer backup heaters turn off at 82.0 seconds. The transient calculations are terminated at 200 seconds, with the system in a stable state.

A sequence of events summary is provided in Table 4.3. Key parameters are plotted in Figure 4.20 to Figure 4.28. ANF-RELAP calculated results are provided for comparison to S-RELAP5. LOFT measurements are provided also.

The calculated response of the S-RELAP5 and ANF-RELAP codes is nearly identical for this event up to the time of reactor trip. Reactor scram occurs on a low cold leg pressure setpoint (14.12 MPa) and is predicted at 15.3 seconds in the ANF-RELAP calculation and at 19.0 seconds in the S-RELAP5 calculation. The post-scram differences between the two calculations are primarily due to the additional energy deposition that occurs during this 3.7 second period in the S-RELAP5 calculation. During the RCS cooldown that results from the increase of steam load, the calculated SG heat transfer rate and the shrinkage of the RCS is almost the same for the two codes. However, differences in the interfacial heat transfer package affect the behavior in the pressurizer during the outsurge so that the pressure in the S-RELAP5 calculation reaches the low pressure setpoint later. At this time, the difference in calculated pressure is less than 0.06 MPa (8.7 psia).

<u>Conclusions</u>

In summary, S-RELAP5 compares very well against ANF-RELAP calculated results and provides a satisfactory calculation of the LOFT L6-3 experiment. S-RELAP5 adequately captures the effects of system cooldown, HPSI injection, pressurizer modeling, and primary-to-secondary heat transfer. In addition, the calculations show reasonable agreement with the experimental data.

Table 4.3 LOFT L6-3 Event Sequence

Figure 4.20 LOFT L6-3 Secondary Feedwater Flow Rate

Figure 4.21 LOFT L6-3 Steam Generator Secondary Side Liquid Level

Figure 4.22 LOFT L6-3 Pressurizer Liquid Level

Figure 4.23 LOFT L6-3 Pressurizer Pressure

Figure 4.24 LOFT L6-3 Steam Generator Secondary Pressure

Figure 4.25 LOFT L6-3 Reactor Power

Figure 4.26 LOFT L6-3 Hot Leg Temperature

Figure 4.27 LOFT L6-3 Cold Leg Temperature

Figure 4.28 LOFT L6-3 Steam Generator Steam Flow

4.5 *LOFT L6-5 Loss of Feedwater*

Event Description

The objectives of Experiment L6-5 are:

- To investigate plant response to a transient in which the feedwater flow to the secondary system is stopped.
- To evaluate the automatic recovery methods.
- To provide data to assess computer code capabilities to predict secondary system initiated events.

The LOFT L6-5 event simulates a loss of feedwater event with AFW disabled. It is a heatup event because the secondary heat rejection capability is degraded. The absence of makeup causes the SG liquid level to drop. When the liquid level reaches the low level setpoint in the SG, the reactor is scrammed.

The primary heatup causes the RCS pressure to increase. The expansion of the primary coolant causes an insurge into the pressurizer. The increase in core coolant temperature induces negative reactivity in the reactor and causes the power to decrease.

Following reactor scram, the MSFCV is ramped closed and at this point, the power-heat rejection mismatch is terminated.

Analysis Results

The calculation is initiated by terminating MFW flow. AFW is disabled in the calculation. Gradually, the heat rejection capability of the SG decreases. The effects are felt in the RCS with a gradual increase in pressure beginning at about 6 to 8 seconds. At about 10 seconds, the core coolant temperature begins to change sufficiently that the power begins to decrease.

The SG level drops gradually and reaches the low level setpoint of –0.13 m at 19.5 seconds, causing reactor scram. Following scram, the MSFCV begins to close at 20.7 seconds and terminates steam flow (except for leakage) at 32.2 seconds. At this point, the system stabilizes and the calculations were terminated at 200 seconds.

A summary of the event sequence is provided in Table 4.4. Plots of key parameters are provided in Figure 4.29 to Figure 4.36. The table and figures include results of ANF-RELAP calculations for comparison. LOFT measured data are included also.

The calculated response of the S-RELAP5 and ANF-RELAP codes is very close for this event up to the time of reactor trip. Reactor scram occurs on a low SG level setpoint (2.824 m) and is predicted at 19.5 seconds in the S-RELAP5 calculation and at 21.9 seconds in the ANF-RELAP calculation. After the time of reactor scram, the calculated results are affected by the additional energy deposition during this 2.4 second interval, consequently, only code-to-code differences leading up to this difference in scram time will be examined.

After the termination of the main feedwater, the steam generator mass decreases at almost exactly the same rate for the two calculations. The response of the SG level, however, is somewhat different due to a difference in the initial SG void fraction profiles. Modifications to the interfacial drag package in S-RELAP5 affect the void profile both in the SG boiler region and at the top of the downcomer. In the boiler region, the interfacial drag is reduced and the initial SG mass is ~2.5 percent greater in the S-RELAP5 calculation for the same indicated level.

The difference in interfacial drag packages also affects the transient SG downcomer behavior. For S-RELAP5, a sharper liquid-vapor interface is calculated and when the SG downcomer volume at the feedwater junction starts to drain, condensation begins that temporarily reduces the downcomer-to-boiler flow rate. The subcooling at the boiler inlet decreases, vapor generation increases, and the SG level begins to drop faster than that of the ANF-RELAP calculation. The result is that S-RELAP5 predicts a reactor trip on low SG level about 2.4 seconds earlier than ANF-RELAP. During this initial heatup period, until the time of reactor trip for S-RELAP5, no significant differences were observed in the calculated response for either the reactor power or the SG heat removal rate.

Conclusions

In summary, S-RELAP5 agrees reasonably with the ANF-RELAP calculations and provides a satisfactory calculation of the LOFT L6-5 LONF experiment. The code adequately captures secondary side heat transfer and inventory changes. The results are consistent with the LOFT measurements.

Table 4.4 LOFT L6-5 Event Sequence

Figure 4.29 LOFT L6-5 Steam Generator Secondary Side Liquid Level

Figure 4.30 LOFT L6-5 Pressurizer Liquid Level

Figure 4.31 LOFT L6-5 Pressurizer Pressure

Figure 4.32 LOFT L6-5 Steam Generator Secondary Pressure

Figure 4.33 LOFT L6-5 Reactor Power

Figure 4.34 LOFT L6-5 Hot Leg Temperature

Figure 4.35 LOFT L6-5 Cold Leg Temperature

Figure 4.36 LOFT L6-5 Steam Generator Steam Flow

4.6 *LOFT Analysis Conclusions*

The results of these analyses of the LOFT L6 experiments indicate good agreement between the S-RELAP5 calculated results and ANF-RELAP calculated results. The analyses test both the component and heat structure nodalization and the simulation capabilities of S-RELAP5. The simulation capabilities tested include the modeling of automatic control components and systems, such as MSFCV, pressurizer spray, pressurizer heaters, feedwater control, pressure control, SG liquid level control, and reactor scram. The S-RELAP5 thermal-hydraulic components and heat structure nodalization provide information on the adequacy of the reactor coolant loop flow dynamics, pressurizer pressure adjustments, core kinetics, reactor coolant loop thermal transport, SG heat transfer, and secondary system thermal-hydraulic behaviors.

5.0 Event-Specific Methodology

This section describes the application of this methodology. It describes the SRP Chapter 15 events for which the methodology is to be applied, the Disposition of Events Review process for both initial and follow-on reloads, and the biasing of parameters. It also includes a discussion of events that were either not described in the previously approved methodology (Reference 3) or that need further clarification.

The MSLB methodology (Reference 4) is merged into this report to replace ANF-RELAP with S-RELAP5 and to have one report that covers non-LOCA transients. The Boron Dilution and Misloaded Assembly events were not described in Reference 3 and are included here for completeness. They are also events which do not require system models. The Control Rod Ejection event is added to describe the thermal-hydraulic evaluation of the event. The Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment event is included to address a Safety Evaluation Report (SER) restriction on Reference 3. The SGTR event is discussed explicitly to address an SER restriction on the Reference 3 methodology.

5.1 *Scope of Application*

The methodology is applicable to all CE and Westinghouse plant designs and all modes of plant operation. The methodology is related to the thermal-hydraulic aspects of the SRP events and does not include analysis of radiological dose consequences. However, this methodology provides input to radiological consequence analyses.

The methodology is applicable to the SRP Chapter 15 non-LOCA events listed in Table 2.1. The events are listed according to the event categories given in the SRP. The methodology is applicable to Condition I, II, III, and IV events. The event frequency classifications are:

- CONDITION I: events expected to occur frequently in the course of power operation, refueling, maintenance, or plant maneuvering.
- CONDITION II: events expected to occur on a frequency of once per year during plant operation.
- CONDITION III: events expected to occur once in the lifetime of the plant.
- CONDITION IV: events not expected to occur that are evaluated to demonstrate the adequacy of the design.

The Condition I events, in part, establish the initial conditions for the analysis of more severe events, while the Condition II through IV events normally constitute the licensing analyses. The Condition II events are Anticipated Operational Occurrences (AOOs), and the Condition III and IV events are Postulated Accidents (PAs). The classification of a given SRP Chapter 15 event may vary depending on a given plant's licensing basis.

Licensing analyses are performed to support plant operation. This is demonstrated by meeting the applicable acceptance criteria for each event. The acceptance criteria are defined in each plant's licensing basis and may differ from the criteria specified by the SRP. The acceptance criteria for Condition II, III, and IV events, as specified in the SRP, are:

Condition II Events (AOOs):

- Pressures in the RCS and main steam system are less than 110 percent of design values.
- Fuel cladding integrity is maintained by ensuring that SAFDLs are not exceeded.
- Radiological consequences are less than the 10 CFR 20 guidelines.
- The event does not generate a more serious plant condition without other faults occurring independently.

Condition III Events (PAs):

- Pressures in the RCS and main steam system are less than 110 percent of design values.
- A small fraction of fuel failures may occur, but these failures do not hinder core coolability.
- Radiological consequences are a small fraction of the 10 CFR 100 guidelines (generally less than 10 percent).
- The event does not generate a limiting fault or result in the consequential loss of the reactor coolant or containment barriers.

Condition IV Events (PAs):

- Radiological consequences do not exceed 10 CFR 100 guidelines.
- The event does not cause a consequential loss of the required functions of systems needed to maintain the reactor coolant and containment systems.

Additional event-specific criteria are described, as required, in the appropriate section.

5.2 *Application Process*

All events listed in Table 2.1 that constitute the licensing basis for a given plant may be analyzed using this methodology. A Disposition of Events may be performed to limit the number of events that are analyzed.

5.2.1 Disposition of Events

The purpose of a Disposition of Events review is (1) to evaluate the impact of changes to key parameters on the safety-related analyses supporting a plant's licensing basis, and (2) to determine the scope of analyses that need to be performed. A Disposition of Events review may be performed for the first FRA-ANP reload in a given plant, for each subsequent reload, and at other times due to changes in plant configuration or operation. The Disposition of Events evaluates changes in (1) plant configuration, operating conditions, Technical Specifications, and reactor protection system (RPS) and other equipment setpoints, (2) fuel design, and (3) neutronics parameters. Additional factors considered in the Disposition of Events include (1) the plant licensing basis, (2) all modes of operation, (3) core exposure, and (4) event initiators.

The Condition II, III, and IV events are divided into event categories in the SRP that have similar characteristics, such as heatup events, cooldown events, and reactivity events. Each SRP event category is considered to determine which events within each category, or events with similar characteristics, are limiting for the specific licensing application.

The Disposition of Events typically dispositions Condition II events and PAs (Condition III and IV events) separately since the acceptance criteria are different for Condition II events and PAs. However, if a Condition III or IV event is analyzed to meet the acceptance criteria of a Condition II event, a Condition III or IV event analysis may bound a given Condition II event in the same category or with similar characteristics.

The Disposition of Events review classifies each event into one of the following categories:

- Event must be reanalyzed.
- Event is bounded by another event.
- Event is bounded by a previous analysis.
- Event is outside the licensing basis of the plant.

5.2.2 Analysis Assumptions

When analyses of the various SRP Chapter 15 events are performed, the analyses will consider the following items:

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5.3 *Biasing of Parameters*

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5.4 *Main Steamline Break (MSLB)*

The methodology described below replaces that described in Reference 4. Detail has been added on how mixing in the reactor pressure vessel is modeled and how event parameters are biased. A description of how the pre-scram portion of the MSLB event is modeled has also been added. S-RELAP5 is used in place of ANF-RELAP for MSLB. Except for these changes, the methodology for MSLB analyses is unchanged from Reference 4.

In a PWR, accidental occurrence of an MSLB, coincident with a negative moderator coefficient and the most reactive control rod stuck in the withdrawn position, can lead to a critical core and a return to power from a previous subcritical state. Analysis of this event is conducted as part of safety analyses required by the NRC for operation of PWR nuclear power plants. Guidelines for NRC review and acceptance of MSLB safety analyses are presented in SRP 15.1.5.

The MSLB event is analyzed to assess the potential for fuel failure from either DNB or FCM. Acceptance criteria allow fuel failure, but require the radiological consequences for an MSLB with the highest worth control rod stuck out of the core and an assumed pre-accident iodine spike to be within 10 CFR Part 100 guideline values. For an MSLB with equilibrium iodine concentrations for continuous full power operation and an assumed accident-initiated iodine spike, the calculated doses must be a small fraction of the 10 CFR Part 100 guideline values.

The FRA-ANP MSLB analytical methodology utilizes S-RELAP5 (Reference 1) for the NSSS calculation, an approved neutronics code for the detailed core neutronics calculation, and XCOBRA-IIIC (Reference 6) for the detailed core thermal-hydraulic calculation.

The use of S-RELAP5, combined with the various assumptions described in Section 5.4.3, provides a conservative simulation of an MSLB accident. The use of a steam-only-out-the-break model, the inclusion of upper head flashing, a point kinetics core model, minimal mixing between the affected and unaffected coolant loops, and the conservative representation of plant systems combine to ensure a very conservative S-RELAP5 model. [

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There is flexibility in the methodology to accommodate vendor and reactor type differences, as well as different approaches to various aspects of MSLB analysis, such as reactivity feedback and mixing within the reactor pressure vessel.

Although the related containment analysis methodology is not part of the methodology described below, the worst case MSLB NSSS calculation may serve as the basis of the mass and energy history sources in the containment calculation for an MSLB transient.

5.4.1 Methodology Overview

The FRA-ANP MSLB methodology is illustrated by the flow diagram in Figure 5.1. The methodology uses S-RELAP5 to calculate the plant transient response to an MSLB, based on a detailed hydraulic model of the reactor coolant and steam systems and a point kinetics model of the core. Fuel failure from either departure from nucleate boiling (DNB) or FCM is assessed, based on the conditions calculated by XCOBRA-IIIC and the neutronics code for the highest powered fuel assemblies.

Core power, core boundary conditions, other primary system conditions, and secondary conditions are computed during the transient with S-RELAP5. At selected points in time during the transient, the power distribution and reactivity are computed with the neutronics code, based on the core power and core boundary conditions from S-RELAP5. At the same points in time, the PWR open lattice (i.e., open channel) core flow distribution is calculated with XCOBRA-IIIC, based on the power distribution from the neutronics code and the core boundary conditions from S-RELAP5.

The MDNBR is determined by using approved correlations such as the XNB (Reference 15), high thermal performance (HTP) (Reference 16), modified Barnett (Reference 17), or Biasi correlation (Reference 18). The potential for fuel failure from DNB is assessed by comparing the calculated MDNBR to the applicable departure from nucleate boiling ratio (DNBR) safety limit.

The peak fuel rod LHGR which occurs during the transient is calculated using the core power from S-RELAP5 and the power distribution from the neutronics code. The potential for fuel failure from FCM is assessed by comparing the calculated peak LHGR to an LHGR limit for FCM, determined with RODEX2 or other approved fuel rod thermal-mechanical computer codes.

The S-RELAP5 calculation includes a sectorized core and allows for flashing in the upper head of the reactor pressure vessel. Asymmetric thermal-hydraulic and related reactivity feedback effects are accounted for with the sectorized core. Upper head flashing capability retards the pressure decay in the reactor coolant system and thus both delays the time of initial delivery and decreases the delivery rate of boron to the core from the Safety Injection System (SIS).

In some cases the complete computation string is not required. For example, in cases where there is no return to power, or where the return to power is extremely small, DNBR and FCM calculations are not necessary.

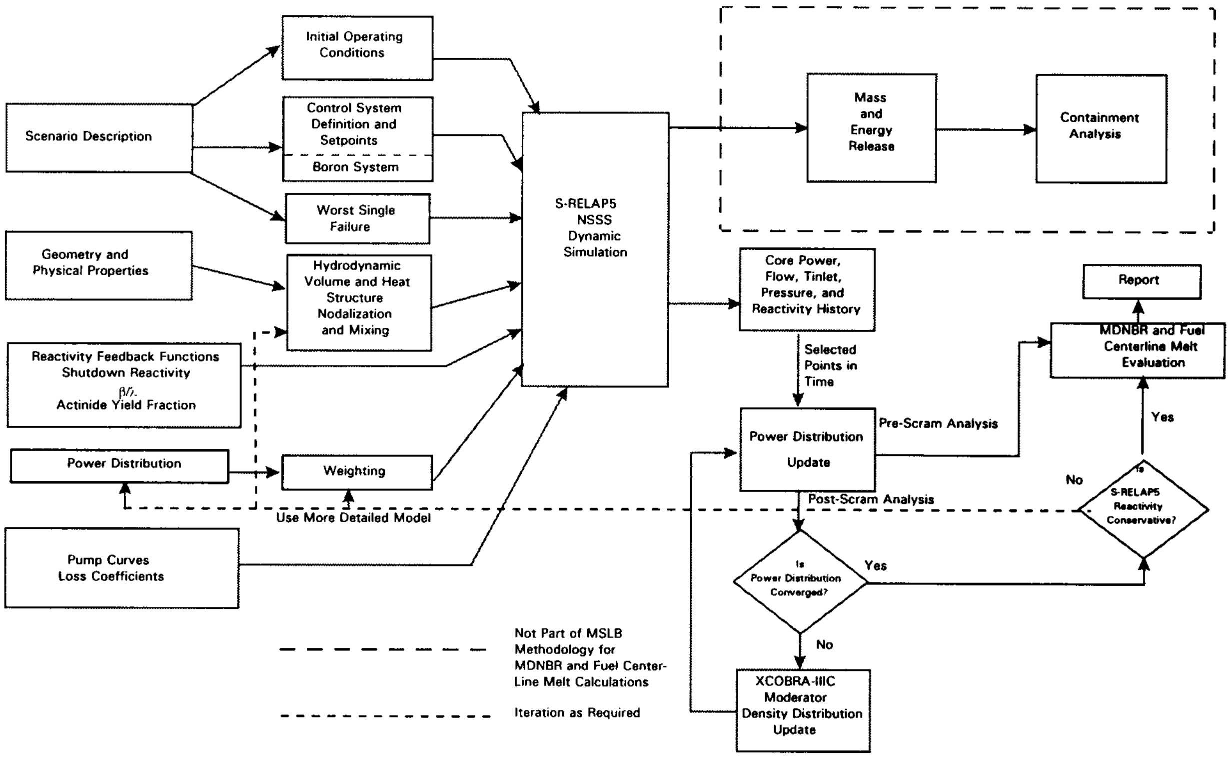

Figure 5.1 FRA-ANP Steamline Break Methodology

5.4.2 Description of Methodology

5.4.2.1 Transient Characteristics

There are many possible MSLB transient scenarios. Factors of importance in determining the consequences of an MSLB include the reactor vendor related differences, the number of loops, the initial operating conditions of the NSSS, availability of offsite power (i.e., natural versus forced circulation of reactor coolant), the worst single failure, the break size and location, the cycle dependent neutronics parameters, and whether or not a stuck rod is assumed in connection with the iodine spiking.

In all large break scenarios, with extended blowdown from one SG, there will be a rapid depressurization and cooldown of the affected SG. This in turn will lead to a rapid cooldown in the reactor coolant loop containing the affected SG and also in the core sector cooled primarily by water entering the core from the cold leg of the affected loop. Other loops and related core sectors will cool at a lesser rate, depending on the various mixing and/or crossflow phenomena present within the reactor pressure vessel, and the time delay before the remaining SGs are isolated from the break with closure of the main steam isolation valves (MSIVs). Due to the cooldown, the reactor system coolant will contract. In the case of a severe steamline break this may cause the pressurizer to empty and the reactor coolant system pressure to decrease rapidly. Water in the reactor pressure vessel upper head may flash if this region is fairly stagnant. Upper head flashing will act to delay the reactor coolant system pressure decay once the saturation pressure of the upper head is reached. This may delay the initiation of borated water injection by the SIS. Higher reactor coolant system back pressure will also result in lower flow from the SIS, further lengthening the time it takes for boron to enter the core.

5.4.2.2 Post-Scram MSLB

The core will be subcritical shortly after initiation of the MSLB, due to a scram at power, a scram at critical HZP, or as a consequence of initiating the transient from subcritical conditions. Shortly after the break, both Doppler and moderator reactivity feedback will be positive at EOC core conditions, due to cooldown of both the fuel and the moderator throughout the entire core.

With the most reactive control rod assumed to be stuck out of the core, the radial neutron flux (and therefore power) distribution will be highly peaked in the region of the stuck control rod.

Under this condition, both positive reactivity feedback effects during cooldown and negative feedback effects during heatup will be dominated by the region around the stuck control rod. If the core sector with the stuck control rod is also the sector being cooled primarily with coolant delivered by the cold leg of the affected loop, positive feedback due to cooldown of the fuel and moderator will be accentuated due to the flux distribution. If criticality is reached and the reactor begins a power excursion, negative Doppler feedback will tend to reduce the core reactivity. In certain regions of the core the moderator feedback may also be negative, due to heatup of the coolant in that region, thereby further reducing the total core reactivity. This is particularly true for the highest power peaked assemblies near the stuck control rod, where, if power levels are high enough, voiding will occur, which will result in significant reductions in total core reactivity. If boron is not injected into the core and if the affected SG does not dry out for an extended period of time into the transient, a quasi-steady-state power level will be reached, as reactivity feedback effects equilibrate and the steam flow rate out the break equilibrates with core power. Eventually, as the affected SG begins to dry out, reactor coolant system temperatures will rise, reactivity feedback effects will reduce power, and a new equilibrium power level will result, with the break steam flow rate equal to the AFW flow rate.

If the boron injected into the core is from a high concentration boric acid tank, then the power excursion will be terminated upon the first pass of borated safety injection water through the core. If the concentrated boric acid tank has been removed, or the boric acid has been removed from the tank, and boron is supplied from the more dilute RWST, then the power excursion will terminate more slowly. Delivery of significant quantities of boron into the core is dependent on (1) the time delay between the SIS signal and the time required to bring the pumps to rated speed, (2) the time delay between actuation of the SIS and decay of the reactor coolant system pressure below the safety injection pump shutoff head, (3) the time delay required to transport the boron from the boron source to the core, and (4) dilution of the boron between the source and the core. Other important factors are the number and characteristics of injection pumps assumed operational.

Main and AFW characteristics also have an impact on the MSLB transient. The higher the feedwater flow rate, the longer the period of flow, and the lower the enthalpy of the feedwater sources—the greater will be the severity of the reactor coolant system cooldown. If the MFW is on, it will be terminated after a short delay following receipt of a main steam isolation signal (MSIS).

Two RCP cases are typically considered in the MSLB analysis. These are (1) offsite power available—with the RCPs operating throughout the transient—and (2) offsite power lost when the transient is initiated—with the RCPs tripped at initiation.

The maximum break size is the most limiting, since it maximizes the rate and extent of cooldown. For plants with integral flow restrictors in the SG heads, the worst break location is either immediately downstream of the flow restrictor or between the flow meter and the MSIV. For plants without integral flow restrictors, the worst break location is between the SG head and the flow meter. In order to bound radiological consequences, break locations both inside and outside containment are considered.

The worst single failure is determined on a plant specific basis. Typically, the worst single failure for the Post-Scram MSLB analysis has been found to be the failure of a single safety injection pump. Failure of an MSIV to close will have no effect on the worst transient scenario, due to the location of the break.

5.4.2.3 Pre-Scram MSLB

The pre-scram phase of an MSLB event can challenge acceptance criteria due to harsh containment conditions and power decalibration. Power decalibration is caused by density-induced changes in the reactor pressure vessel downcomer shadowing the power-range excore detectors during heatup or cooldown transients. The nuclear power level indicated by the excore detectors is lower than the actual reactor power level when the coolant entering the reactor pressure vessel is cooler than the normal full power temperature (and higher when the inlet coolant is warmer than the normal full power temperature). This effect is taken into account in the modeling of any power-dependent reactor trips credited in the analysis.

A break located downstream of a main steamline check valve will allow steam to flow to the break from all SGs prior to MSIV closure and will be referred to as a "symmetric" break. An "asymmetric" break is located upstream of a check valve and allows steam to flow to the break from the upstream SG only (because the check valve precludes backflow to the break from the downstream SGs).

The worst single failure for an asymmetric break is the failure of one nuclear instrumentation (NI) channel. There are typically four channels of NI, using excore detectors located around the reactor which provide power indication to the RPS. Since the power in the affected region will always be higher than in the unaffected region, the NI channel closest to the affected sector is conservatively assumed to be failed. If plant operation allows one NI channel to be out-of-service, it is selected from the remaining NI channels as that closest to the affected region. The response of the remaining excore detectors are conservatively modeled, and provide the signal for initiation of a reactor trip on an over-power condition.

There is no single failure which could worsen the event consequences for a symmetric break. A full range of break sizes, up to a double-ended guillotine break of a main steamline, is considered. The moderator temperature coefficient (MTC) is varied over each break size analyzed to sufficiently bound the timing effects between a Low SG Pressure trip (dependent on break size) and an over-power trip (dependant on the primary side cooldown and the value of the MTC).

5.4.3 S-RELAP5 NSSS Model

5.4.3.1 General Overview

The general input requirements for S-RELAP5 include a description for the primary and secondary systems in terms of hydrodynamic volumes and the structures which interact with these volumes from a heat transfer standpoint. A typical nodalization is provided in Figure 6.2, Figure 6.4 and Figure 6.5. (This nodalization is specific to the sample problem and is meant to illustrate how the sample problem was modeled. In general, nodalizations differ for each specific application due to differences in reactor design.)

Reactor kinetics, power distributions, and reactivity feedback weighting are all required as part of the input. Pump curves and hydraulic loss coefficients are also part of the input. Additional input information required to describe the transient scenario of interest to MSLB analysis can be specified through the control system model integral to S-RELAP5.

Items of particular importance to the S-RELAP5 NSSS model are discussed in detail in the following sections.

5.4.3.2 Sectorized Core and Other Reactor Pressure Vessel Components

In order to simulate the asymmetric thermal-hydraulic and reactivity feedback effects that can occur during an MSLB transient, the core is divided into two sectors. The division is made between the core sector which is directly impacted by the affected SG and the sector which is not directly impacted. The core sector and loop which are directly impacted by the break will be termed the "affected" sector and loop. The remainder of the core and the remaining loops will be termed "unaffected."

This division of the core into two sectors, or parallel flow channels, for a CE plant is shown in Figure 5.2. [

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The upper and lower plenums and other reactor pressure vessel components are divided similarly. To further refine the prediction of core thermal-hydraulic behavior, a [

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The division of the core into two sectors for a three-loop Westinghouse plant is shown in Figure 5.3. [

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5.4.3.3 Mixing Between Sectors

During an MSLB transient, mixing between the parallel affected and unaffected sectors within the reactor pressure vessel will occur in the lower plenum, the core, and the upper plenum—due to lateral momentum imbalances, turbulence or eddy mixing, and the relative angular positions of the cold legs to the hot legs. Some mixing will also occur in the downcomer. Mixing and/or crossflow will act to reduce the positive reactivity feedback effects—due to a reduced rate and magnitude of cooldown of the affected loop.

In the FRA-ANP methodology, [

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5.4.3.4 Power Distribution, Reactivity Feedback, and Feedback Weighting

The FRA-ANP methodology [

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One approach is to utilize a [

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5.4.3.5 Upper Head Flashing

Flashing in the upper head is modeled by using [

]. The more stagnant the region is, the more flashing will occur—which will subsequently retard the pressure decay within the reactor coolant system.

5.4.3.6 Initial Power Level and Offsite Power Availability

The initial power level and availability of offsite power are two major factors in determining the most limiting MSLB transient.

The post-scram phase of the MSLB event typically considers two initial power levels, HFP and HZP. The pre-scram phase of the event is analyzed only at HFP conditions because the initial margin to the SAFDLs is the smallest.

Two offsite power availability assumptions are typically considered. These are offsite power available for operation of RCPs and safety injection pumps, and offsite power not available for operation of these pumps. In this latter case, the RCPs are assumed to be tripped at initiation of the MSLB, and a delay time to start diesel generators for operation of the SIS pumps is included in the analysis.

5.4.3.7 Safety Injection, Feedwater, and MSIVs

The SIS, feedwater system, and MSIVs have important impacts on the Post-Scram MSLB analysis.

The SIS delivers boron to the core and may be one of the means for terminating the MSLB power excursion. The main factor of importance is the time delay from initiation of the break to the time when boron of adequate concentration to terminate the power excursion is delivered to the core. This time is determined by the concentration of the boron source, the flow delivery characteristics of the SIS versus reactor coolant system pressure, the number of SIS pumps available, the delay time required to bring the pumps to speed after receipt of the actuation signal, the SIS trip setpoint, the piping volume between the boron source and the reactor coolant system injection location, and the dilution that occurs between the source and the injection point.

The feedwater system consists of the MFW system utilized during normal operation and an AFW system of much reduced capacity for use when the main system is not available. For the post-scram phase of the MSLB event, the higher the total feedwater flow is and the lower the inlet enthalpy is, the greater the cooldown and subsequent return to power will be. Upper bounds on the flow and lower bounds on the enthalpy of the main and AFW are used during the transient calculation.

The primary consideration for the MSIVs is the period of time the unaffected SGs are able to blow down before the MSIVs close. Closure will occur after a short time delay following a closure signal.

5.4.3.8 Containment Model

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The Post-Scram MSLB analysis does not credit the containment high pressure trip, therefore a containment model is not necessary.

5.4.3.9 Input Parameter Biasing

Following is a description of the parameters to be biased with this methodology, including in which code it is to be biased. The biases discussed below are applicable to the pre-scram and post-scram phases of the MSLB event unless otherwise noted.

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5.4.4 Core Neutronics Model

5.4.4.1 Post-Scram MSLB

An NRC-approved neutronics code is used to calculate radial and axial power distributions and total core reactivity for a given core power level and moderator density distribution. Input to the neutronics code includes the core power level from S-RELAP5, the core coolant density distribution from XCOBRA-IIIC, and a reference set of nuclear cross sections appropriate for the core of interest, plus the conditions under which the cross sections were generated. For these imposed conditions, the code determines the distribution of power within the core, both axially and radially, [

] The code also determines the resultant core reactivity under the imposed core conditions.

The nodal moderator densities from XCOBRA-IIIC are transferred into the neutronics code, and iterative neutronics code/XCOBRA-IIIC calculations are performed until the power distribution converges.

5.4.4.2 Pre-Scram MSLB

For asymmetric cases, the neutronics code is used to calculate the case-specific radial power distribution. [

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For symmetric cases, the thermal-hydraulic conditions and power distributions of the affected and unaffected sectors of the core are essentially identical. [

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5.4.5 Core Thermal-Hydraulic Model

Input to the XCOBRA-IIIC core thermal-hydraulic analysis consists of assembly geometry and hydraulic descriptive information, including information regarding which assemblies are adjacent to each other. The S-RELAP5 calculated core outlet pressure, core inlet flow distribution, core inlet temperature distribution, and core-average LHGR, along with the neutronics code calculated axial and radial power distributions, must be input. [

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5.4.6 Reactivity Comparison

5.4.6.1 Post-Scram MSLB

A three-channel S-RELAP5 core model can only accommodate relatively simple radial and axial power distributions, associated reactivity feedback, and feedback weighting models. This tends to result in simple and conservative representations of highly complex neutronics and thermal-hydraulic phenomena. The inherent conservatisms are demonstrated by comparing the reactivity change calculated with S-RELAP5 against that calculated with the neutronics code at points in time of particular interest. An important point of interest is the time at which MDNBR occurs.

The reactivity change calculated with the neutronics code is increased to account for an MTC bias adjustment and, for HFP cases, a scram curve adjustment. [

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5.4.6.2 Pre-Scram MSLB

The Pre-Scram MSLB analysis does not require a reactivity comparison due to the calculations performed to find the most limiting combination of MSLB size and MTC. These calculations, in essence, cover a spectrum of reactivity insertions by varying the break size and MTC. Additionally, the neutronic response prior to scram is not as complex as that in the post-scram portion of the event.

5.4.7 MDNBR and FCM Analysis

The end result of an MSLB MDNBR and FCM analysis is to determine how many, if any, fuel rods penetrate the DNBR safety limit and/or the FCM limit. If the MDNBR is below the DNBR limit or if the peak LHGR is above the FCM LHGR limit, then the total number of fuel rods expected to fail is determined. The methods utilized as part of the MSLB methodology to determine the MDNBR and peak LHGR are discussed in the following sections.

5.4.7.1 XCOBRA-IIIC Subchannel DNBR Evaluation Method

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5.4.7.2 Alternate DNBR Evaluation Method

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5.4.7.3 Peak LHGR Evaluation Method

5.4.7.3.1 Post-Scram MSLB

The peak fuel rod LHGR which occurs after scram is calculated using the peak post-scram core power from S-RELAP5 and the corresponding power distribution from the neutronics code. [

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5.4.7.3.2 Pre-Scram MSLB

The peak fuel rod LHGR which occurs prior to scram is calculated using the peak pre-scram core power from S-RELAP5. Asymmetric cases use the corresponding power distribution from the neutronics code while symmetric cases use a conservatively limiting axial power profile and radial power distribution. [

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Figure 5.2 Core Model for CE Plant

Figure 5.3 Core Model for Three-Loop Westinghouse Plant

Figure 5.4 [ ] for Three-Loop Westinghouse Plant

5.5 *Steam Generator Tube Rupture (SGTR)*

This event is generally categorized as a Condition IV event. The acceptance criteria are given in Section 5.1. The system analysis provides the boundary conditions for use in the evaluation of radiological consequences. The system response is evaluated using the tools and methods applied for other non-LOCA events with appropriately bounding assumptions.

Event Description

The SGTR event is Event 15.6.3 of the SRP and is initiated by a break of a single steam generator tube. Coolant from the RCS begins to escape through the break, driven by the pressure differential between the RCS and the SG secondary side, increasing the inventory and pressure in the SGs.

As the break flow begins to de-pressurize the RCS, the charging pumps activate in order to make-up the lost inventory. If the RCS inventory and pressure are stabilized via the charging pumps, no reactor trip will occur. However, if the break flow exceeds the capacity of the pumps, the RCS pressure and inventory will continue to decrease resulting in a reactor trip on a low-RCS-pressure signal. Following the reactor trip, the turbine will trip and, in the case where offsite power is lost, the coolant pumps will coast down and make-up flow will terminate. If offsite power is available, a fast transfer to the offsite power will keep the pumps running and the make-up flow available.

The loss of offsite power results in the loss of condenser vacuum and the steam dump valves are closed to protect the condenser. The continued mass and energy transfer between the primary and secondary side results in a rapid increase in SG pressure and discharge to the atmosphere via the Main Steam Safety Valves (MSSVs) and Atmospheric Steam Dump Valves (ADVs).

As the RCS pressure continues to decrease, a low pressurizer pressure signal activates the SIS. The emergency diesels start and HPSI flow begins. For some plants, the HPSI pumps have a very high delivery head which may result in a rapid pressurization of the reactor coolant system. In this case, a high break flow rate is maintained leading to a more rapid filling of the SG. This may lead to liquid in the steamlines and MSSVs. Liquid in the steamlines may cause the MSSVs to fail open and potentially damage the steam piping.

The event can proceed in several directions from this point and is highly dependent on the emergency operating procedures (EOPs) for the plant. The ECCS tends to exacerbate the releases for this event by maintaining the pressure in the RCS and increasing the flow to the SGs. The HPSI and AFW flows may be secured, ADVs may be opened to de-pressurize the SGs and the RCS, and the pressure operated relief valves (PORVs) may be opened to bring the RCS pressure down and stop flow through the break. The operators will take a series of actions to regain control of the plant systems and to bring the RCS to a condition allowing for initiation of the residual heat removal (RHR) system. To regain control of the plant systems, the operators must first identify the event. The identification is based on a high secondary side activity in conjunction with a high water level reading for the affected steam generator.

The depressurization of the RCS does not generally present as great a challenge to fuel failure as the inadvertent opening of a PORV and the potential for fuel failure is quite low.

The key elements of this event are the primary-to-secondary pressure differential and the break flow path. The temperature of the RCS usually sets the pressure, which determines the flow. The temperature is established by the power in the primary and the secondary pressure. For plants with very high head HPSI pumps, the reactor coolant pressure can be dependent on the HPSI flow.

Events Analyzed

The event can be initiated from HZP or HFP conditions. Due to the lack of decay heat load for the HZP case, it may not be the limiting inventory release case. However, given the technical specification limits on activity, the HZP transient may lead to more limiting radiological consequences. Therefore, unless the HZP case can be dispositioned, it will be analyzed.

The loss of offsite power must be addressed due to the impact on condenser availability for the steam dump bypass system. The lack of such availability may lead to more limiting radiological consequences.

The potential for overfill of the secondary side exists. For some PWR designs, the make-up flow and HPSI flow are provided by the same pump. These designs have the highest potential for over-filling the secondary side and introducing liquid into the steamlines. The SGTR event analysis will address overfill for these designs which utilize very high head HPSI pumps.

Analysis Method

The system response is modeled using S-RELAP5, including cooldown to RHR operation. The phenomena that determine the release to the atmosphere and the challenge to fuel integrity are similar to those encountered in other non-LOCA events, with the exception of the break flow. The break flow is addressed using models similar to those used for small break LOCAs. The de-pressurization of the RCS is bounded by that experienced by Event 15.6.1. The SG pressurization transient is bounded by that modeled for Event 15.2.1. The SG level increase is bounded by Event 15.2.1 and by Event 15.1.2 (the Increase in Feedwater Flow).

The early portion of the transient (prior to operator action) is modeled using the ESFAS and RPS responses using a slightly modified S-RELAP5 model for Chapter 15 non-LOCA events. The modifications include the following:

- Added SG Tube – The normal non-LOCA model lumps all of the tubes in a SG together. The SGTR model has one SG tube modeled explicitly, with the remainder lumped. The rupture model is a double-ended guillotine break in this tube just above the tube sheet. Critical flow is modeled using the Moody model which provides a conservative model for choked flow and is used in FRA-ANP's LOCA and MSLB methodologies.
- HPSI Flow – HPSI models are added. The normal non-LOCA model does not include HPSI pumps. All pumps are assumed to be available and to operate at design capacity. This produces conservatively high flows.
- Upper Head Flashing - When a loss of offsite power is assumed, the cooldown of the reactor coolant system is based on natural circulation. If voiding occurs in any of the loops, it can affect the natural circulation flows. Voiding is strongly affected by the system pressure. Heat structures are added to model the metal masses in the upper head region of the reactor vessel. These modifications are based on the MSLB model and are made to increase the accuracy of the calculation of the pressure in the upper head. The boron injected with the HPSI flow is modeled for each volume. The pressure is also important for boron injection, since it determines the HPSI flow, which introduces borated water into the RCS.
- Control System - Since this event requires operator intervention, the S-RELAP5 input model for non-LOCA events is further modified to properly simulate operator actions consistent with the plant-specific EOPs. Generally, operation of the MSSVs, ADVs and PORVs are modeled to cool the plant down. Also, isolating the SGs and terminating HPSI flow is modeled, as appropriate.

Bounding Input

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5.6 *CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)*

The analysis of the boron dilution event does not use the system code S-RELAP5. The methodology for performing Boron Dilution analyses is described in this section.

Identification of Causes and Event Description

One means of positive reactivity insertion to the core is the addition of unborated, primary grade coolant from the demineralized and reactor makeup coolant systems. This coolant is introduced to the RCS through the reactor charging/makeup portion of the CVCS.

The most limiting event resulting in an inadvertent boron dilution is typically a malfunction of the CVCS valve which causes pure coolant to be delivered to the RCS by all available charging/makeup pumps. The CVCS and makeup coolant systems are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which will allow sufficient time for automatic or operator response to terminate the dilution. Typically, the sources of dilution may be terminated by closing isolation valves in the CVCS. The lost shutdown margin may be regained by the opening of isolation valves to the RWST, thus allowing the addition of highly borated coolant to the RCS.

The acceptance criteria includes SRP requirements in Section 5.1 for a Condition II event. If operator action is required to terminate the transient, the minimum time intervals to respond are:

- 30 minutes (during refueling); and
- 15 minutes (for all other modes).

These times apply between either (a) the time when an alarm announces an unplanned moderator dilution, or (b) the initiation of the dilution, and the time of loss-of-shutdown margin.

The choice of (a) or (b) is determined from the plant licensing basis.

Analysis Method

To cover all modes of plant operation, boron dilution during modes 6 through 1 (Refueling, Cold Shutdown, Hot Shutdown, Hot Standby, Start-up, and Power operation) are considered. The purpose of the analysis is to demonstrate that sufficient time exists for termination by the operator before the shutdown margin is lost. Conservative values for parameters are used, i.e., high RCS critical boron concentration, minimum shutdown margin, minimum RCS volume, and maximum unborated water charging rate. These assumptions result in conservative determinations of the time available for operator or system response after initiation of a dilution transient.

There are two models that can be used to represent the mixing of the borated mass in the RCS and the dilution flow: (a) instantaneous mixing model; and, (b) dilution front model. For plant modes in which one or more reactor coolant pumps are operating, the assumption of complete (i.e., instantaneous) mixing of boron with water in the RCS is appropriate and the instantaneous mixing model is used. For modes where the shutdown cooling system (SDCS) is in operation, flow rates may be insufficient to assure complete mixing of the reactor coolant system and the dilution front approach is applied.

The instantaneous mixing model assumes complete and instantaneous mixing of boron within the applicable mixing volume in the RCS. The boron concentration vs. time, $C_{RCS}(t)$, and time to dilute the RCS boron concentration from the shutdown to critical states, $t_{Critical}$, are:

$$C_{RCS}(t) = C_{RCS}(0) \cdot \exp\left(-\frac{W_{Charge}}{M_T} \cdot t\right), \quad \text{ppm}$$

$$t_{Critical} = \left(\frac{M_T}{W_{Charge}}\right) \cdot \ln\left[\frac{C_{RCS}(0)}{C_{RCS}(\text{Critical})}\right], \quad \text{minutes}$$

where:

$C_{RCS}(0)$	=	initial (shutdown) boron concentration (ppm)
$C_{RCS}(\text{Critical})$	=	critical boron concentration (ppm)
W_{Charge}	=	charging (dilution) mass flow rate (lb_m/m)
M_T	=	fluid mass of the mixing volume (lb_m)

In the dilution front model, the dilution is viewed as a series of 'dilution fronts' progressing through the reactor coolant system. A dilution front tracking model is used to calculate the RCS boron concentration vs. time and the time to reach criticality. The model is based on the following assumptions:

- The charging flow mixes with the RCS flow and results in a reduced boron concentration at the mixing location;
- The SDCS and dilution flows are fully mixed in the lower plenum prior to entrance into the core;
- The diluted mixture transit time to the bottom of the core is based on the flow volume (between the mixing location and the bottom of the core) and the flow rates of both the charging and RCS flow; and,
- If the diluted boron concentration for any front is higher than the critical concentration, the diluted mixture must sweep through the entire mixing volume and pass by the dilution location another time.

This dilution scenario continues until the RCS boron concentration in the core is diluted below the critical concentration. The time-to-criticality is the number of complete transit times required to achieve a core boron concentration less than the critical value plus the transit time from the mixing location to the bottom of the core.

The general equations relating the N^{th} front boron concentration, C_N, and front transit time to reach the bottom of the core, t_N, are given by:

$$C_N = C_0 \cdot \left(\frac{W_{SDCS}}{W_{SDCS} + W_{Charge}} \right)^N, \quad \text{ppm}$$

and

$$t_N = \left(\frac{M_{CL}}{W_{SDCS} + W_{Charge}} \right) + \left(\frac{M_T}{W_{SDCS} + W_{Charge}} \right) \cdot (N-1), \quad \text{minutes}$$

where

C_0 = initial (shutdown) boron concentration (ppm)

W_{SDCS} = SDCS mass flow rate (lb_m/m)

W_{Charge} = charging (dilution) mass flow rate (lb_m/m)

M_T = fluid mass of the mixing volume (lb_m)

M_{CL} = mass in the volume between the mixing location and core inlet that contains the diluted mixture (lb_m)

The initial boron concentration is determined according to the Technical Specification or COLR SDM requirement based on the assumption that the most reactive rod is stuck out of the core. The critical boron concentration, however, may be determined assuming that all rods are inserted for Modes 4 and 5. Following a reactor shutdown, plant procedures typically require the operator to verify the control rods are fully inserted and to increase the RCS boron concentration requirements to compensate for any control rod not fully inserted. This action would occur prior to a controlled cooldown to Modes 4 and 5, and provides assurance that the reactor is subcritical by the shutdown margin requirement plus the worth of a stuck rod. If the plant has procedures that increase the shutdown boron requirements to compensate for a stuck rod, then the critical boron concentration is determined assuming that all rods are inserted for Modes 4 and 5. Otherwise, the critical boron concentration is determined using the assumption that the most reactive rod is stuck out of the core.

For Mode 6, the initial and critical boron concentrations are based on the SDM requirements.

Refueling (Mode 6)

An uncontrolled boron dilution transient during this mode of operation is typically prevented by administrative controls that isolate the RCS from the potential source of unborated water. If an analysis for Mode 6 is required, the following conditions are typically assumed for inadvertent boron dilution:

a. Dilution flow is based on a maximum value relative to the capacity of the normally operable charging/makeup pumps during this mode;

b. The SDCS flow rates are limited by the range of normal operation for this mode;

c. A mixing volume is defined consistent with the minimum active volume of the RCS (e.g., water level drained to mid-nozzle in the vessel), one train of SDCS, and the volume of CVCS that is in the circulation flow path; and,

d. Initial and critical boron concentrations are based on values corresponding to the shutdown margin required by Technical Specifications for this mode.

Cold Shutdown (Mode 5)

The following conditions are typically assumed for inadvertent boron dilution while in this operation mode:

a. Dilution flow is based on a maximum value relative to the capacity of the normally operable charging/makeup pumps during this mode;

b. The SDCS flow rates are limited by the range of normal operation for this mode;

c. For SDCS operation, a mixing volume is defined consistent with the minimum active volume of the RCS (e.g., water level drained to mid-nozzle in the vessel), one train of SDCS, and the volume of CVCS that is in the circulation flow path;

d. For RCP operation, a mixing volume is used consistent with the minimum active volume of the RCS minus the pressurizer volume;

e. If the plant has procedures that increase the shutdown boron requirements to compensate for an identified stuck rod, then the critical boron concentration is based on all control rods inserted in the reactor core at the time the event initiates; otherwise, the critical boron concentration is based on all rods inserted with the most reactive rod stuck out of the core; and,

f. Initial boron concentration is based on the value corresponding to the shutdown margin required by Technical Specifications for this mode.

Hot Shutdown (Mode 4)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

a. Dilution flow is based on a maximum value relative to the capacity of the normally operable charging/makeup pumps during this mode;

b. The SDCS flow rates are limited by the range of normal operation for this mode;

c. For SDCS operation, a mixing volume is defined as the minimum active volume of the RCS, the volume of one train of SDCS, and the volume of CVCS that is in the circulation flow path;

d. For RCP operation, a mixing volume is used consistent with the minimum active volume of the RCS minus the pressurizer volume;

e. If the plant has procedures that increase the shutdown boron requirements to compensate for an identified stuck rod, then the critical boron concentration is based on all control rods inserted in the reactor core at the time the event initiates; otherwise, the critical boron concentration is based on all rods inserted with the most reactive rod stuck out of the core; and

f. Initial boron concentration is based on the value corresponding to the shutdown margin required by Technical Specifications for this mode.

Hot Standby (Mode 3)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

a. Dilution flow is based on a maximum value relative to the capacity of the normally operable charging/makeup pumps during this mode;

b. A mixing volume is used consistent with the minimum active volume of the RCS minus the pressurizer volume;

c. The RCS is filled and vented and at least one RCP is running; and,

d. Initial and critical boron concentrations are based on values corresponding to the shutdown margin required by Technical Specifications for this mode.

Startup (Mode 2)

During this mode of operation, the plant control systems are assumed in manual mode. The Technical Specifications typically require that all RCPs be operating. Other conditions assumed are:

a. Dilution flow is based on a maximum value relative to the capacity of the normally operable charging/makeup pumps during this mode;

b. A mixing volume is used consistent with the minimum active volume of the RCS minus the pressurizer volume; and,

c. Initial and critical boron concentrations are based on values corresponding to the shutdown margin required by Technical Specifications for this mode.

This mode of operation is typically a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a very high awareness of the plant status. For a normal approach to criticality, the operator may manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The plant Technical Specifications typically require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus assuring that the reactor does not go critical with the control rods below the insertion limits.

In the event of an unplanned dilution during power escalation while in the Startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power and will activate a power-related trip. There must be sufficient time to prevent return to criticality, as defined in the plant licensing basis.

Power Operation (Mode 1)

Since the slow power and temperature rise will cause a decrease in the DNBR, the event may result in a challenge to the SAFDLs. The boron dilution transient is to be bounded by the minimum reactivity insertion rate assumed for an Uncontrolled Rod Withdrawal at Power.

The erosion of shutdown margin following trip in Mode 1 is bounded by the Mode 2 Boron Dilution analysis.

5.7 *Inadvertent Loading and Operation of a Fuel Assembly in an Improper Location (Misloaded Assembly)*

The misloaded assembly event is included here as part of the non-LOCA transient methodology. It does not require a thermal-hydraulic system analysis. The previously approved misloaded assembly methodology, Reference 19, is the basis for this methodology description.

Event Description

The misloaded assembly event is characterized by loading one or more fuel assemblies into improper locations and, where physically possible, with incorrect orientation. These fuel loading errors can result in changes in the core power distribution and increases in local power density (LPD) which may challenge the core safety limits.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and is loaded in accordance with a specified core loading pattern. Following core loading, the identification number of each assembly loaded in the core is checked against the desired core loading pattern.

Additional safeguards against fuel loading errors include startup physics test measurements, excore instrumentation measurements, and incore instrumentation measurements. Although any of these measurements could detect power distribution anomalies, the incore instrumentation is used to perform an initial low-power measurement of the core power distribution specifically to ensure that the core is properly loaded.

A fuel loading error changes the core power distribution by an amount proportional to the change in reactivity of the misloaded assembly. Large deviations in the measured power distribution relative to the calculated power distribution are readily detectable at the initial low-power measurement. However, small deviations between the measured and calculated power distributions may go undetected, resulting in full-power operation with the misloaded core. The most limiting misloaded configuration would be one that is undetectable and results in the highest core power peaking during the operating cycle.

The primary concerns with this event are the penetration of the DNB fuel design limit and violation of the FCM criterion.

Analysis Method

The standard FRA-ANP neutronics methodology is used to model several misloaded core scenarios. The three-dimensional steady-state core power distribution is calculated at the conditions of the initial low-power incore measurement for the correctly loaded core and for each misloaded core case. The power distribution from each misloaded case is used to represent measured data, and the power distribution for the correctly loaded core is used to represent calculated data. For each misloaded case analyzed, deviations between the measured and calculated data at incore detector locations and deviations between measured data in radially symmetric incore detector locations are evaluated. If the deviations exceed criteria used in plant procedures for detecting misloads, the misload is assumed to be detectable.

A spectrum of misloaded core cases is analyzed. Each misload scenario assumes that the core locations of two assemblies are swapped. These cases represent the misloading of assemblies into core locations which are designated to be occupied by exposed or fresh fuel with different reactivity characteristics.

Since the Technical Specification typically requires that the minimum fraction of incore detectors operable during the initial low-power incore measurement is 75 percent, each incore detector has at least a 75 percent probability of being operable during the measurement. Based on this probability, the detectors which are required to detect the misload are assumed to be operable.

For those cases which are undetectable at the initial low-power measurement, the cycle is depleted at nominal full-power conditions with the control rods withdrawn. The power distribution at each exposure may be used to detect the misloaded core consistent with plant procedures. The depletions provide calculated power peaking factors for those exposures at which the misload remains undetected. The resultant power peaking distributions are examined to determine if the Technical Specification power peaking values are exceeded. If the power peaking values for the misloaded core are calculated to not exceed Technical Specification limits, no further evaluation is necessary, because the DNB fuel design limit and FCM criterion will not be exceeded.

If the event calculations indicate core power peaking limits would be exceeded, additional analyses become necessary. These analyses include applying the approved CHF correlation to obtain the MDNBR and calculating the steady-state peak LHGR to determine if the FCM limit is violated. Conservative values of local and assembly power distributions are input into the DNB and the FCM calculations if Technical Specification limits are violated. The DNBR and FCM calculations are performed at rated power conditions. If either DNBR or the FCM limit is penetrated, a fuel failure assessment is necessary to determine the radiological consequences of the event. The radiological consequences must be less than 10 percent of 10 CFR 100 limits.

5.8 *Control Rod Ejection*

Control Rod Ejection is designated event number 15.4.8 in the SRP. The event is postulated to be caused by mechanical failure of a control rod drive mechanism pressure housing resulting in rapid ejection of the control rod and drive shaft. The Control Rod Ejection event is characterized by positive reactivity insertion in conjunction with an increase in radial power peaking. The event is mitigated by Doppler reactivity feedback from increased fuel temperature. The transient is terminated by either the high flux trip on Westinghouse type PWRs or by the variable high power (VHP) trip on CE PWRs. The event is a very fast reactivity transient. The scram has no effect on the initial peak rise in power. The scram timing does however affect the fuel temperatures and the rod heat fluxes.

Guidance for analysis of this event is provided in Regulatory Guide 1.77 (Reference 20). The acceptance criteria for the Control Rod Ejection event are:

1. The radial average pellet enthalpy at the hot spot must be less than 280 cal/g.
2. The maximum reactor pressure during any portion of the transient must be less than the value that will cause stresses to exceed emergency condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.
3. Fuel failure from DNB or FCM will be limited to keep off-site dose consequences well within the guidelines of 10 CFR Part 100, namely 25 percent of 10 CFR 100 limits.

Reference 21 describes the approved methodology for evaluating criterion 1. The overall system response and fuel centerline temperature for the Control Rod Ejection event is calculated with S-RELAP5. XCOBRA-IIIC is used to obtain the predicted MDNBR (Reference 6). If FCM and/or DNB are predicted, the percentage of fuel failures is computed as input for a radiological assessment.

Four cases are considered: HFP and HZP, each evaluated for BOC and EOC conditions. Key parameters biased to ensure a bounding calculation of the impact of control rod ejection are:

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5.9 *Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment*

This event is initiated by an outside-containment rupture of a small line connected to the RCS. The flow of reactor coolant out the rupture releases activity. The event is a Condition III event, and the acceptance criteria are presented in Section 5.1. This event does not require a system model such as S-RELAP5 to evaluate the radiological consequences. FRA-ANP will evaluate the event using the following calculational process:

- Identify the small lines postulated to fail. These lines are separated into two categories: those with isolation valves inside and outside containment and those with only isolation valves outside containment. With a single failure of an isolation valve, the former will blow down to the environment until the other isolation valve is closed. With the latter, the line will blow down until the reactor coolant system is depressurized.
- Choked flow at the break, based on the reactor coolant pressure, is assumed for all cases.
- The flashing fraction downstream of the break is used to model the amount of activity becoming airborne.

A separate radiological analysis would be performed.

6.0 Sample SRP Transients

A selected set of sample SRP (Reference 2) non-LOCA transients has been analyzed to demonstrate the adequacy of the non-LOCA transient methodology. The analyses have been performed for a CE 2x4 plant.

The nodalizations for the reactor vessel, reactor coolant system piping, and SG secondary side are shown in Figure 6.1 to Figure 6.3 and the nominal initial conditions for the sample problems are given in Table 6.1. The reactor coolant system piping includes two SGs and the four pumps in four cold legs. Pressurizer and HPSI systems are also included.

The MSLB transient requires modified nodalizations for the vessel and SG secondary side. These are shown in Figure 6.4 and Figure 6.5. The reactor vessel nodalization features a sectored core, containing an affected sector and an unaffected sector. Within the affected sector is a stuck rod region. The secondary side of the SG is a simplified model, featuring a steam-only junction, consistent with the methodology in Section 5.4.

Seven transients were analyzed for a CE 2x4 PWR plant. These seven transients were:

- Pre-Scram MSLB (SRP 15.1.5)
- Post-Scram MSLB (SRP 15.1.5)
- LOEL/TT (SRP 15.2.1 and 15.2.2)
- LONF (SRP 15.2.7)
- LOCF (SRP 15.3.1)
- UCBW at Power (SRP 15.4.2)
- SGTR (SRP 15.6.3)

These seven transients were chosen to exercise both the primary and secondary systems in the plant input model. The results presented in the following sections demonstrate the adequacy of the developed methodology. Note, this sample problem is new and is not the same as the sample problem presented in Reference 3.

Table 6.1 Sample Problem Initial Conditions

System Parameter	HFP Value
Core Power (MW)	2700
Primary Pressure (psia)	2250[a]
Pressurizer Level (% of span)	65.6
Cold Leg Temperature (°F)	548
Primary Flow Rate per Loop (lb_m/s)	21,320
Secondary Pressure (psia)	868
Total SG Mass (lb_m) per SG	130,000
Steam Flow (lb_m/s) per SG	1646
MFW Temperature (°F)	435

[a] In SGTR, the pressure was 2300 psia, consistent with the methodology.

Figure 6.1 Sample Problem Vessel Nodalization

Figure 6.2 Sample Problem Reactor Coolant System Piping Nodalization

Figure 6.3 Sample Problem SG and Secondary Nodalization

Figure 6.4 Sample Problem Vessel Nodalization for MSLB

Figure 6.5 Sample Problem Steam Generator and Secondary Nodalization for MSLB

6.1 *Pre-Scram Main Steamline Break (MSLB)*

Event Description

The limiting pre-scram MSLB event for the sample problem is initiated with a break in a main steamline outside containment with the reactor operating at HFP conditions. Coincident with the break, the turbine control valves open fully. The increased steam flow and consequent secondary depressurization lead to a power-cooling mismatch between the heat generated in the core and that being extracted in the SGs. Due to the break location, both SGs are equally affected so that the cooldown transient is essentially symmetric, i.e., all cold legs, all regions of the core and both hot legs are affected in the same manner. Power decalibration results from density-induced changes in the downcomer shadowing of the power-range excore detectors so that lower than actual power is indicated. If the MTC is negative, the cooldown of the reactor system coolant would cause an insertion of positive reactivity and this, coupled with the delayed trip due to power decalibration, would lead to an erosion of the thermal margin.

Definition of Events Analyzed

This event is predominantly an increase in steam flow event with the potential for a more pronounced power level increase. At full power, the margin to the SAFDLs is the smallest. Therefore, the event initiated from full power conditions will bound the event initiated from lower power levels.

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Analysis Results

The MDNBR for this event occurred for a symmetric steamline break outside the containment with an area of 4.0 ft^2 and a -16 pcm/°F MTC. For this break location, both SGs were affected so that the cooldown was maximized. [

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The response of key system variables is given in Figure 6.6 to Figure 6.15. For comparison purposes, the predictions of ANF-RELAP are included with the S-RELAP5 results. The sequence of events is given in Table 6.2. [

] and resulted in an XCOBRA-IIIC calculated MDNBR of 1.27 with an applicable safety limit of 1.164. The peak fuel centerline temperature calculated with S-RELAP5 was 4002°F, and the applicable fuel melting temperature is 4967°F.

The S-RELAP5 and ANF-RELAP calculations for this event are nearly identical up until the time of scram. However, a small difference exists in the time at which the reactor trip signal is generated and the RPS setpoint that generates this signal is different for the two codes. S-RELAP5 calculates that the reactor will trip on high indicated thermal power at 19.9 seconds, while ANF-RELAP predicts the trip to occur on low SG pressure about 0.38 seconds earlier. This small difference in trip time makes an insignificant difference in the parameters that affect the MDNBR. Specifically, no observable differences exists in the core inlet flow rate or inlet temperature, and the peak rod heat flux calculated by S-RELAP5 is 133.4 percent of rated compared to 133.2 percent for ANF-RELAP.

Although the effect of this difference in scram time is negligible, it is still important to understand the underlying cause of the difference in predicted behavior. With ANF-RELAP the break flow is calculated to be about 2 percent higher than that of S-RELAP5 (see Figure 6.6). Therefore, SG inventories and pressures decline at a slightly faster rate and the SG low pressure setpoint is reached before the indicated thermal power setpoint. In the MSLB event, a steam-only constraint is placed on the flow leaving the SGs. The break flow, however, is a high quality two-phase mixture due to a small amount of condensation in the steamline, as frictional losses cause the pressure to decrease between the SGs and the break location. Consequently, the magnitude of the break flow would be affected by code modifications that affect the upstream pressure and quality. Specifically, modifications to the interfacial drag package and the improved formulation of the energy equation (see Section 3.3) would account for this small (~2 percent) difference in critical flow rate.

Conclusion

The results of the analysis demonstrate that S-RELAP5 provides a satisfactory representation of the event. Furthermore, despite minor differences in the predicted value of the break flow, the S-RELAP5 results were in close agreement with the ANF-RELAP results. Specifically, the reactor trip signal times differed by only 0.38 seconds and the peak core power was the same, 137.6 percent of rated power.

Also, the predicted MDNBR is greater than the applicable safety limit and the peak fuel centerline temperature is well below the fuel melting point. These results indicate that no fuel failures due to either DNB or to FCM would occur and, therefore, the event acceptance criteria are met.

Table 6.2 Pre-Scram MSLB Event Summary

Event	Time (s)
4.0 ft^2 Break in Steamline	0.0
Turbine Control Valves Open Fully	0.0
VHP Trip Setpoint Reached (Nuclear)	18.8
Reactor Trip Signal Generated	19.2
RCPs Trip (Loss of Offsite Power)	19.2
Peak Core Power	19.3
Scram CEA Insertion Begins	19.9
Peak Fuel Centerline Temperature	20.5
MDNBR	21.3

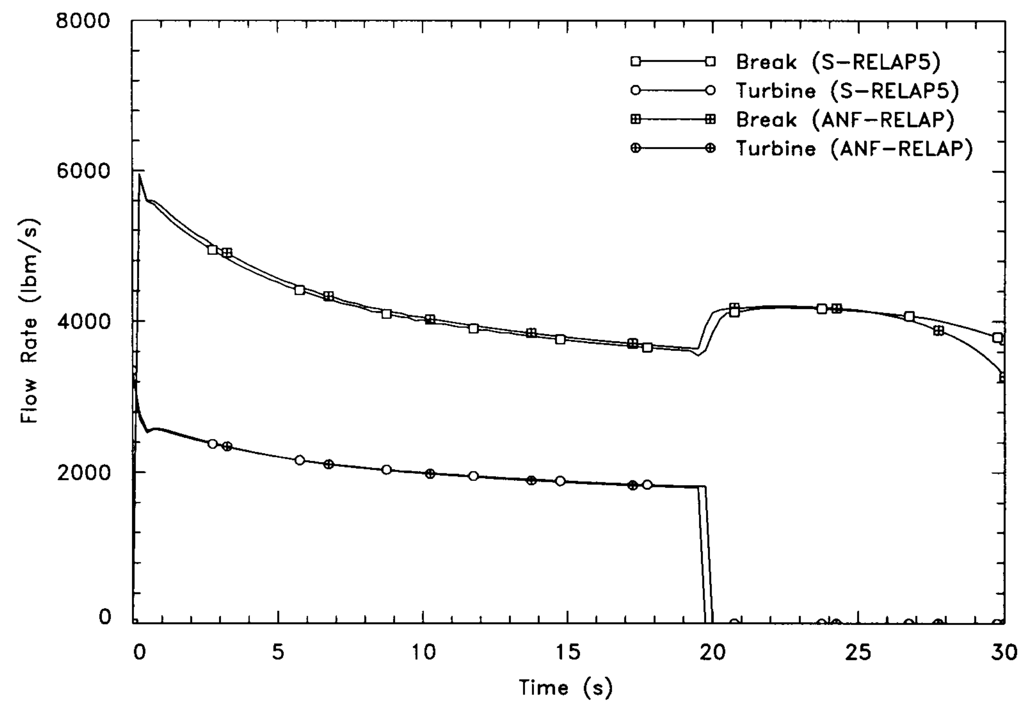

Figure 6.6 Pre-Scram MSLB Break and Turbine Steam Flow Rates

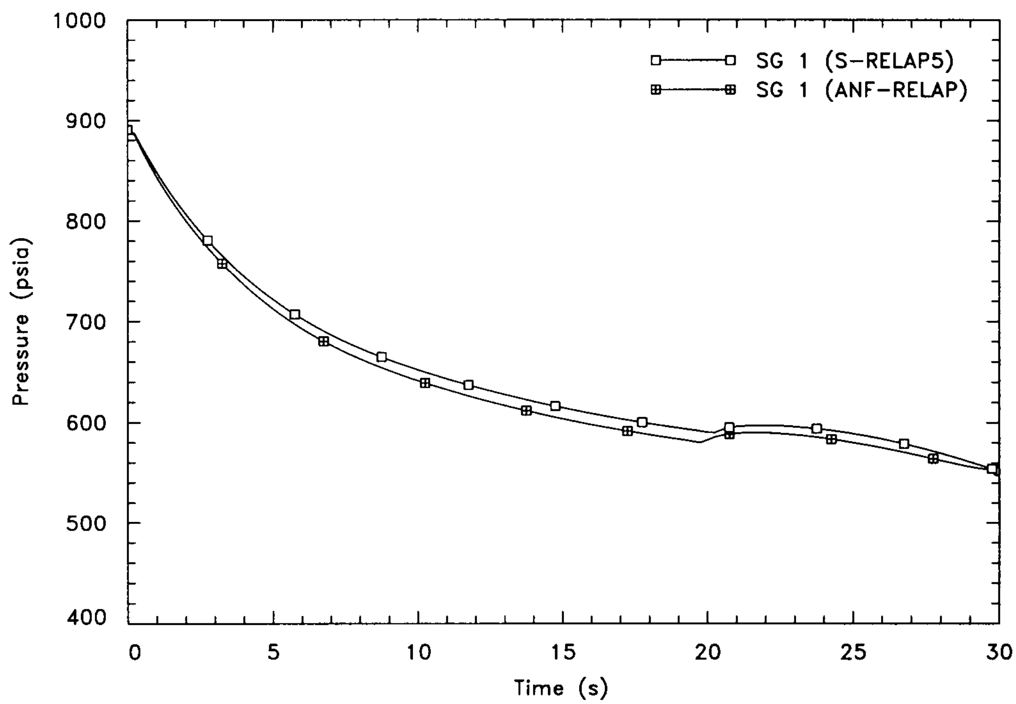

Figure 6.7 Pre-Scram MSLB Steam Generator Pressures

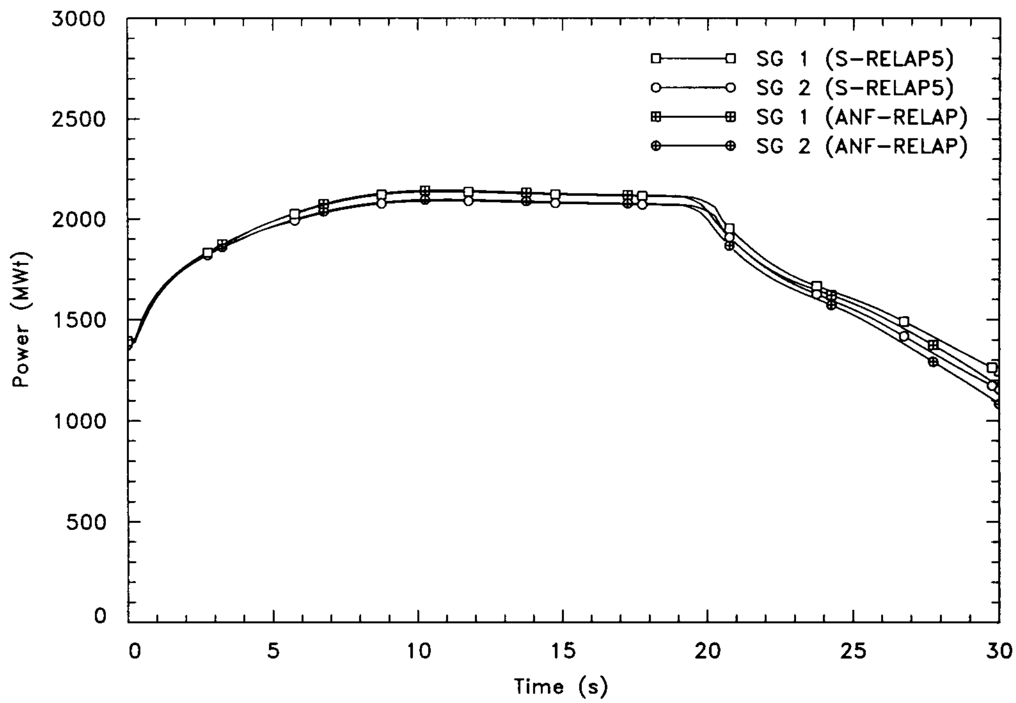

Figure 6.8 Pre-Scram MSLB Steam Generator Heat Transfer Rates

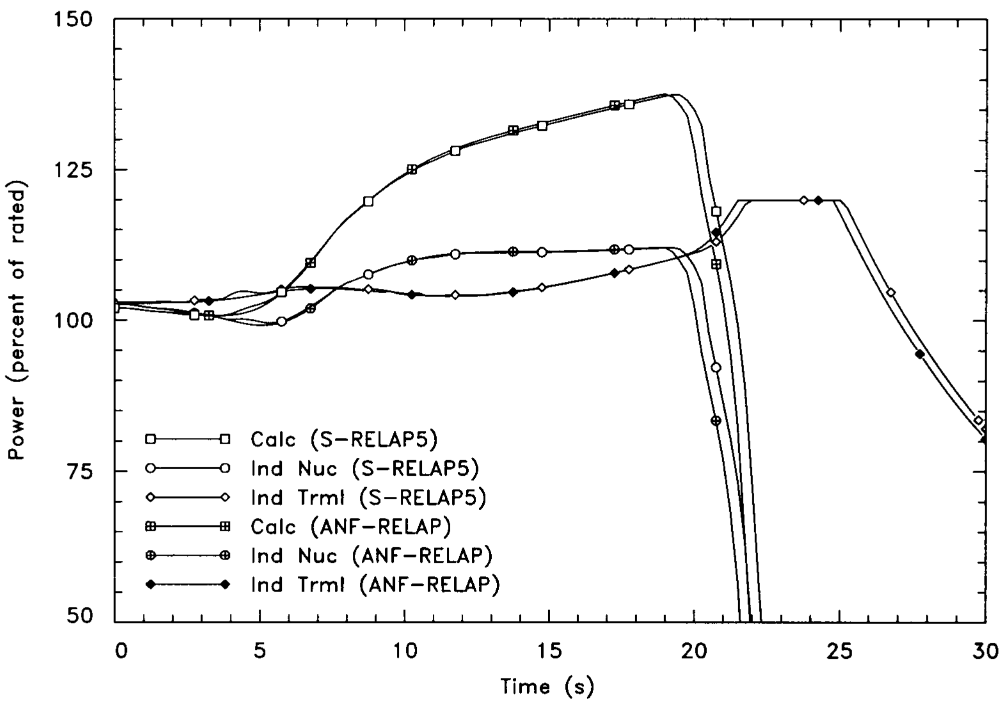

Figure 6.9 Pre-Scram MSLB Calculated Reactor, Indicated Nuclear, and Indicated Thermal Core Powers

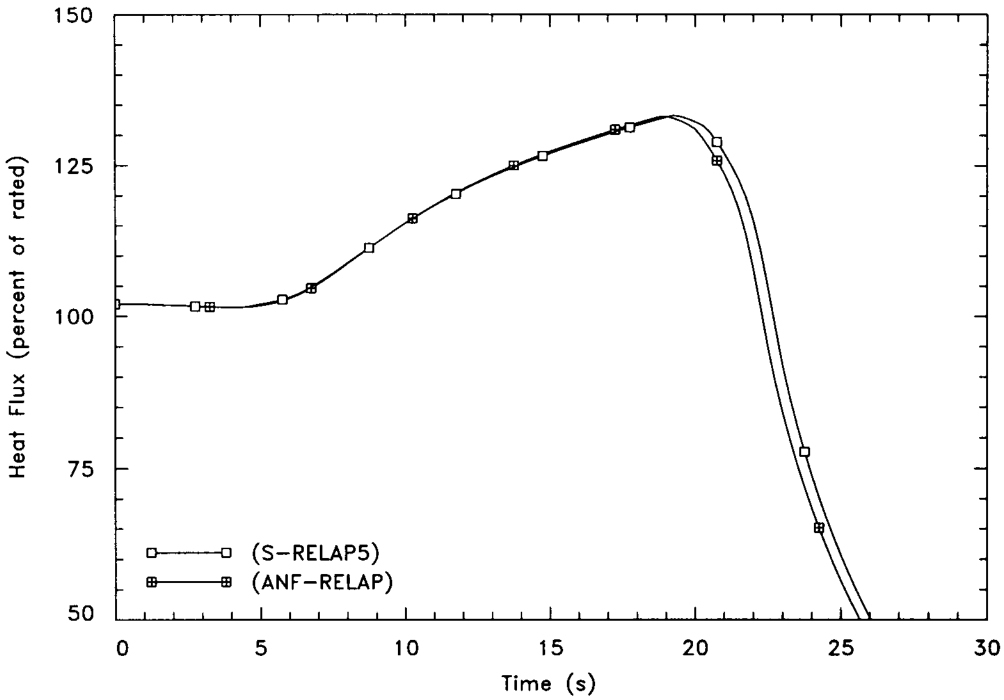

Figure 6.10 Pre-Scram MSLB Average Fuel Rod Heat Flux

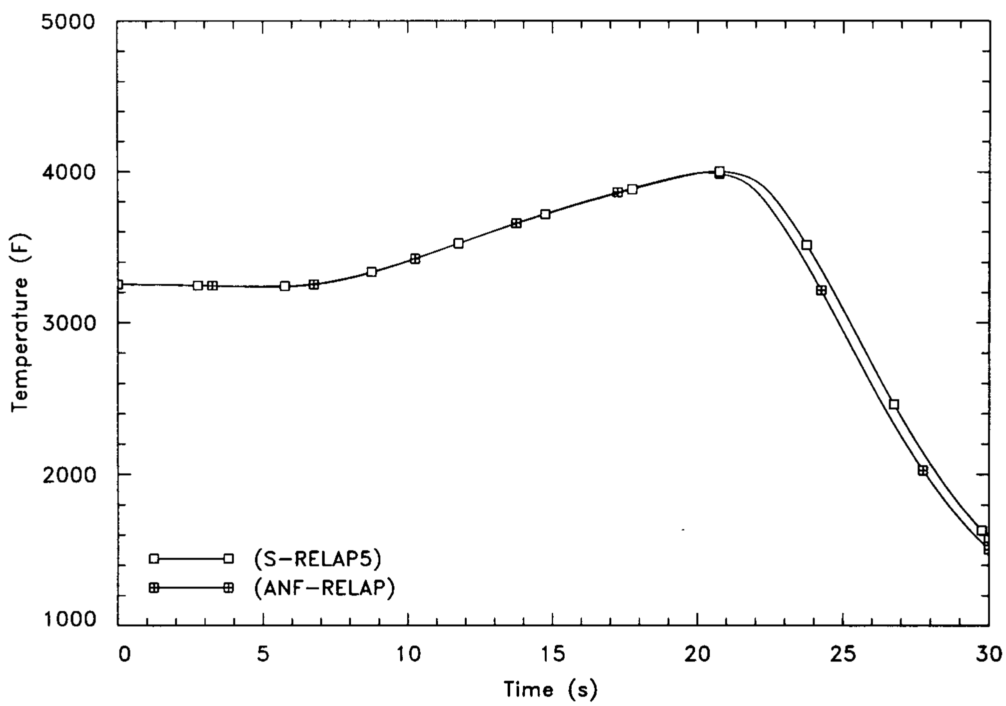

Figure 6.11 Pre-Scram MSLB Peak Fuel Centerline Temperature

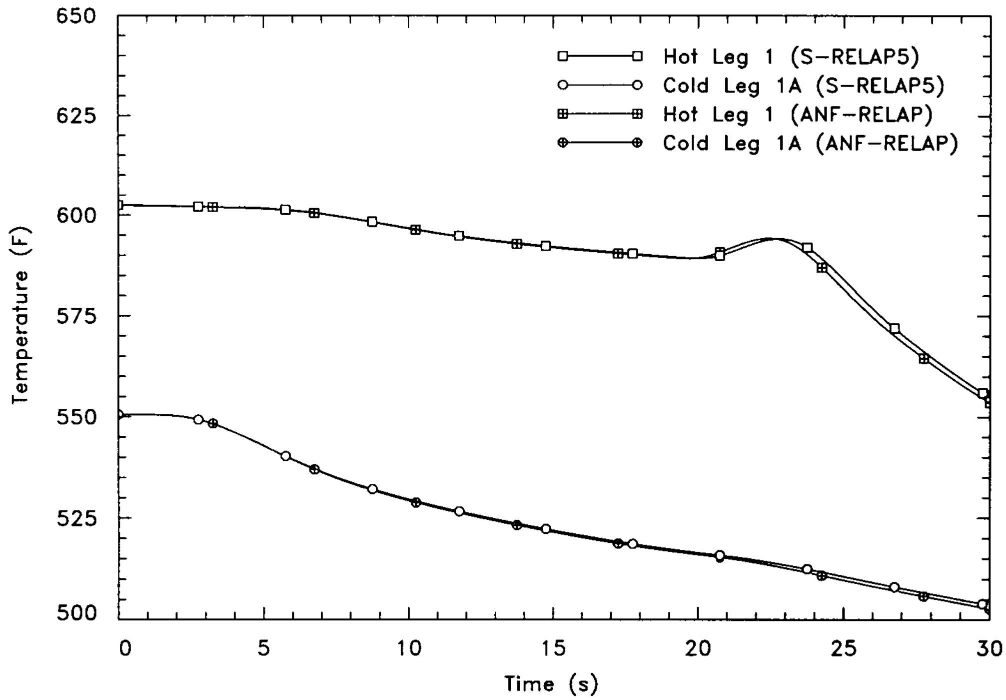

Figure 6.12 Pre-Scram MSLB RCS Hot Leg and Cold Leg Temperatures

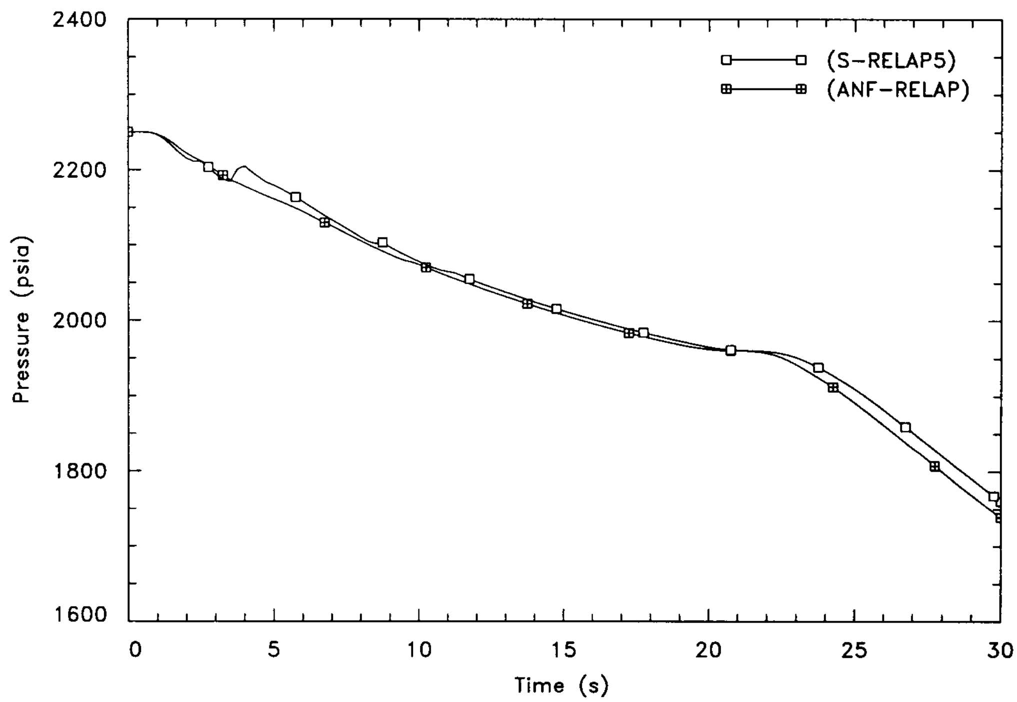

Figure 6.13 Pre-Scram MSLB Pressurizer Pressure

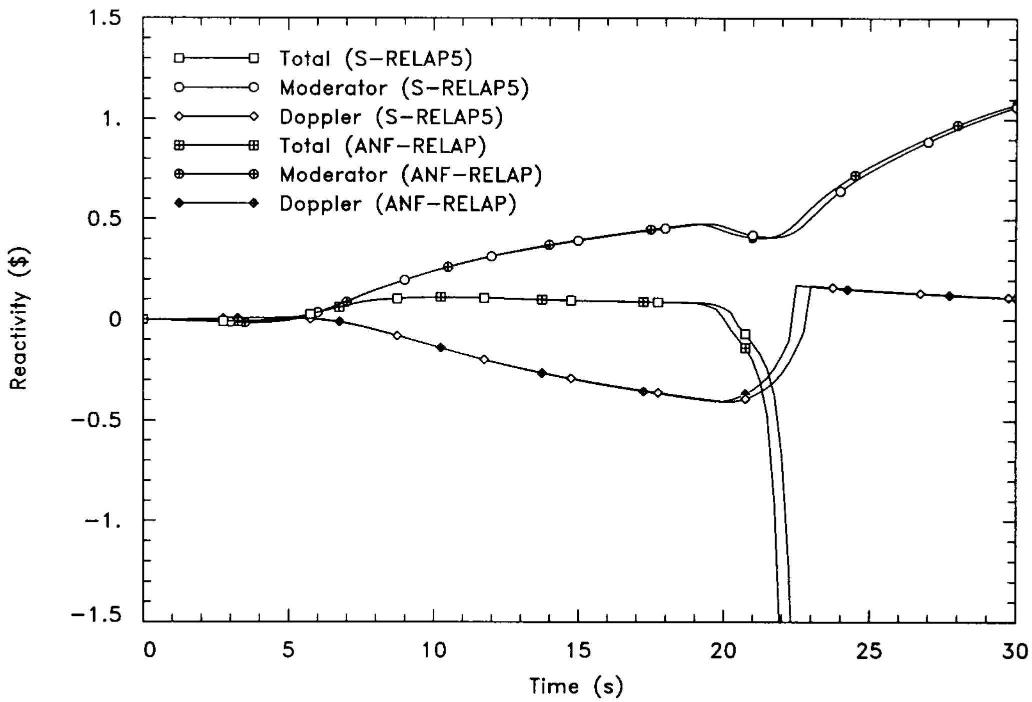

Figure 6.14 Pre-Scram MSLB Reactivity Components

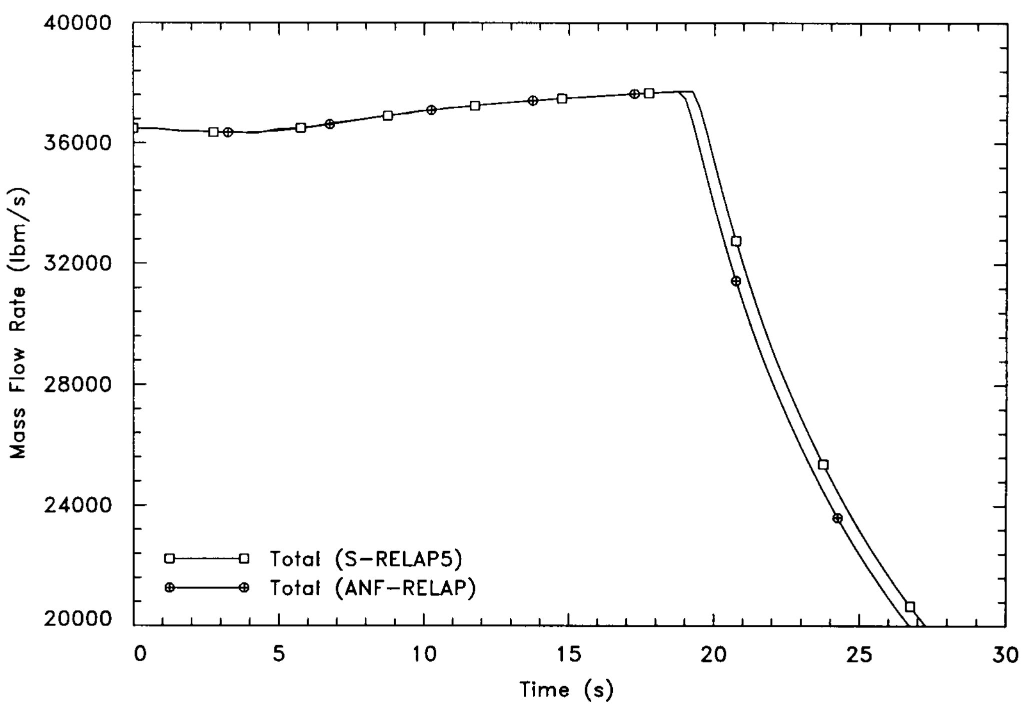

Figure 6.15 Pre-Scram MSLB Core Inlet Flow Rate

6.2 *Post-Scram Main Steamline Break (MSLB)*

Event Description

The most limiting Post-Scram MSLB event for the sample problem is initiated by a double-ended guillotine break in a main steamline upstream of the MSIV at EOC conditions. After closure of the MSIVs on low SG pressure, the transient becomes substantially asymmetric, with only the affected SG continuing to blow down. The release of high-energy steam through the break creates a power-cooling mismatch between heat generated in the core and that removed in the SGs. For the sector of the core associated with the affected SG, a rapid cooldown results. If the MTC is negative, this cooldown would cause an insertion of positive reactivity with a potential for a return to power and an erosion of the thermal margin.

Definition of Events Analyzed

The most limiting case was determined to be an inside-containment break initiated at HFP conditions with offsite power available to operate the RCPs. All four RCPs were assumed to be operational throughout the transient so that forced flow conditions are maintained in the RCS. EOC conditions were selected to maximize the magnitude of the negative MTC, thereby maximizing the positive reactivity insertion. Following reactor scram on low SG pressure, all control element assemblies (CEAs) were assumed to be inserted except for the most reactive CEA which is assumed to be stuck in the withdrawn position. Additional conservatism is obtained by locating the stuck CEA in the core sector being cooled with inlet water from the affected loop.

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In accordance with the worst-single-active-failure analysis requirement, it was postulated that one of the two HPSI pumps required to be in service fails. However, note that this transient simulation is completed before the SI lines fill with borated water and begin delivery to the RCS cold legs.

Analysis Results

Table 6.3 presents the sequence of events and the responses of key system variables are given in Figure 6.16 through Figure 6.26. To provide a direct comparison with S-RELAP5, the ANF-RELAP results are included in these figures.

Initially, the release of high-energy steam through the break causes an increase in the primary-to-secondary heat transfer rate for both SGs. Upon MSIV closure (on low SG Pressure ESFAS signal), the cooldown of the loop with the unaffected SG ends, but the cooldown of the affected loop continues until the AFW is terminated and the SG dries out.

Shortly after the transient is initiated, the reactor is scrammed (on a Low SG Pressure RPS signal). However, as the cooldown progresses, the shutdown worth is eroded by moderator and Doppler feedback (accentuated by the EOC conditions) until a return to power occurs. The increase in core power above the decay heat level is eventually terminated by negative Doppler and moderator feedback after the AFW flow is shut off by operator action. The core power peaks at 8.8 percent of the rated power, with most of the power produced in the stuck-CEA region. The resulting MDNBR is 3.21 and the peak LHGR is 17.54 kW/ft.

Only small differences between the S-RELAP5 and ANF-RELAP results are observable for this event. The key parameter is the degree to which the plant experiences a post-scram return to power due to the reactivity insertion associated with the cool down. S-RELAP5 calculated a very modest return to power of only 8.8 percent of rated, and the ANF-RELAP results were within 0.1 percent of this value. Still, there were some minor differences in the predicted plant behavior as discussed below.

Early in the transient, see Figure 6.26, the return to power begins a little sooner in the ANF-RELAP calculation due to a slightly more rapid cool down for the affected sector (see Figure 6.21). This initial difference in the core inlet temperature for the affected sector is a result of the slightly more rapid SG blowdown (see Figure 6.17) associated with the difference in the calculated critical flow noted above for the MSLB pre-scram event.

Later in the transient, for the unaffected SG, a noticeable difference in pressure occurs (see Figure 6.17) with S-RELAP5 predicting a higher value than ANF-RELAP. Similarly, there is a small difference in the core inlet temperature for the unaffected sector and the S-RELAP5 value for the primary pressure (see Figure 6.22) is slightly higher as well. For this part of the transient, the heat removal rate to the unaffected SG is minimal (see Figure 6.18) with just enough heat transfer to cause its pressure to slowly approach equilibrium with the primary. The increase in the primary pressure for the S-RELAP5 calculation relative to that of ANF-RELAP is small and appears to be due to increased RCP heat generation which in turn is a result of the increased wall drag for the SG tubes due to the improved formulation for the single-phase friction factor.

The most obvious difference between the two calculations is the difference in HPSI flow rate as shown in Figure 6.24. For this event, the RCS pressure is very close to the HPSI pump shut off head, so that the maximum calculated HPSI flow is only about 25 percent of the full minimum degraded flow for one HPSI pump. Consequently, small differences in the calculated RCS pressure are reflected as relatively large changes in the HPSI flow rate. However, the difference in the calculated values for HPSI flow rate is only a small fraction (~ 2-4 percent) of the rated HPSI flow and is a negligible fraction of the core flow since the RCPs were not tripped. As was the case with the pressure in the unaffected SG, the difference in HPSI flow is caused by the small difference in primary pressure.

These observable differences in the predicted behavior for this event are attributable to two improvements made to the S-RELAP5 code, the more exact formulation for wall drag and the improved energy equation due to its effect upon the critical flow. However, none of these differences had a significant effect upon the predicted peak power since the values calculated by the two codes agreed to within 0.1% of rated power.

Conclusion

This analysis demonstrates that S-RELAP5 provides a satisfactory representation of the event. Also, the predicted MDNBR is much greater than the applicable safety limit, and the peak LHGR is well below the FCM threshold. These results indicate that no fuel failures due to either DNB or FCM would occur and, therefore, the event acceptance criteria are met.

The S-RELAP5 calculated results are in close agreement with those of ANF-RELAP except for the HPSI flow rate (Figure 6.24). The observed difference in HPSI flow rate was discussed above and has no significant effect for this event.

Table 6.3 Post-Scram MSLB Event Summary

Event	Time (s)
Double-Ended Guillotine Break in Main Steamline Upstream of MSIV	0.0
Turbine Valve is Assumed to Open Fully	0.0
Low SG Pressure and MSIS Setpoints Reached	7.4
Reactor Trip	8.3
Scram CEA Insertion Begins and Turbine Trips	9.1
MSIVs Fully Closed	14.3
Low Pressurizer Pressure Signal Initiates HPSI Pump Startup	26.4
MFW Valves Closed	67.4
RCS Pressure Reaches HPSI Pump Shutoff Head and Borate Water Begins Filling SI Lines	124.6
AFW Starts and is all Directed to Affected SG	170.0
Core Returns to Critical Condition	229.0
AFW Terminated (Operator Action)	600.0
Peak Post-Scram Power Reached	602.0

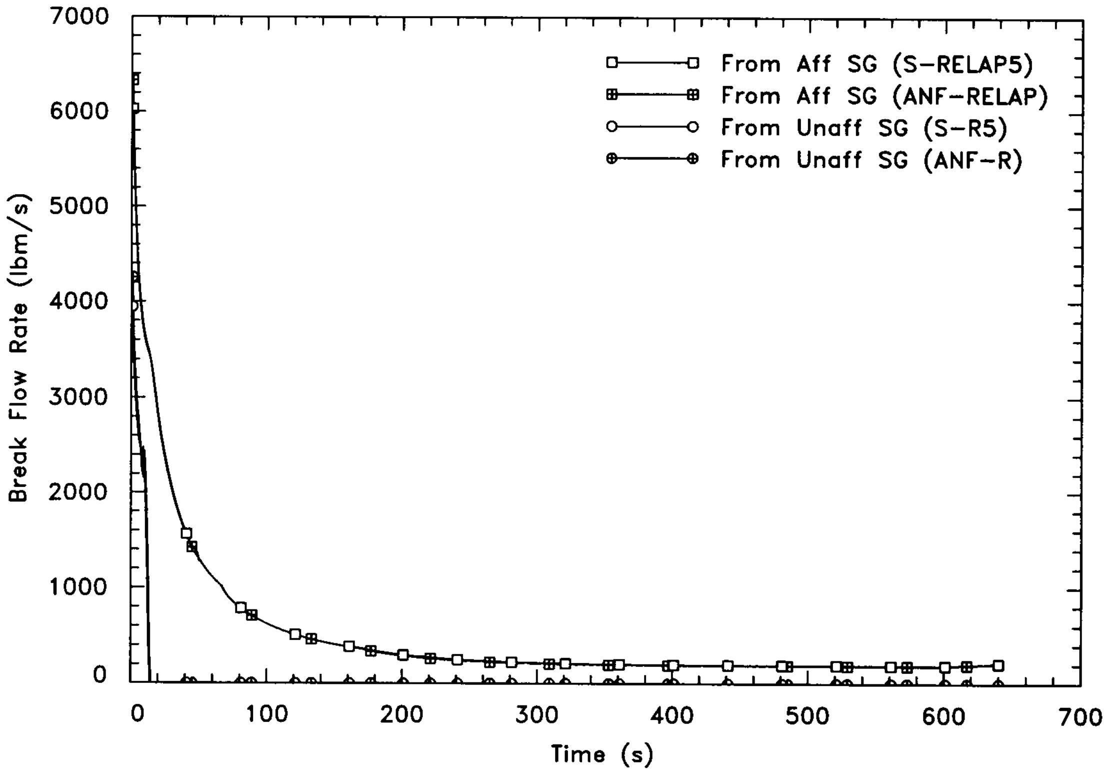

Figure 6.16 Post-Scram MSLB Break Flow Rates

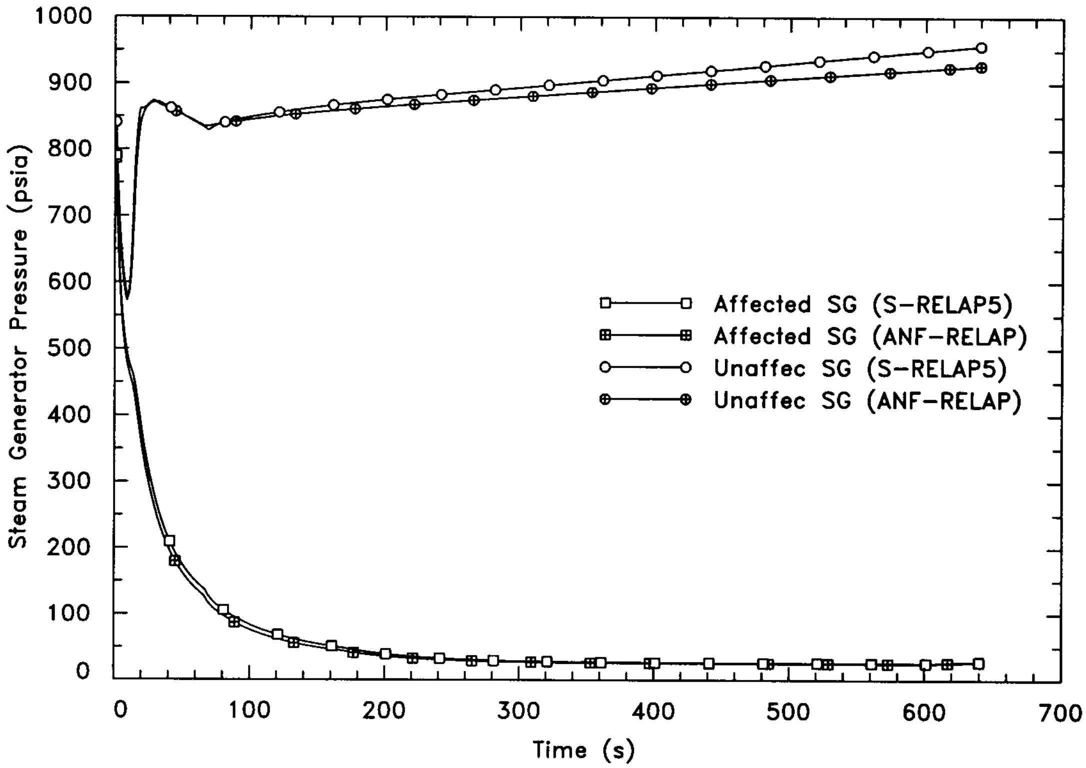

Figure 6.17 Post-Scram MSLB Steam Generator Pressures

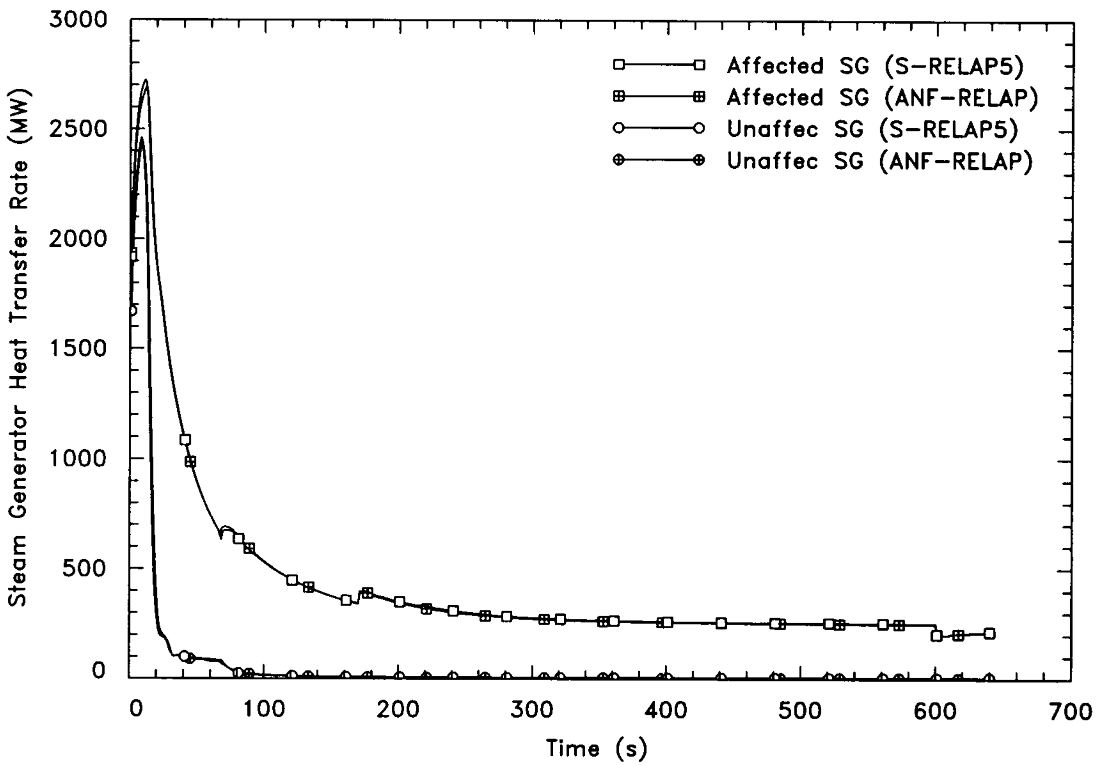

Figure 6.18 Post-Scram MSLB Steam Generator Heat Transfer Rates

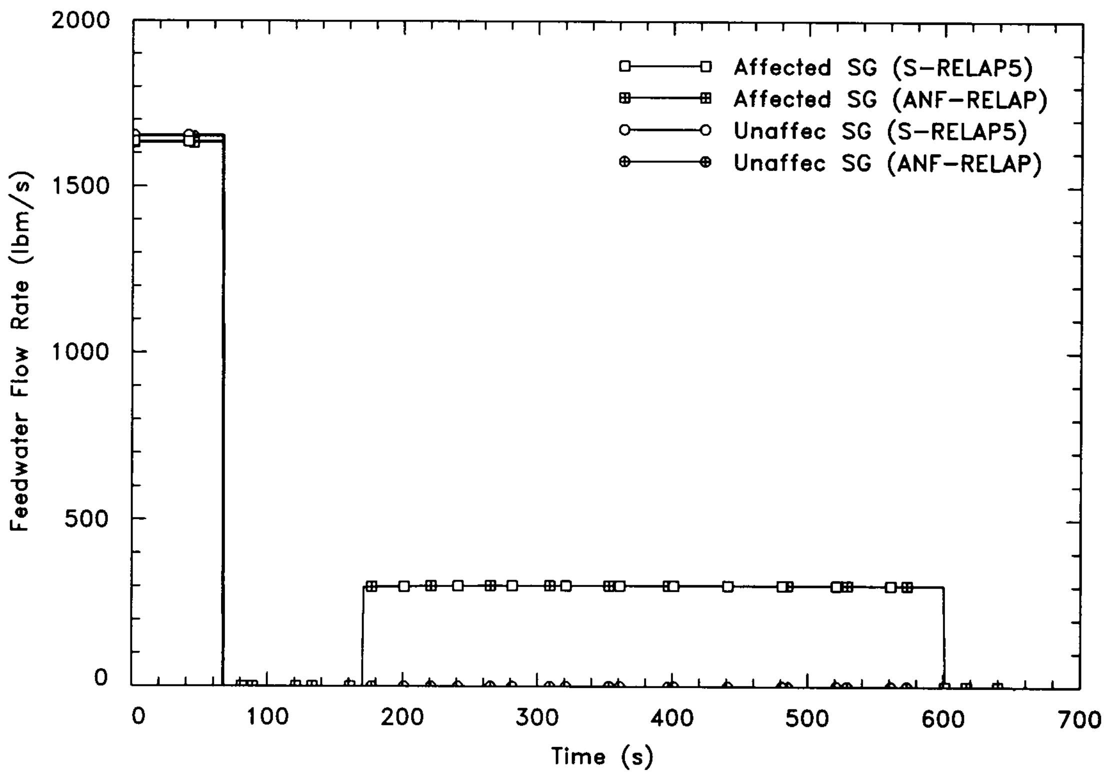

Figure 6.19 Post-Scram MSLB Feedwater Flow Rates

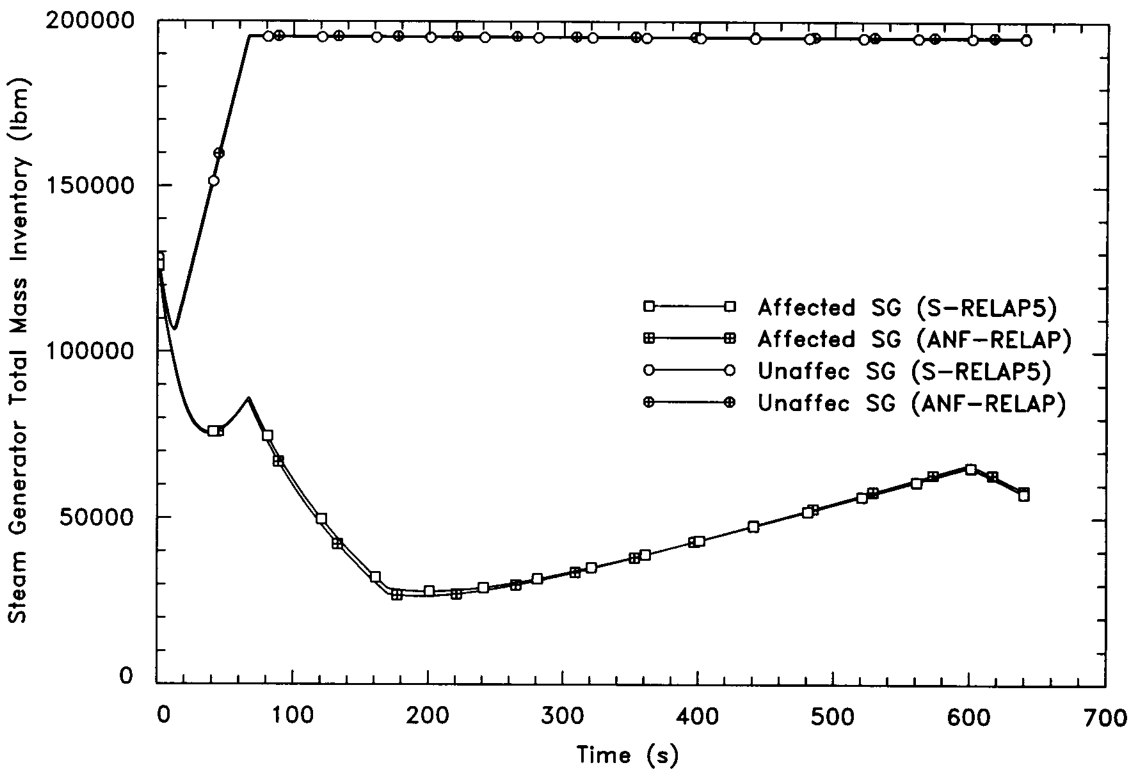

Figure 6.20 Post-Scram MSLB Steam Generator Total Mass Inventories

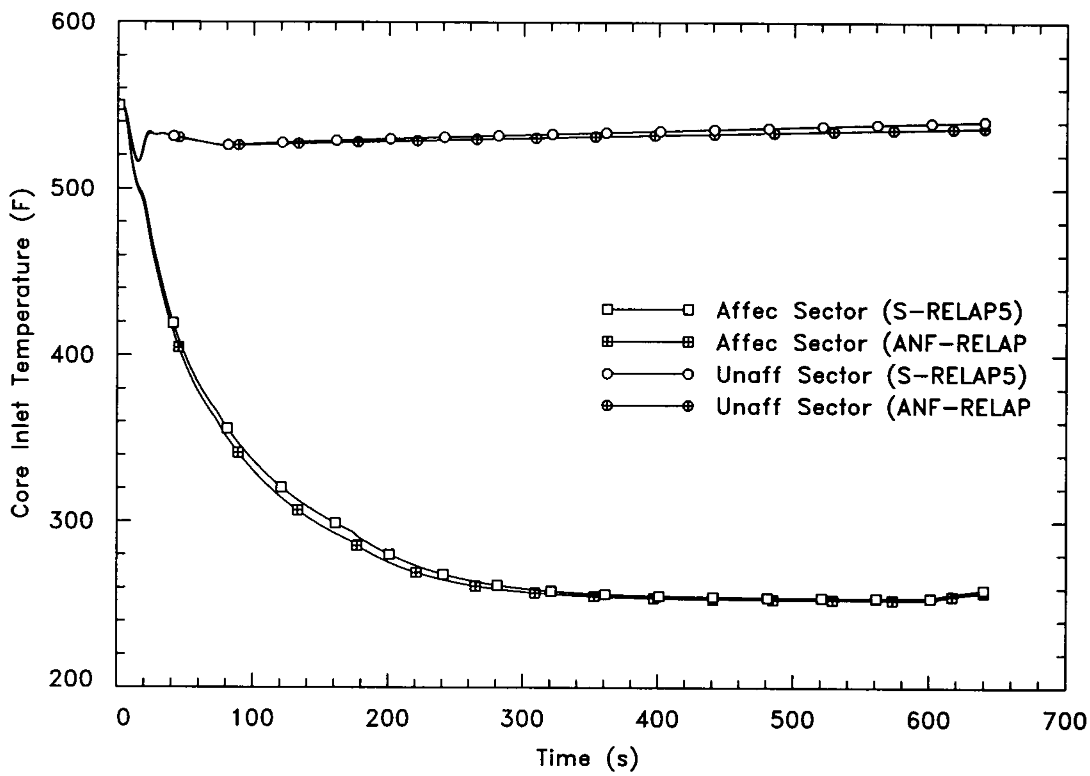

Figure 6.21 Post-Scram MSLB Core Inlet Temperatures

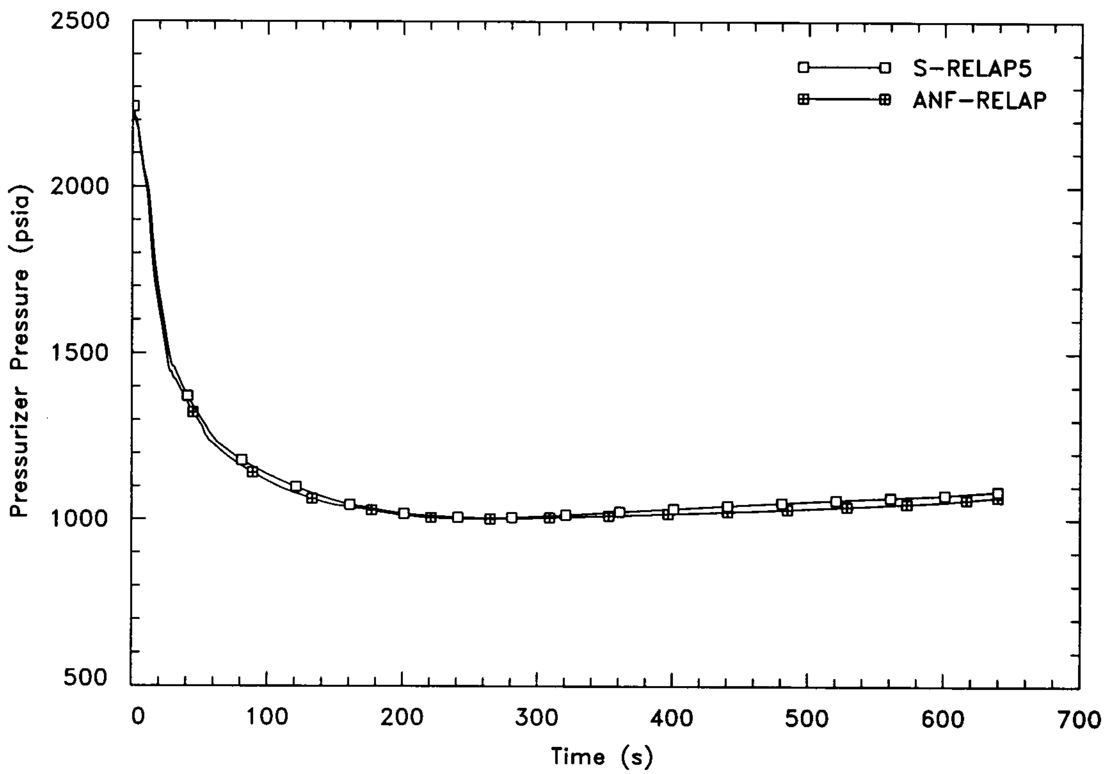

Figure 6.22 Post-Scram MSLB Pressurizer Pressure

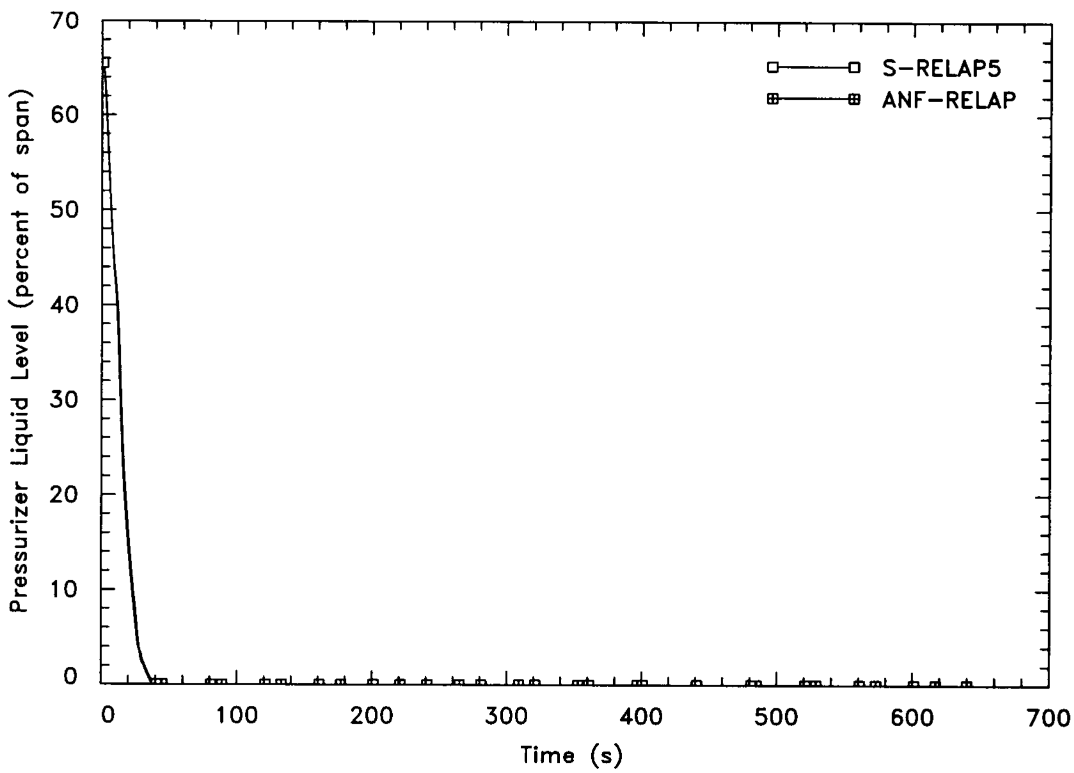

Figure 6.23 Post-Scram MSLB Pressurizer Liquid Level

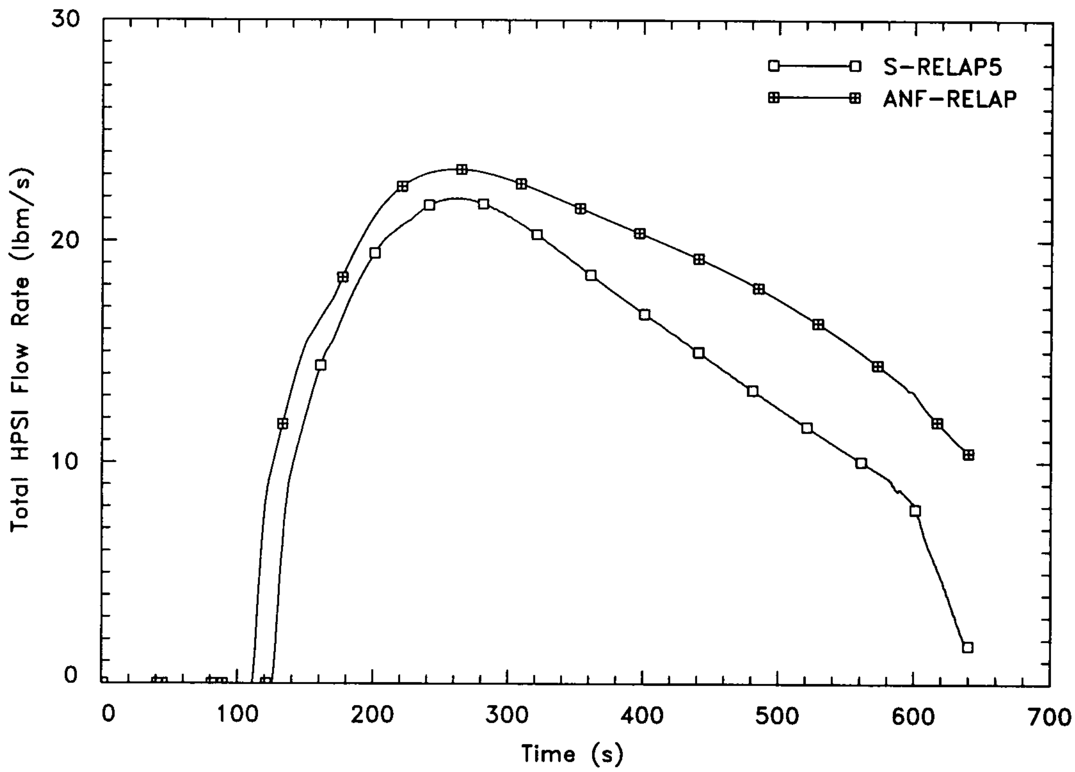

Figure 6.24 Post-Scram MSLB Total HPSI Flow Rate

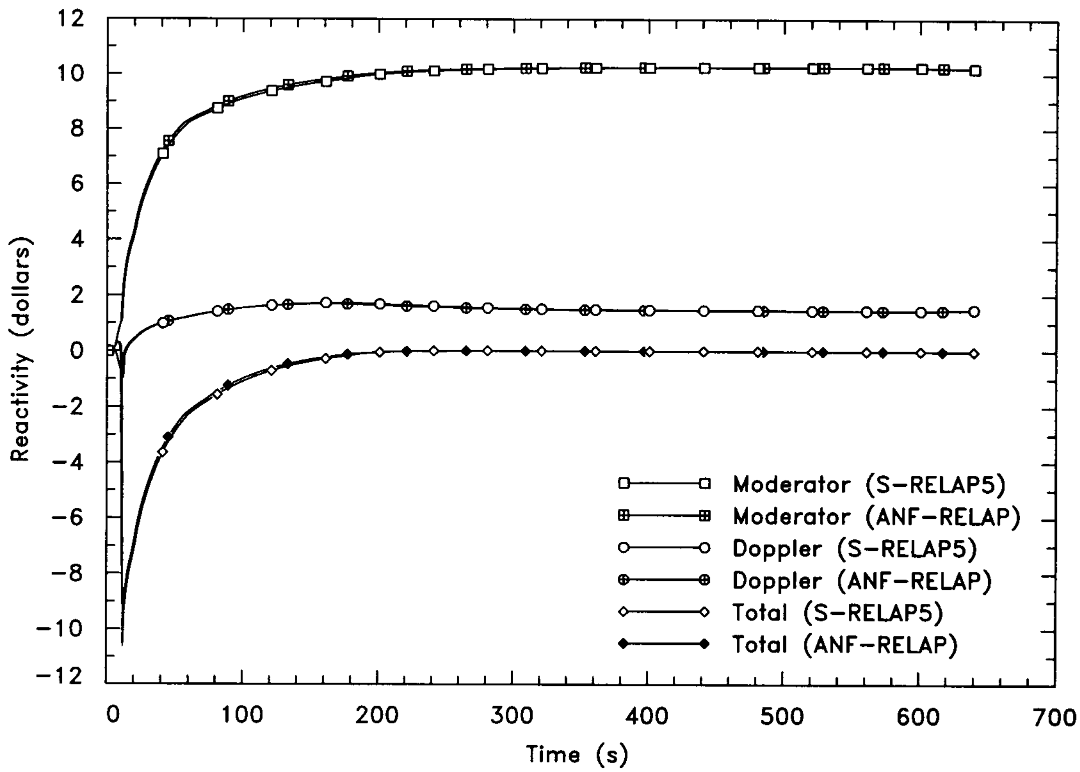

Figure 6.25 Post-Scram MSLB Reactivity

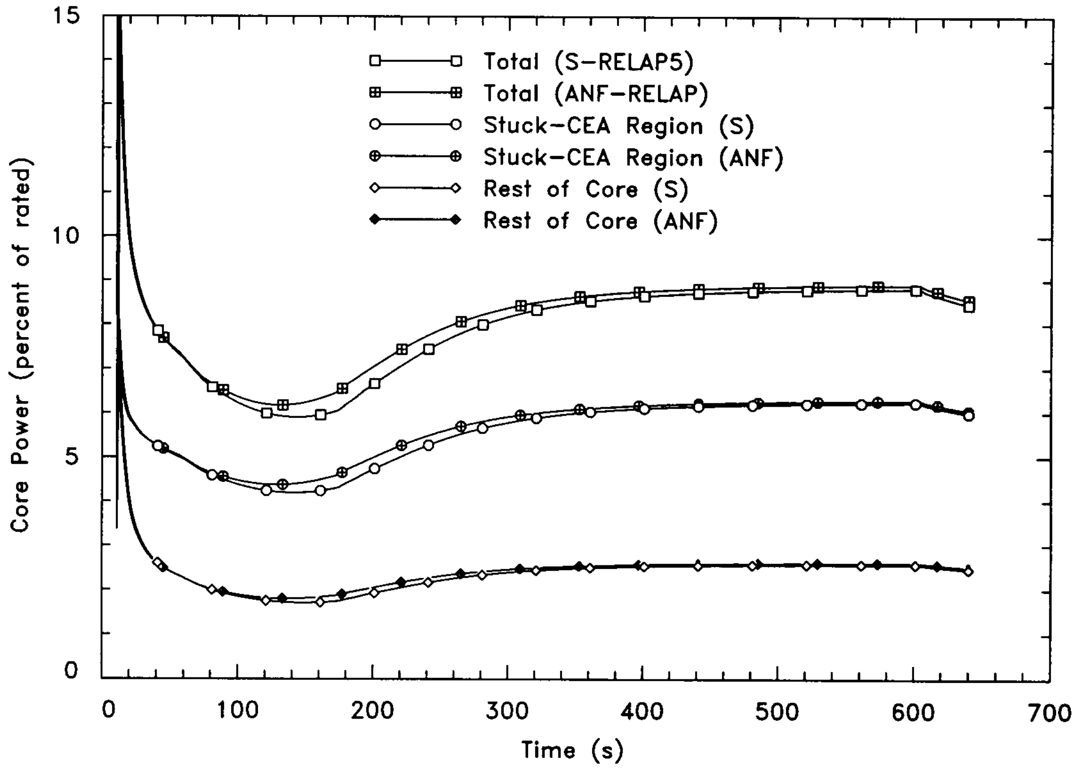

Figure 6.26 Post-Scram MSLB Core Power

6.3 *Loss of External Load (LOEL)*

Event Description

This event is initiated by either an LOEL (Event 15.2.1) or a TT (Event 15.2.2). The major difference between the two events is the rate at which steam flow is reduced. Following a LOEL, a runback is initiated and the turbine throttle valves close at a moderately fast rate, but not instantaneously. In a turbine trip, the turbine stop valves close almost instantly (typically within 0.1 second). When sufficient margin exists, a transient scenario is constructed so that the safety analysis results bound the consequences for both LOEL and TT events as illustrated in this sample problem.

Upon either of these two conditions, the turbine stop valve is assumed to rapidly close (0.1 s). Normally an anticipatory reactor trip would occur on a turbine trip; however, to calculate a conservative system response, the reactor trip on turbine trip is disabled. The atmospheric steam dump valves (ADVs) are also assumed to be unavailable. These assumptions allow the analysis to bound the consequences of Event 15.2.1 (Loss of External Load), Event 15.2.2 (Turbine Trip - Steam Atmospheric Dump Unavailable) and Event 15.2.4 (Closure of both MSIVs - valve closure time is greater than 0.1 s).

The LOEL/TT event challenges the acceptance criteria for both primary and secondary system overpressure and DNBR. The event results in an increase in the reactor coolant system temperatures due to an increase in the secondary side temperature. As the reactor coolant system temperatures increase, the reactor coolant throughout the RCS expands causing an increase in the pressurizer pressure. The reactor coolant system is protected against overpressurization by the pressurizer safety and relief valves. Pressure relief on the secondary side is afforded by the steamline safety and relief valves. Actuation of the primary and secondary system safety valves limits the magnitude of the reactor coolant system temperature and pressure increase.

With a positive MTC, increasing reactor coolant system temperatures result in an increase in core power. The increasing primary side temperature and power reduces the margin-to-thermal limits and challenges the DNBR acceptance criterion.

Definition of Events Analyzed

The objectives in analyzing this event are to demonstrate that: 1) the reactor coolant pressure relief capacity is sufficient to limit the pressure to less than 110 percent of the design pressure, 2) the secondary side pressure relief capacity is capable of limiting the pressure to less than 110 percent of the design pressure, and 3) the MDNBR remains above the safety limit. To conservatively bias the calculation, no credit is taken for direct reactor trip on turbine trip, for the turbine bypass system, or for the steam dump system. For each of the above three objectives, a separate analysis would be conducted with the plant parameters biased so as to maximize the challenge for the particular criterion being examined.

In this sample problem, the analysis is biased to challenge the RCS design pressure limit. [

] This procedure provides for a conservative estimate of the peak RCS pressure during the transient.

Analysis Results

This maximum RCS pressurization case initiates with a ramp closure of the turbine valve in 0.1 seconds. The pressurization of the secondary side results in decreased primary-to-secondary heat transfer, and a rise in reactor coolant system temperatures. An insurge into the pressurizer occurs, compressing the steam space and pressurizing the reactor coolant system. The reactor trips on high pressure. The capacity of the pressurizer safety valves is sufficient to contain the maximum RCS pressure (bottom of the vessel) to a maximum value of 2692 psia.

The sequence of events is given in Table 6.4 for this maximum RCS pressure case. The responses of key system variables are given in Figure 6.27 to Figure 6.33. For code-to-code comparisons, ANF-RELAP results are included in the figures.

The S-RELAP5 calculated results are in excellent agreement with those of ANF-RELAP. In particular, the peak RCS pressure was the key parameter and differed by only 0.8 psia for the two codes. This small difference is insignificant compared to the margin remaining to the RCS pressurization acceptance criterion (about 57 psia). Until the time that the peak primary pressure occurs, the results of the two calculations are virtually indistinguishable. After this time, a minor difference in the calculated SG flow rates is observed due to a difference in the MSSV re-seating behavior.

This difference is not the result of any difference in the valve models between the two codes but rather is the product of the way the control variable logic (user input) has been set up. The safety relief valves are modeled using a motor valve with trips specified for valve opening and closing. If one of these trips is true, then the valve opens (or closes) at a specified rate, however, if neither of these trips is true the valve position remains unchanged. Consequently, insignificant differences in the computed variables that govern the trips can lead to noticeable differences in the position of a partially open valve.

Conclusion

The S-RELAP5 results are in excellent agreement with the ANF-RELAP results and reasonably represent the plant transient. The difference in the peak pressure calculated with the two codes is only 0.8 psi. The maximum predicted RCS pressure (2692 psia) remains below 110 percent of the design pressure (2748 psia). Therefore, the RCS pressurization criterion for the LOEL and TT events is met.

Table 6.4 LOEL/TT Event Summary

RCS Overpressurization Case Event Summary

Event	Time (s)
Turbine Trip	0.00
Turbine Stop Valve Fully Closed	0.1
MSSVs Open	4.0
Reactor Trip Setpoint Reached on High Pressurizer Pressure	5.3
Scram Rod Insertion Begins (Instrumentation and Holding Coil Delays)	6.9
Peak Core Power	6.9
Pressurizer Safety Valves Open	7.4
Peak RCS Pressure (Bottom of Vessel)	8.3
Pressurizer Safety Valves Close	10.9

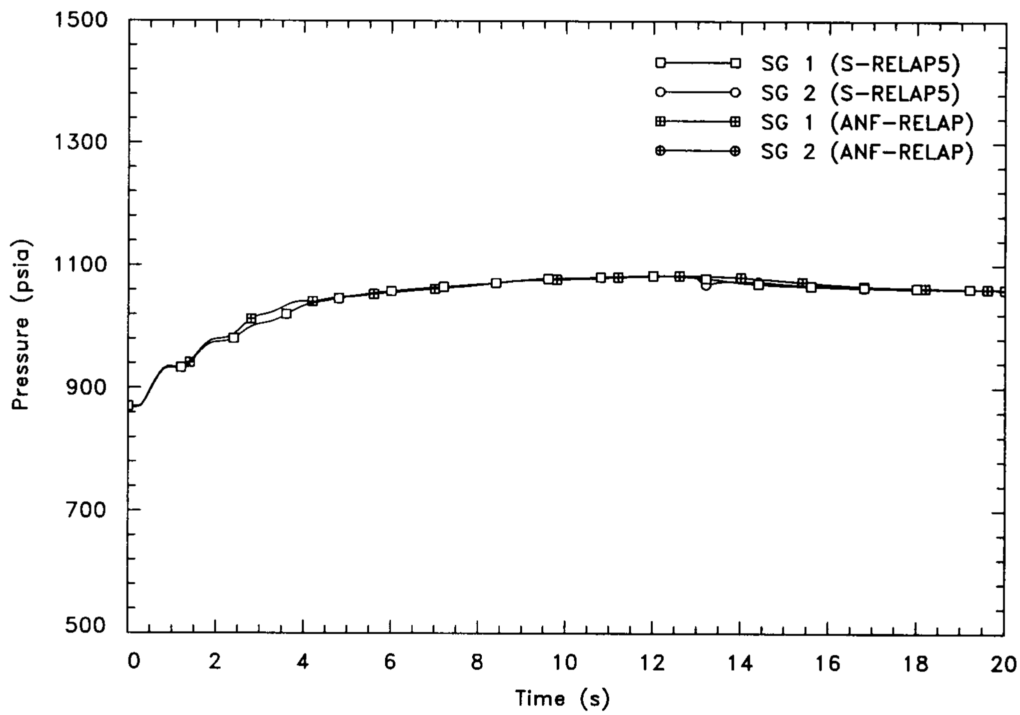

Figure 6.27 LOEL/TT Steam Generator Pressures

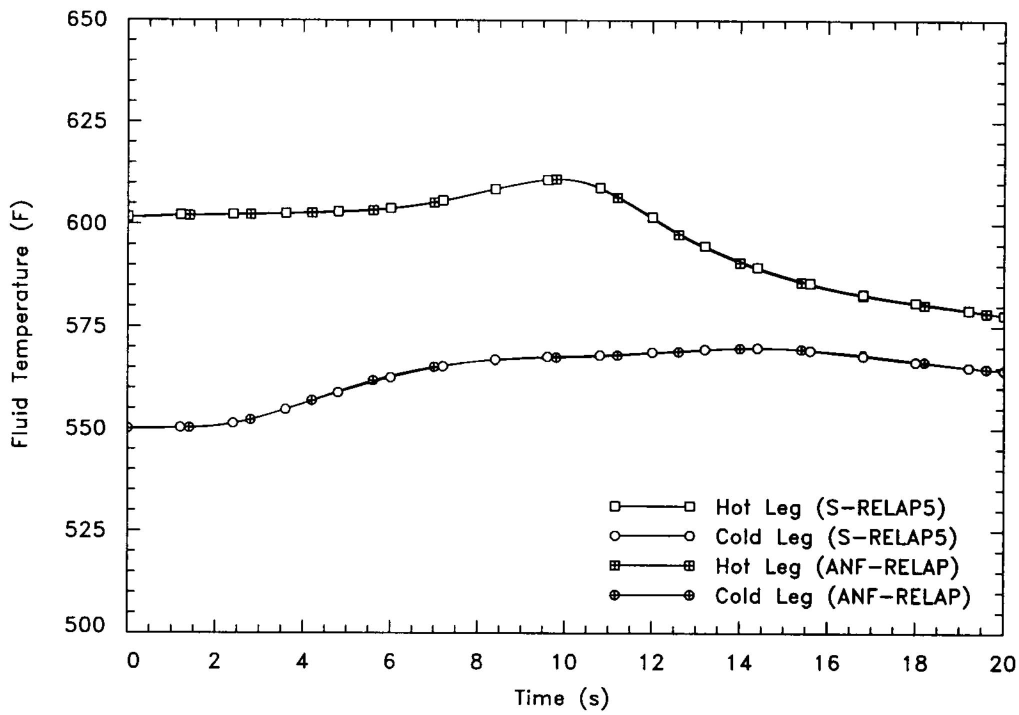

Figure 6.28 LOEL/TT RCS Temperatures

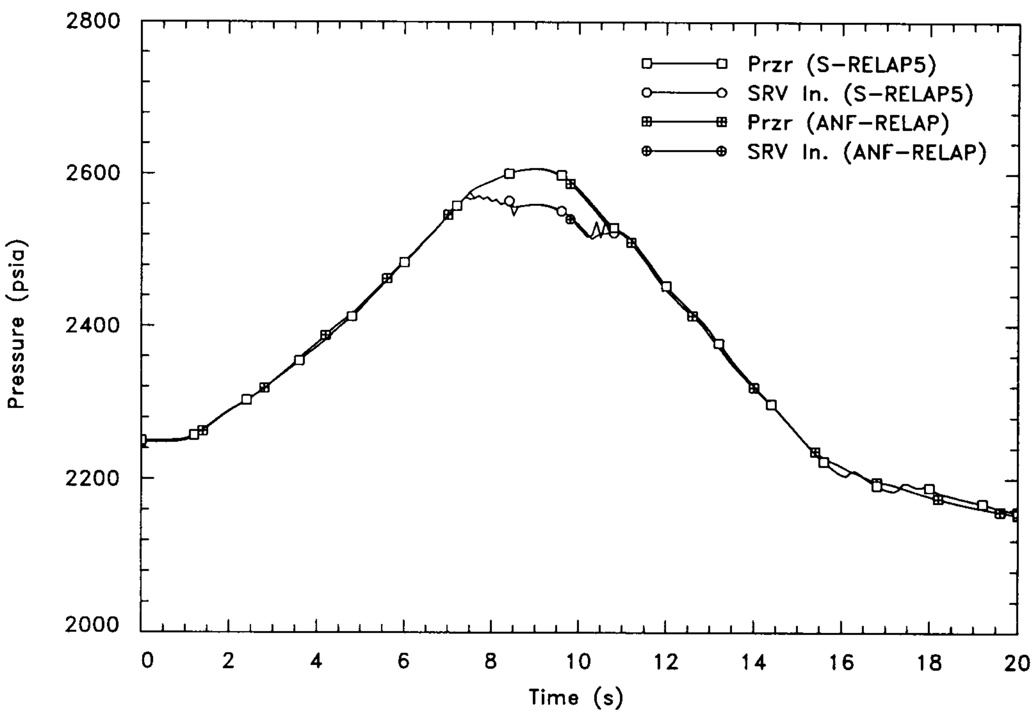

Figure 6.29 LOEL/TT Pressurizer and SRV Inlet Pressures

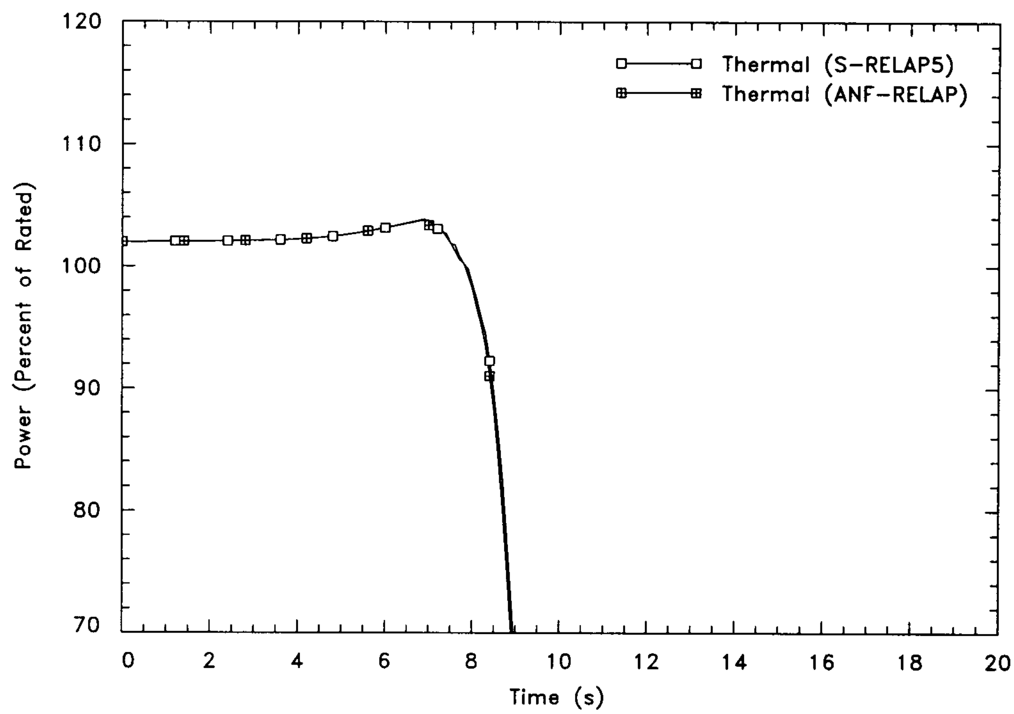

Figure 6.30 LOEL/TT Reactor Power

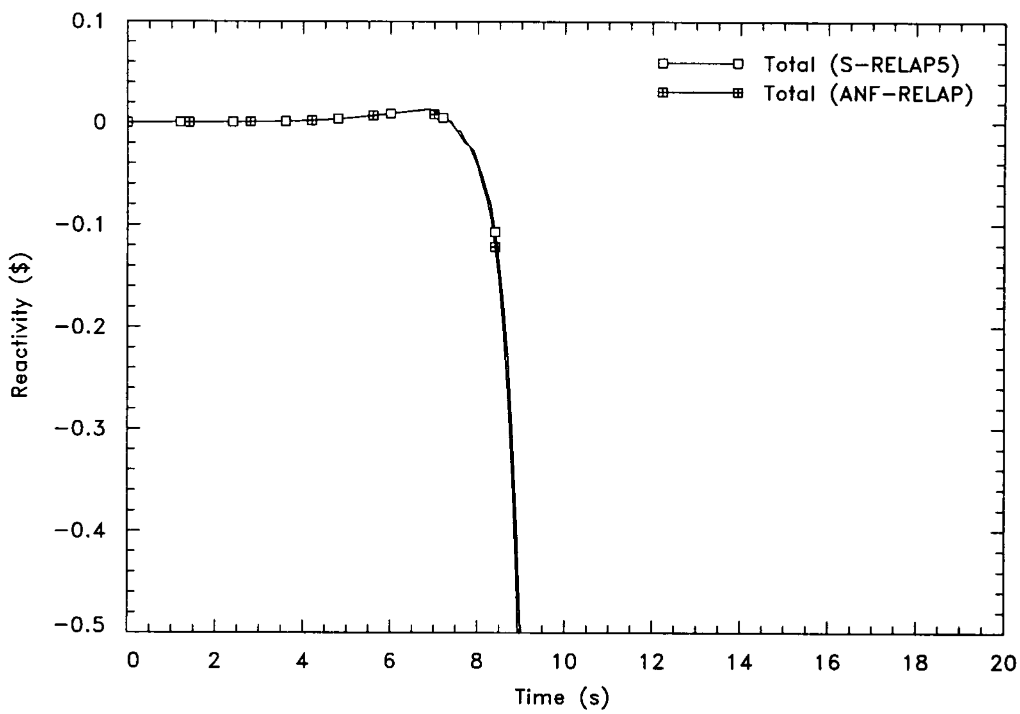

Figure 6.31 LOEL/TT Total Reactivity

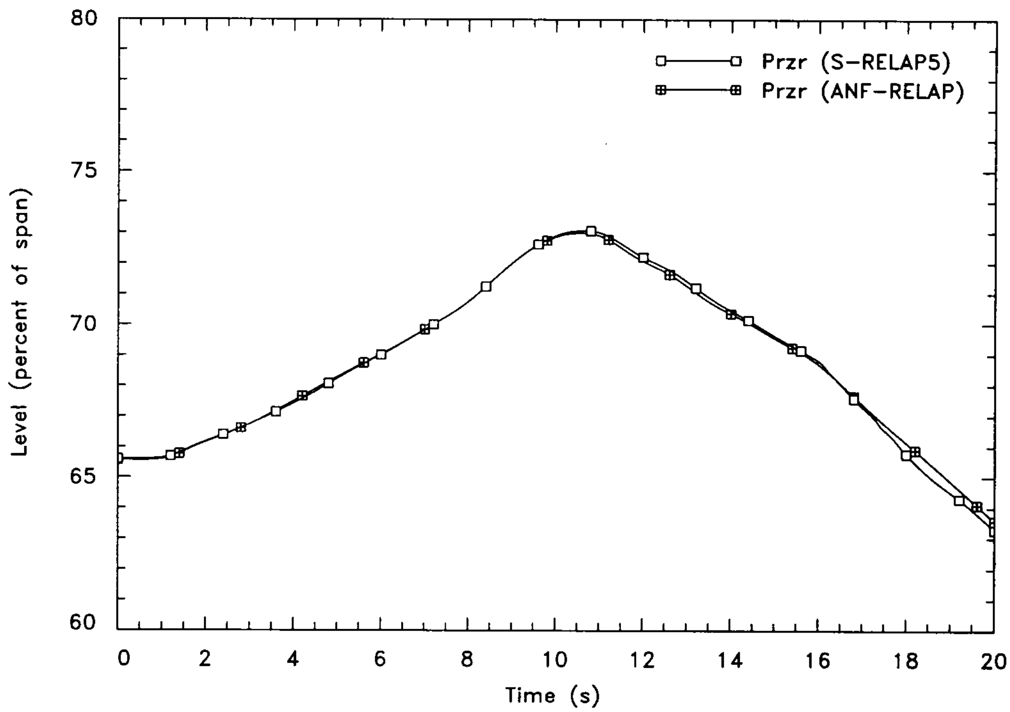

Figure 6.32 LOEL/TT Pressurizer Level

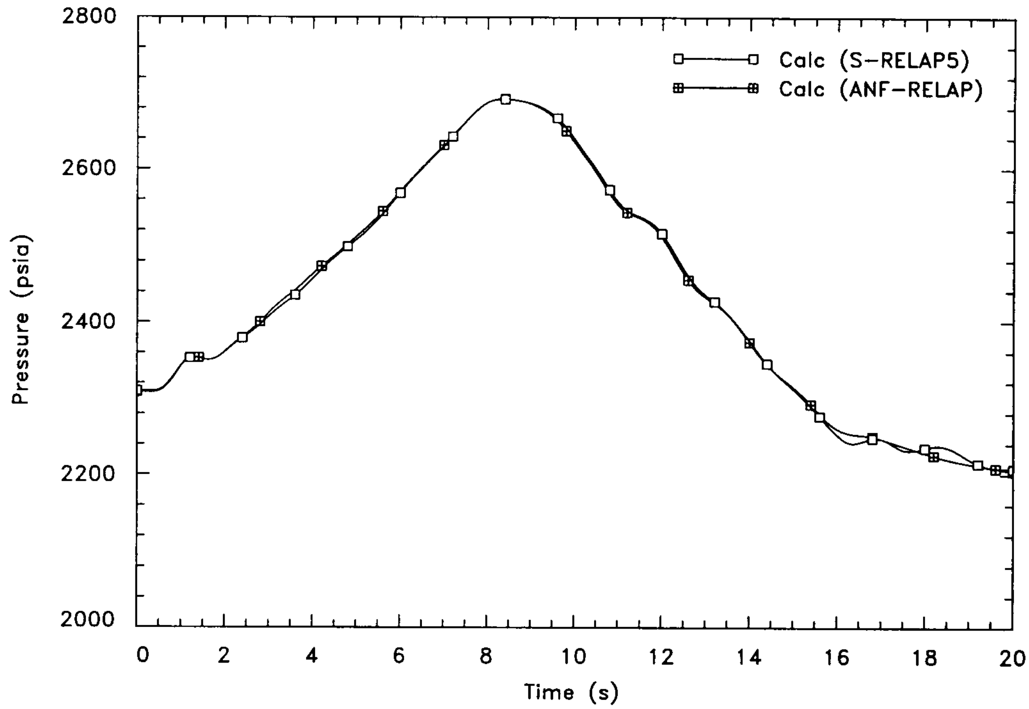

Figure 6.33 LOEL/TT Pressure at Bottom of Reactor Vessel

6.4 *Loss of Normal Feedwater (LONF) Flow*

Event Description

A LONF Flow transient is initiated by the termination of the MFW flow due to failures in the MFW or condensate systems. (The termination of MFW flow that results from a loss of power is considered in the Loss of Nonemergency AC Power event.) The termination of MFW flow while the plant continues to operate at power will eventually result in reactor scram on low SG level (or TM/LP or OTΔT[a]) with long-term cooling subsequently provided by the AFW system.

This event is evaluated to confirm that the low SG level reactor trip setpoint, the low-low SG level[b] AFW actuation setpoint, and the AFW flow capacity are adequate to provide for long-term decay heat removal. This event is also evaluated to confirm that the plant design and operating conditions preclude pressurizer overfill.

The loss of normal feedwater flow while the plant continues to operate at power causes the primary-to-secondary heat transfer rate to decrease. The resulting heatup of the reactor coolant causes a pressurizer insurge due to the fluid expansion. Reactor coolant pressure increases and the pressurizer sprays actuate, leading to further filling of the pressurizer. SG liquid levels, which have been steadily dropping since the termination of the MFW flow, soon reach the low SG level reactor trip setpoint. This initiates a reactor scram which ends the short-term heatup phase of the event. The reactor trip and subsequent cooling of the reactor coolant act to reduce the fluid expansion and prevent pressurizer overfill.

The automatic turbine trip at reactor scram and the continuing primary-to-secondary transfer of the decaying core power and the RCP heat (for cases with offsite power available) cause SG pressures to rapidly increase. When SG pressures become high enough, the steam dump system and the ADVs (or, if they are not available, the MSSVs) serve to limit the increase in SG pressure.

SG levels continue to drop and soon reach the low-low SG level AFW actuation setpoint. When the delivery of AFW begins, the rate of level decrease in the fed SGs slows. If AFW flow is

[a] The OTΔT trip applies to Westinghouse designed PWRs.

[b] For this sample problem, the difference between these two setpoints is only 2.5% of the instrument span and the time difference is negligible.

sufficient to prevent dryout in the SGs then, as the decay heat rate diminishes, liquid levels in the SGs stabilize and begin to rise. Reactor coolant temperatures also stabilize and begin to decrease, marking the end of the challenge to the event acceptance criteria.

Definition of Events Analyzed

The objective of this analysis is to demonstrate the adequacy of the SG level setpoints and the AFW capacity to avoid the expulsion of liquid from the PORVs and pressurizer safety valves and assure long-term cooling capability to a safe shutdown condition.

There are four potential acceptance criteria that could apply: 1) the DNB SAFDL, 2) the FCM SAFDL, 3) the pressure limit, and 4) the plant condition restriction (event must not generate a more serious plant condition without other faults occurring independently). For the short-term heatup phase, the MDNBR is bounded by the LOCF event, and for the long-term heat-up phase, the DNB SAFDL is not challenged, provided that the SGs retain liquid inventory (or the reactor coolant subcooling margin satisfies the plant-specific criterion). The FCM criteria is bounded by other Condition II events and is not credibly challenged by this event.

The peak primary and secondary pressures for this event are less than those of the LOEL/TT events provided that the pressurizer retains a steam "bubble" for pressure control, that is, the pressurizer does not overfill. Finally, the plant condition restriction is satisfied if the pressurizer does not become so full that liquid is expelled through the PORVs (the pressurizer level remains below the PORV inlet piping penetrations). In summary, the acceptance criteria for this event reduce to the requirements that: 1) the pressurizer level must remain below the PORV inlet piping penetrations, and 2) the fed SGs must not dry out (or the reactor coolant subcooling margin must satisfy the plant-specific criterion).

Consequently, the plant state and RPS setpoints are conservatively biased to maximize the potential for pressurizer overfill and SG dryout. Thus, a number of event specific analysis conservatisms are applied in addition to the more general ones that are routinely applied. [

]

Analysis Results

The event is initiated by tripping both MFW pumps for the two SGs. The liquid levels of both SGs drop rapidly and at 27.45 seconds, a low SG level signal trips the reactor. The sequence of events for the transient is presented in Table 6.5 and the transient responses of key parameters are presented in Figure 6.34 through Figure 6.39. For code-to-code comparisons, ANF-RELAP results are included in the figures.

There is a large margin to pressurizer overfill. Both codes predicted the maximum pressurizer level to be at 70.6 percent of the span and the top of the span is approximately 3.5 feet below the PORV inlet piping penetrations. Similarly, both codes predicted that the AFW flow capacity was sufficient to arrest the SG level decrease and prevent dryout so that long-term cooling was assured. However, the minimum calculated SG inventory was somewhat different, with S-RELAP5 giving a value of 20.1 percent (relative to initial inventory) while ANF-RELAP gave a value of 27.4 percent. While there are minor differences in some of the other variables (e.g., RCS fluid temperatures), the SG inventory is the one significant difference and is addressed here.

The difference in minimum predicted SG inventory is about 7.3 percent of the initial inventory as shown in Figure 6.39. S-RELAP5 calculates a larger reduction in SG inventory primarily because of a delay in the reactor trip of almost 5 seconds. During this 5 second period, the S-RELAP5 calculation continues at full power with the consequent boil-off of SG inventory as all of the reactor heat is absorbed by the latent heat of the SG residual mass. This calculated scram delay accounts for about 90 percent of the difference in minimum SG inventory with the remaining 10 percent due to the difference in RCP energy deposition as discussed above.

In both the S-RELAP5 and ANF-RELAP calculations, the reactor tripped on low SG level. The reason that the S-RELAP5 trip occurred later in time is due to the initial distribution of liquid within the SG secondary side which in turn is a result of differences in the interfacial drag package between the two codes. At the initial steady-state conditions, the SG inventory for both calculations is the same. However, for S-RELAP5, more water is present in the boiler so downcomer loss coefficients were adjusted to reduce the recirculation ratio allowing the initial mass to be matched.

Conclusion

The S-RELAP5 calculated results are shown to be in general agreement with the ANF-RELAP calculated results and reasonably represent the plant transient, with a negligible difference in the maximum pressurizer level and approximately a 7 percent difference in SG minimum inventory. This difference in SG inventory is the result of S-RELAP5 predicting the scram time approximately 5 seconds later than ANF-RELAP. For both codes, the reactor trip occurred on low SG level, however, differences between the codes' interfacial drag packages led to a difference in the predicted water holdup and in trip timing.

The capacity of the AFW system was shown to be more than adequate to allow a safe and orderly plant shutdown and to prevent SG dryout. Since SG dryout does not take place, the LONF event does not result in the violation of SAFDLs.

Table 6.5 LONF Event With Offsite Power Available Event Summary

Event	Time (s)
MFW Valve Closes	0.0
Pressurizer Spray On	18.0
Low SG Level Reached	26.55
Reactor Trip on Low SG Level Signal	27.45
Turbine Trip	28.2
Control Rods Begin to Fall	28.2
MSSVs Open	30.0
Pressurizer Backup Heaters On	30.1
Maximum Pressurizer Level	31.0
Pressurizer Proportional Heaters On	43.0
AFW Flow Starts	197.0
Maximum Pressurizer Pressure	1100
Pressurizer PORVs Open	1100
Pressurizer Backup Heaters Off	1220
Pressurizer Proportional Heaters Off	1230
Pressurizer PORVs Close	1250
Minimum Inventory – SG 1	1890
Minimum Inventory – SG 2	1900

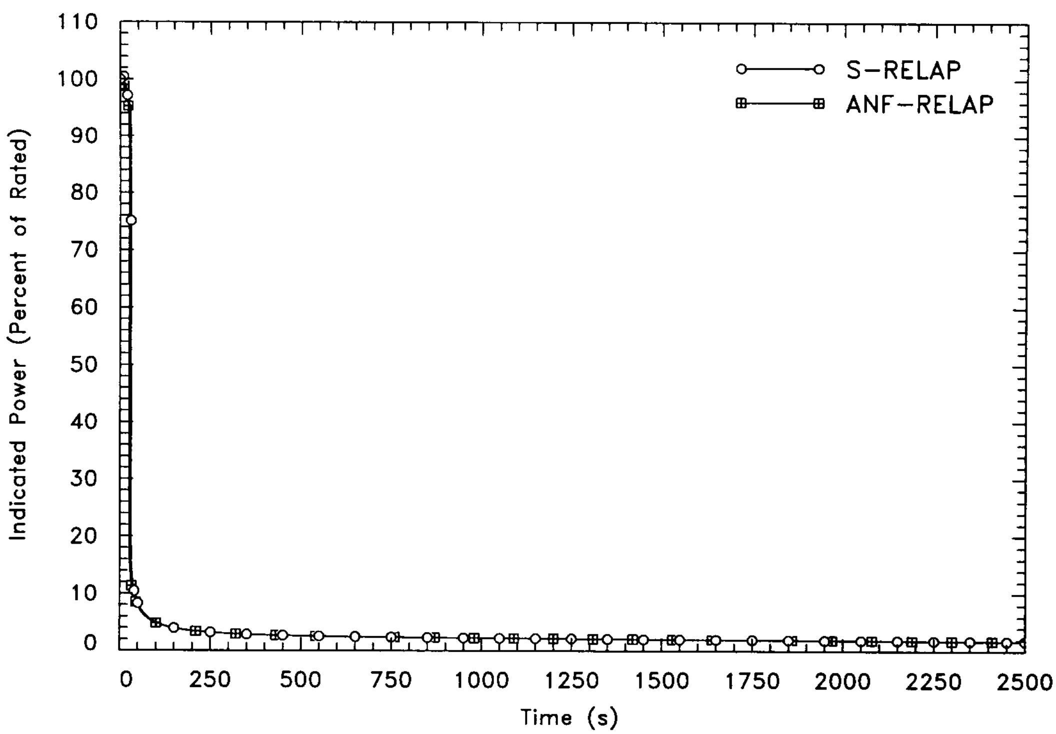

Figure 6.34 LONF (With Offsite Power) Reactor Power Level

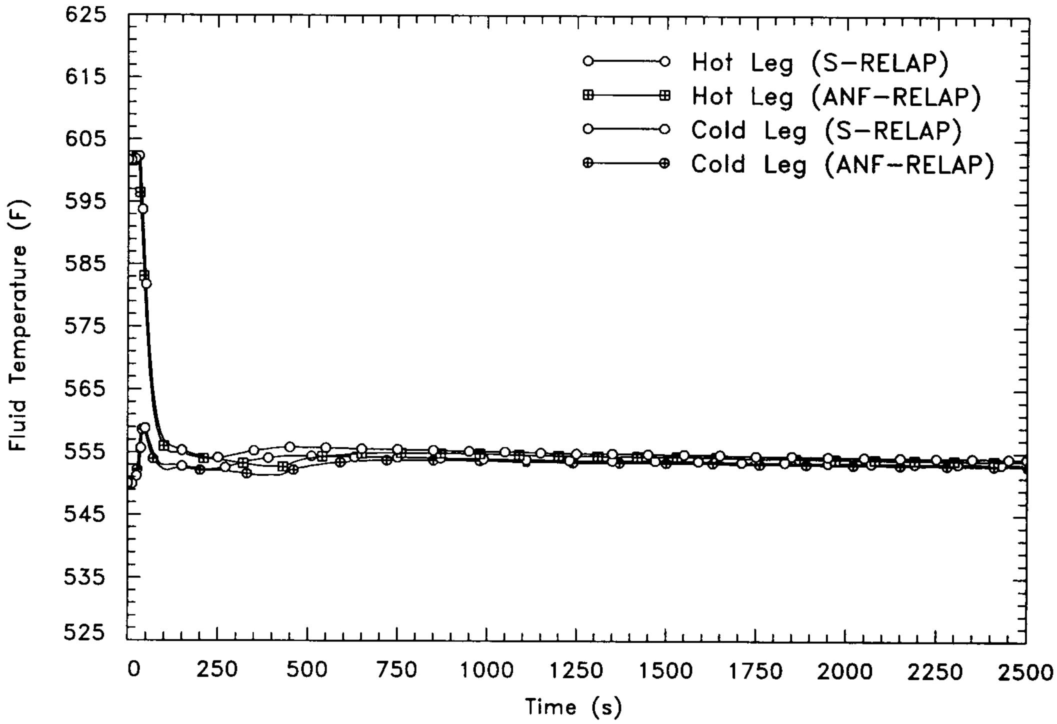

Figure 6.35 LONF (With Offsite Power) RCS Temperatures

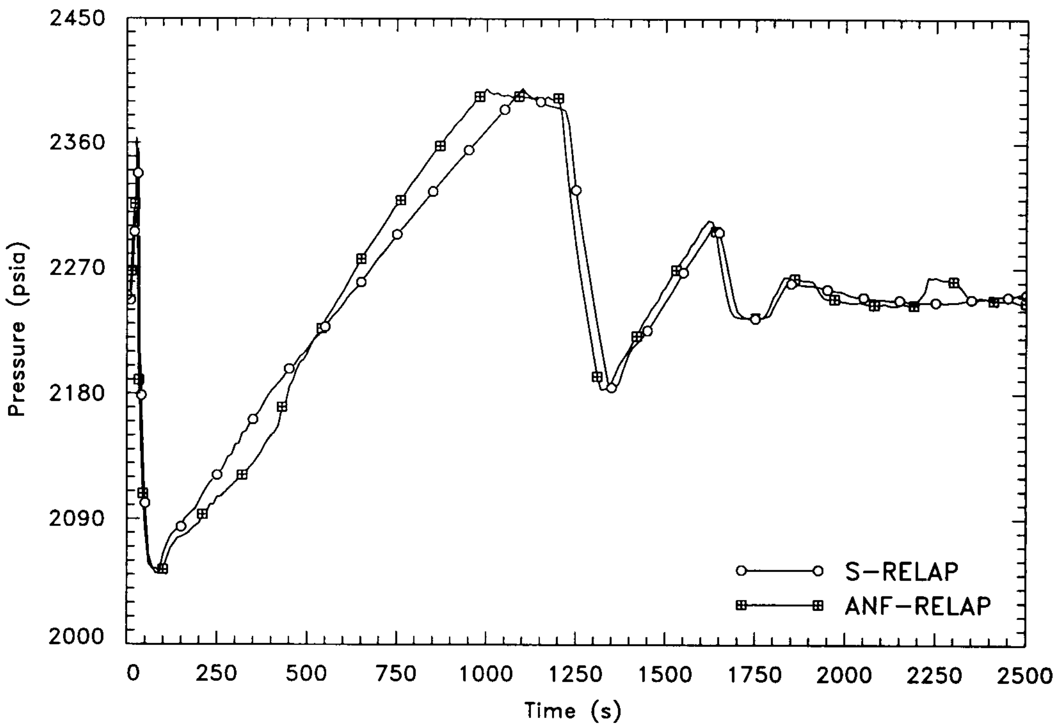

Figure 6.36 LONF (With Offsite Power) Pressurizer Pressure

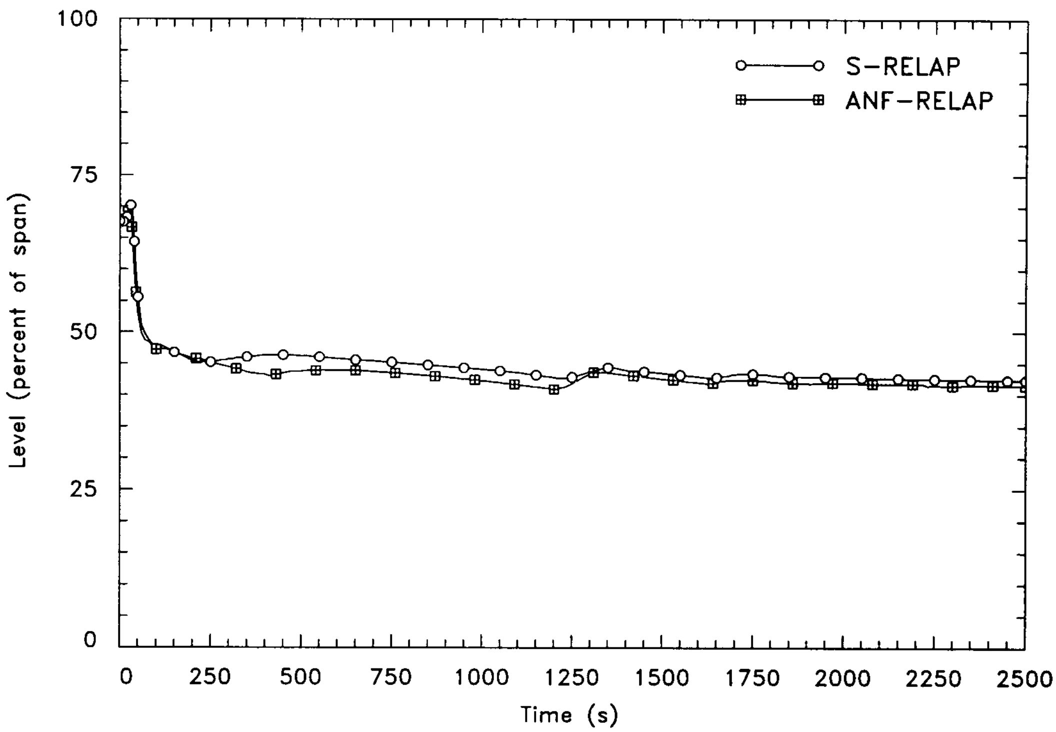

Figure 6.37 LONF (With Offsite Power) Pressurizer Liquid Level

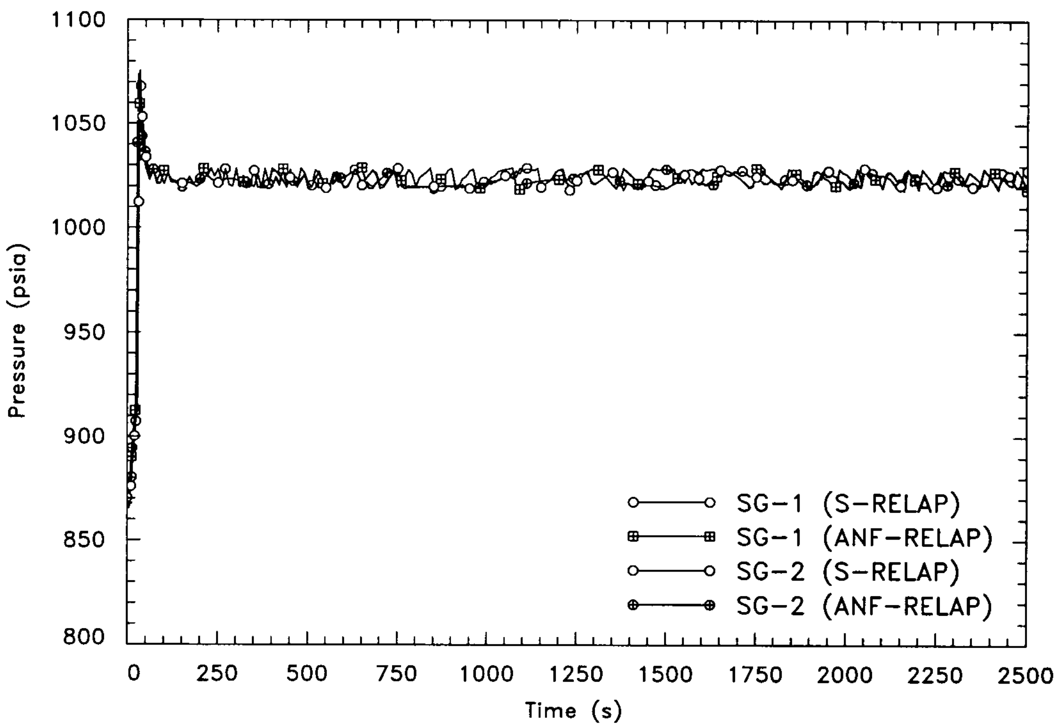

Figure 6.38 LONF (With Offsite Power) Steam Generator Pressure

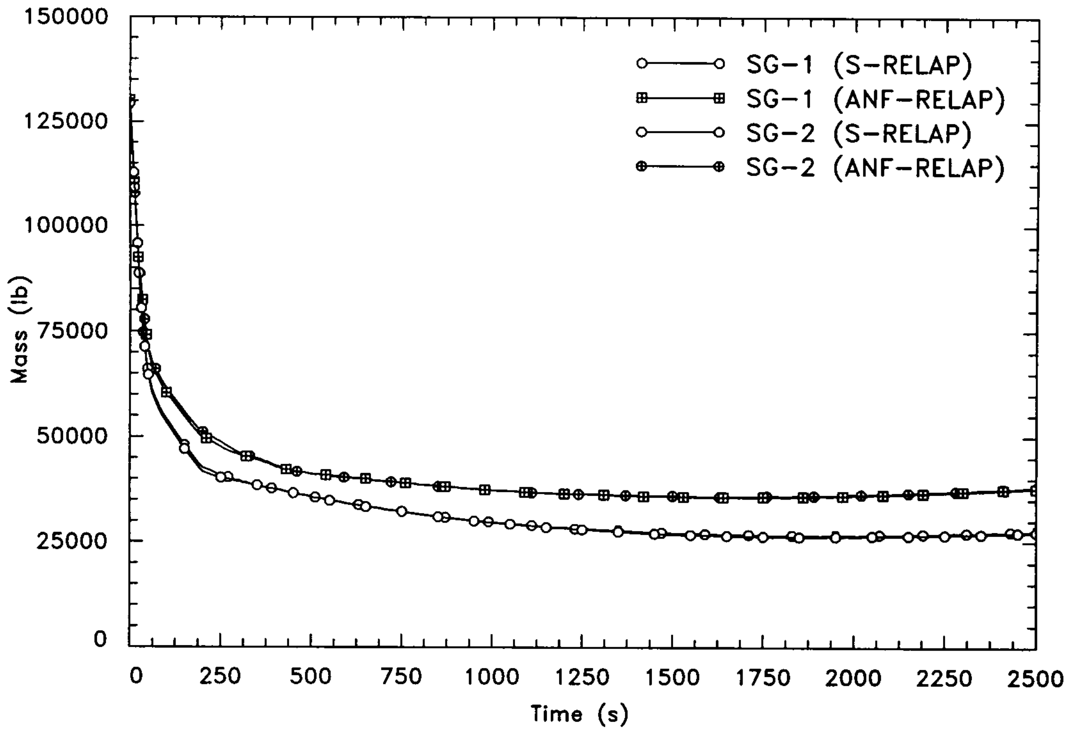

Figure 6.39 LONF (With Offsite Power) Steam Generator Inventory

6.5 *Loss of Forced Reactor Coolant Flow (LOCF)*

Event Description

The LOCF transient is initiated by a disruption of the electrical power supplied to, or a mechanical failure, in an RCP. These failures may result in a complete or partial loss of forced coolant flow. The complete LOCF with scram on low flow rate is the most limiting transient, from the perspective of challenge to the DNB SAFDL. This scenario occurs when an under-frequency or under-voltage event causes the RCPs to trip without removing power from the control rod restraints. Furthermore, between the time when the RCPs trip and the time when their breakers trip, the RCPs act as generators and an electrical braking occurs, accelerating the coastdown.

The impact of losing one or more RCPs is a decrease in the active coolant flow rate in the reactor core and, consequently, an increase in core temperatures. The reactor trips on low flow. Prior to reactor trip, the combination of decreased flow and increased temperature poses a challenge to the DNB SAFDL. The FCM SAFDL is not challenged since there is no significant increase in core power. This event also produces an increase in system pressure due to increased temperatures and reduced heat transfer to the secondary side of the SGs, but it does not create a credible challenge to system pressure limits.

This event is terminated by reactor scram on the RCS low flow trip, and the purpose for analyzing this event is to verify that the RPS can respond fast enough to prevent violation of the DNB SAFDL.

Definition of Events Analyzed

The partial loss of coolant flow event is a less severe transient than the complete loss of coolant flow event. This sample problem simulates a complete loss of coolant flow event.

The issue being evaluated is the challenge to the DNB SAFDL. Therefore the plant state and trip points are biased so as to maximize this challenge. This event is analyzed from full power initial conditions and the core thermal margins are minimized. [

]

Analysis Results

The overall response of the primary and secondary systems for this event is calculated by S-RELAP5. The MDNBR for this event is calculated using the thermal-hydraulic conditions from S-RELAP5 as input to XCOBRA-IIIC.

The transient is initiated by tripping all four RCPs. As the pumps coast down, the core flow is reduced causing a reactor scram on low flow. The flow decrease causes reactor coolant temperatures to increase with a subsequent power rise due to moderator reactivity feedback. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperatures. Using XCOBRA-IIIC, the MDNBR is calculated to be 1.58.

The sequence of events is given in Table 6.6. The responses of key system variables for this event are given in Figure 6.40 to Figure 6.45. For code-to-code comparisons, the ANF-RELAP predictions are included on the figures.

The key parameter is the MDNBR and both codes predicted the MDNBR to be well above the applicable DNB SAFDL of 1.164. The predicted response for most of the key system variables is nearly identical. However, the MDNBR calculated by XCOBRA-IIIC using S-RELAP5 results was about 2.5 percent higher than that using the ANF-RELAP results.

The cause of this difference in the predicted MDNBR is the calculated behavior of the flow coast down. As shown in Figure 6.45, the RCS flow rate calculated by S-RELAP5 degrades somewhat more slowly than that of ANF-RELAP. At the time of MDNBR, about 3.1 seconds for both codes, the RCS flow rate is about 3 percent higher for the S-RELAP5 calculation. The root cause for this difference in transient response is the increased wall drag inside the SG tubes for S-RELAP5 due to the improvement to the single-phase wall drag model. Specifically, due to the increased pressure drop for the RCS (about 10 percent higher), the initial pump speed in S-RELAP5 is higher than the pump speed for ANF-RELAP and the ensuing flow coast down is slightly slower for S-RELAP5.

Conclusion

The results of the analysis demonstrate that S-RELAP5 provides a satisfactory representation of the event. Furthermore, the S-RELAP5 results are in close agreement with the ANF-RELAP results, because most of the predicted responses for key system variables are virtually indistinguishable. The largest predicted variation is in the XCOBRA-IIIC MDNBR based on S-RELAP5 and ANF-RELAP results and has a magnitude of 3.0 percent; the DNB margin is about 36 percent above the applicable limit of 1.164.

Since the predicted MDNBR is greater than the applicable safety limit, this result indicates that no fuel failures due to DNB would occur. Therefore, the event acceptance criteria are met.

Table 6.6 LOCF Event Summary

Event	Time (s)
RCPs Trip	0.0
RCP Breakers Trip	0.5
Flow Reaches Low Flow Trip Setpoint	0.8
Peak Power Occurs	2.5
Reactor Scram (Begin Rod Insertion)	2.6
Turbine Isolates (Stop Valve Closed)	2.6
Pressurizer Spray Actuates	3.1
MDNBR	3.1
Peak Pressurizer Pressure	5.8

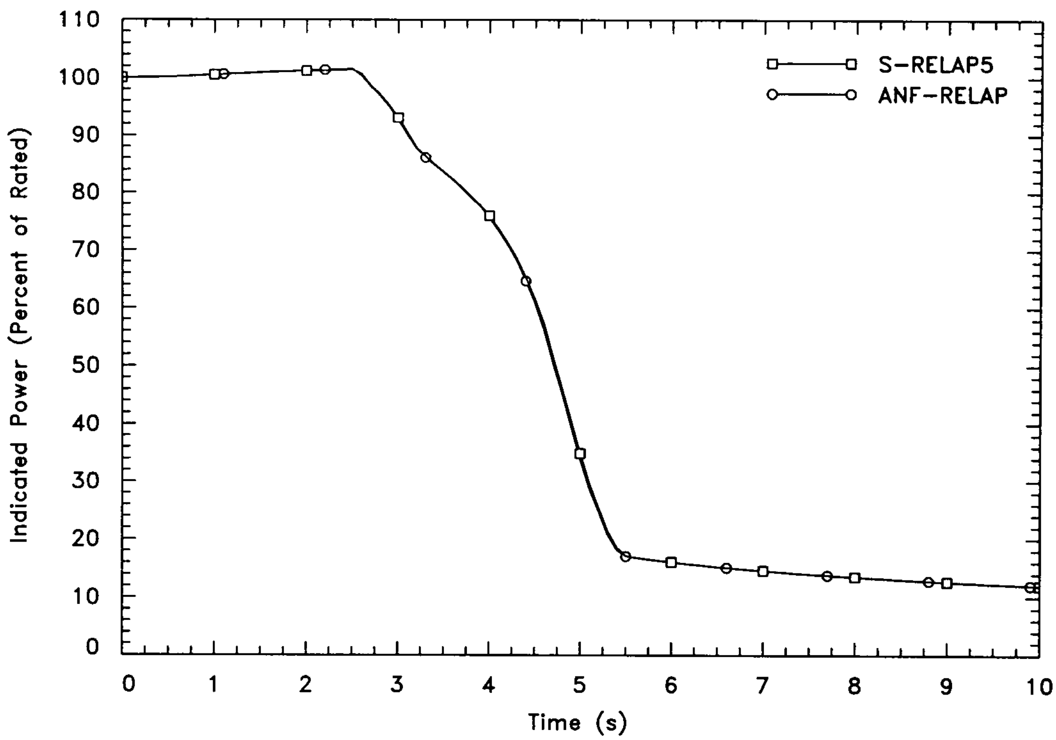

Figure 6.40 LOCF Reactor Power Level

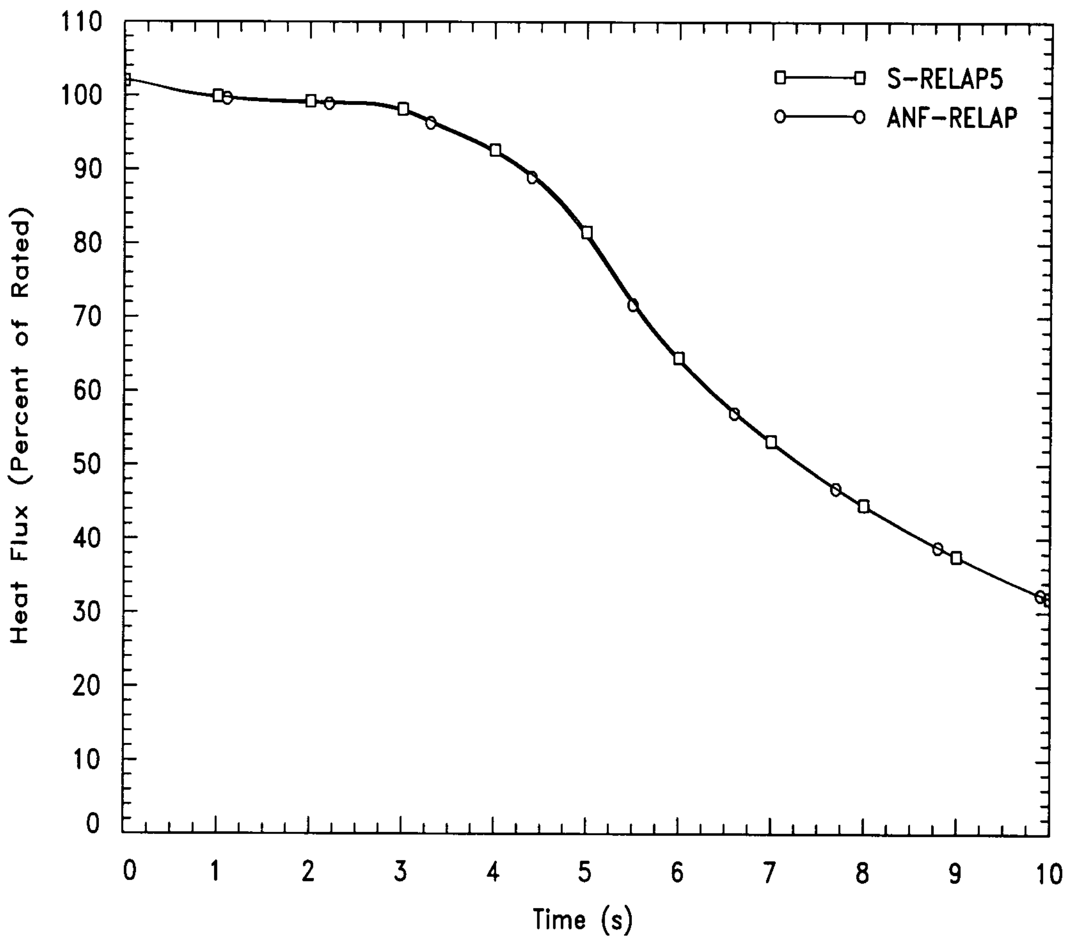

Figure 6.41 LOCF Core Average Heat Flux

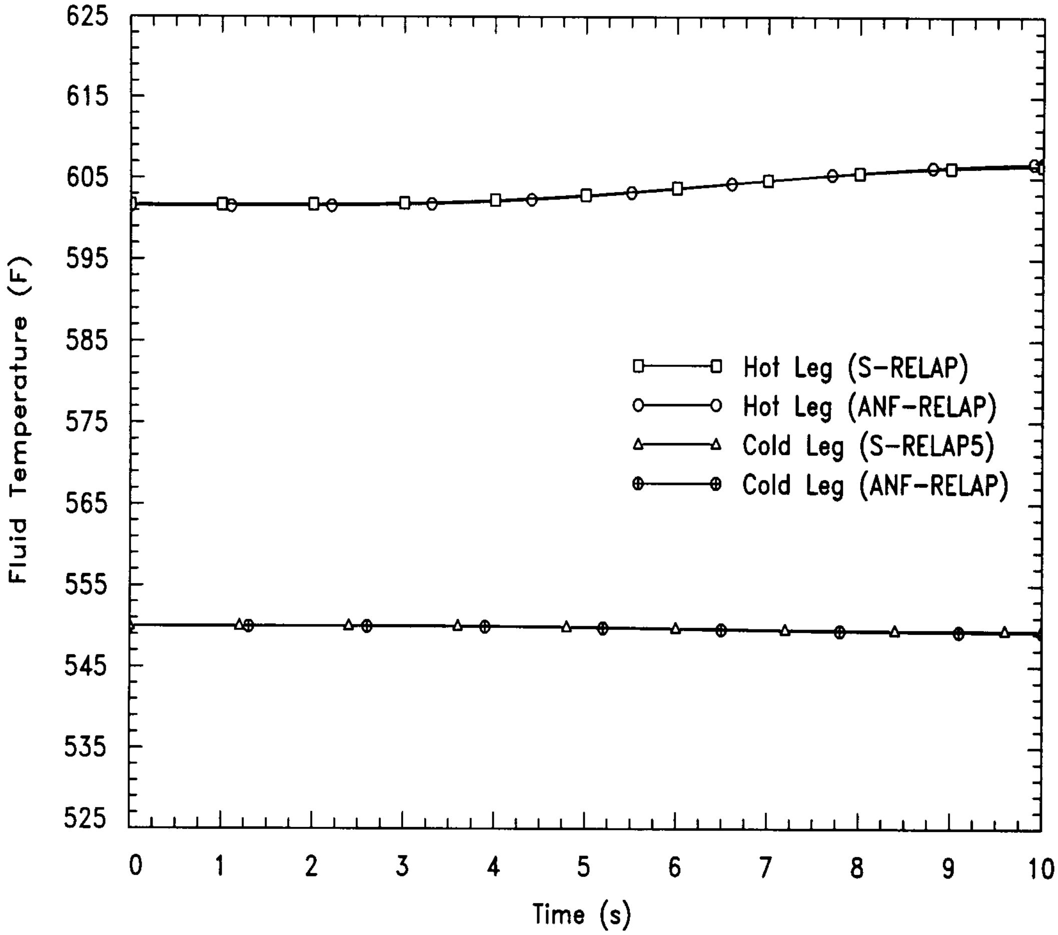

Figure 6.42 LOCF RCS Temperatures

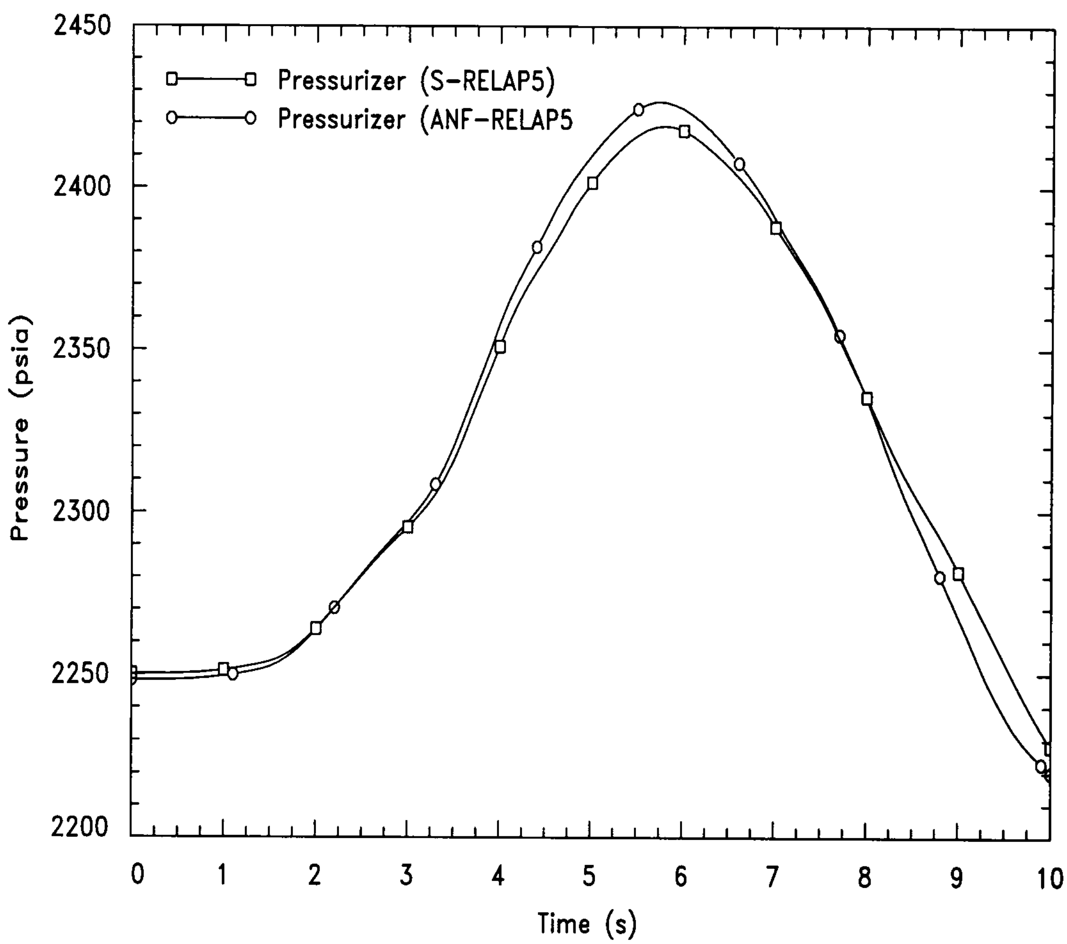

Figure 6.43 LOCF Pressurizer Pressure

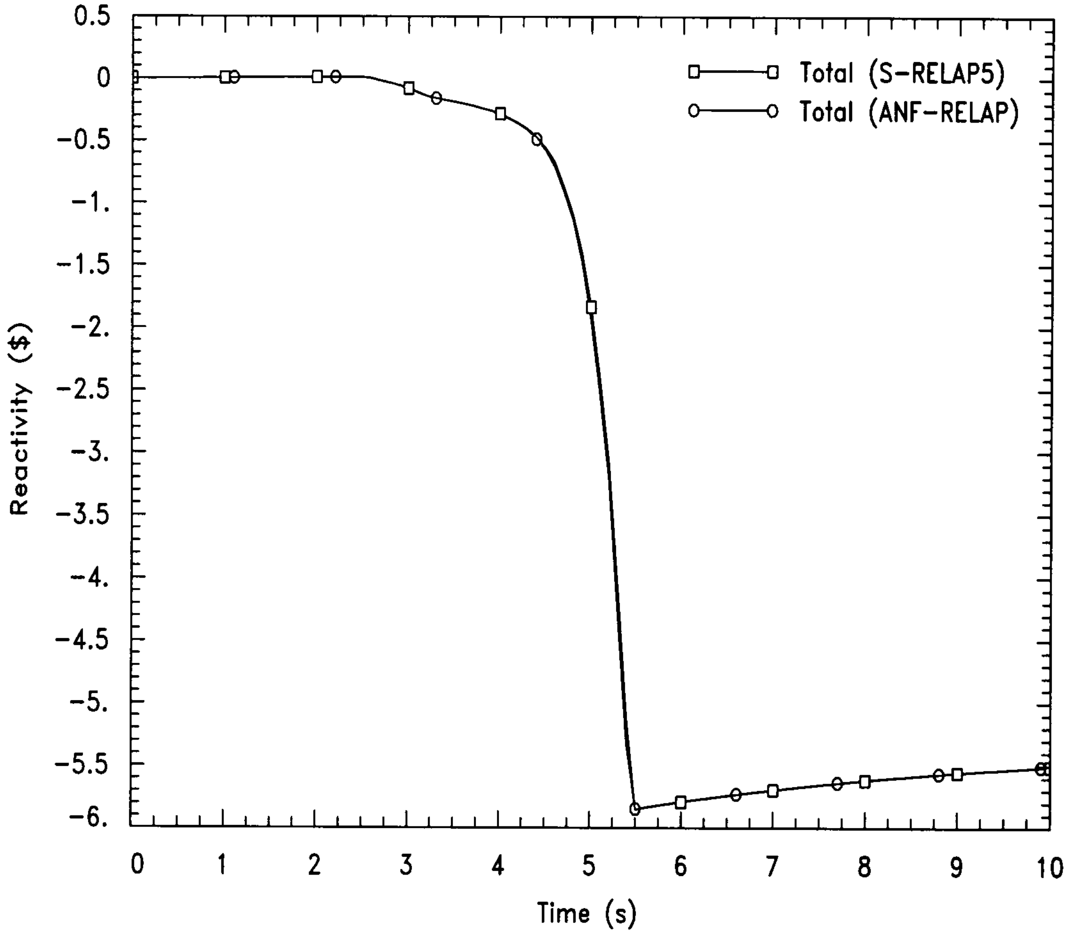

Figure 6.44 LOCF Reactivity

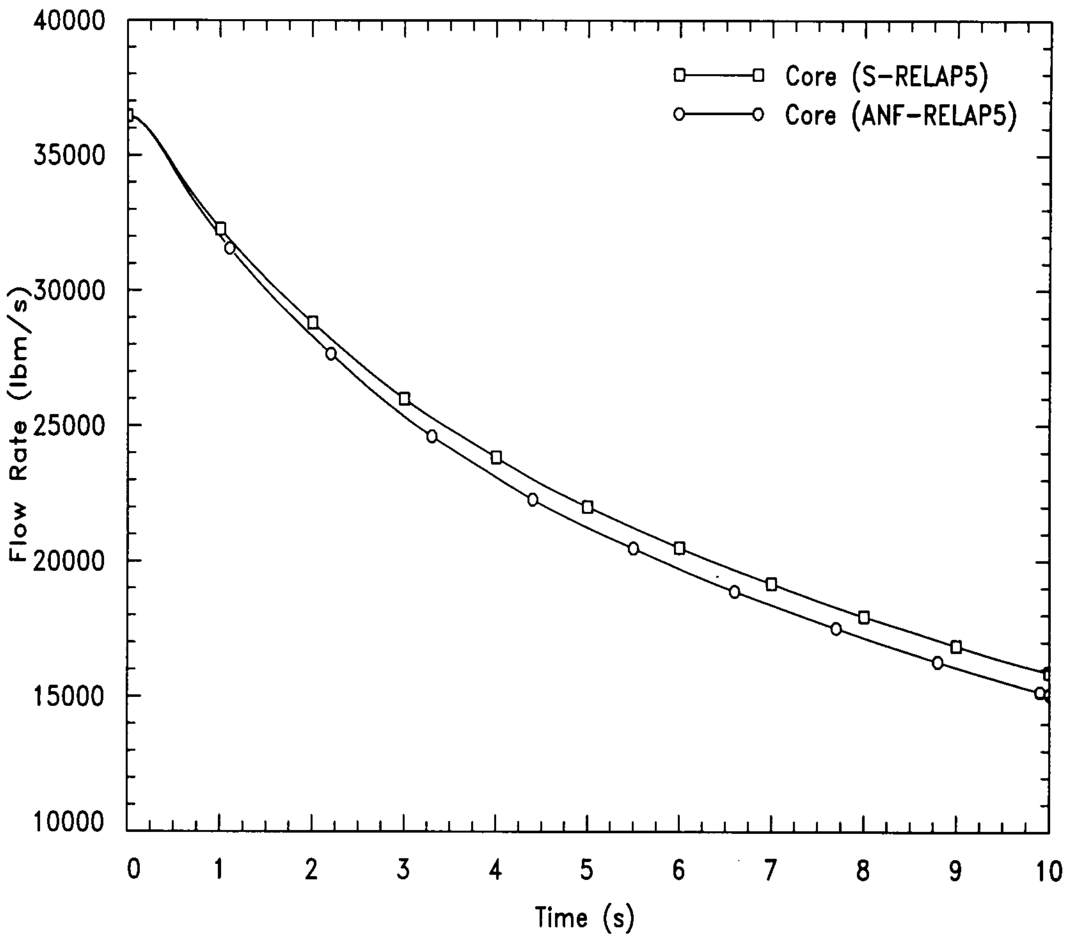

Figure 6.45 LOCF Reactor Coolant Flow Rate

6.6 *Uncontrolled Control Rod Bank Withdrawal (UCBW) at Power*

Event Description

This event is initiated during power operation (mode 1) by an uncontrolled withdrawal of a control rod bank due either to a failure in the rod control system or to operator error. The positive reactivity addition results in a power transient, increasing the core heat flux and creating a challenge to the DNB margin. The DNB margin is further reduced by an increase in the reactor system temperature resulting from the power-cooling mismatch, due to the increased energy generation rate in the core.

The RPS is designed to terminate this transient before the DNB limits are reached. The principal protective trips in this case for the sample plant are the VHP trip and the TM/LP trip. The TM/LP trip is specifically designed to protect against DNB for slow transients where the coolant temperature is able to respond to the reactor power changes. One of the primary objectives of this event analysis is to check the adequacy of the TM/LP setpoint algorithm.

The trip margin to DNB for the TM/LP trip decreases as the reactivity insertion rate increases due to thermal inertia and trip delay. This decrease in DNBR continues with reactivity insertion rate increases until the point where the neutron power challenges the VHP trip. MDNBR is typically found close to where the two trips act simultaneously and occurs just after control rod insertion begins.

Definition of Events Analyzed

This analysis evaluates the consequences of an uncontrolled control rod bank withdrawal from full power conditions. (CEA bank withdrawals at lower power levels, with correspondingly lower VHP reactor trip setpoints, offer less challenge to the DNB acceptance criterion and, therefore, have not been evaluated in this analysis.) A matrix of cases considering a range of reactivity insertion rates, from very slow (e.g., gradual boron dilution) to the maximum possible CEA bank withdrawal rate at maximum worth for two banks moving in normal sequence and overlap, and at BOC and EOC was calculated. Only the most limiting DNBR case is described here. The limiting DNBR case occurred for a slow CEA bank withdrawal rate (3.30×10^{-4} \$/s) at BOC conditions.

[

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Analysis Results

The overall response of the primary and secondary systems for this event is calculated by S-RELAP5. The MDNBR for this event is calculated using the thermal-hydraulic conditions from the S-RELAP5 calculation as input to XCOBRA-IIIC.

The DNB-limiting uncontrolled control bank withdrawal transient was analyzed for full power conditions (102 percent of rated) with BOC kinetics and with an insertion rate of 3.30×10^{-4} \$/s. The MDNBR was calculated to be 1.50. The scram occurred on a VHP trip near the point where the two trips would have acted simultaneously (the TM/LP trip signal would have been received 0.5 seconds after the time of the VHP signal).

An event summary is presented in Table 6.7. The transient responses for key parameters are presented in Figure 6.46 through Figure 6.52. For code-to-code comparisons, the ANF-RELAP predictions are included on the figures.

The focus of this reactivity insertion event is the challenge to the DNB SAFDL resulting from the power increase. The MDNBR calculated for the DNB-limiting transient was 1.50 which is well above the applicable limit of 1.164. The predicted response of the key system parameters that govern DNBR (e.g., see Figure 6.49 for the RCS fluid temperatures) was essentially identical for the two codes up until the time of the reactor trip.

The only significant difference is in the behavior of the pressurizer pressure, see Figure 6.50, after the PORVs open. The sensitivity of the control logic that governs the opening/closing of the motor valve used to model the PORVs, as noted in the LOEL sample problem, and small differences in the calculation, cause the differences in the predictions of the two codes to be magnified. However, the effect on the MDNBR is minimal.

Conclusion

The S-RELAP5 results are nearly identical to those of ANF-RELAP and reasonably represent the plant transient response. The MDNBR was calculated to be 1.50 and the applicable safety limit is 1.164. This indicates that no fuel failures due to DNB would occur and, therefore, that the acceptance criteria are met.

Table 6.7 DNB-Limiting UCBW at Power Event Summary

Event	Time (s)
Slow CEA Bank Withdrawal Begins	0.0
Pressurizer Spray On	47.0
Pressurizer PORVs Open	151.0
SG 1 MSSVs Open	202.0
SG 2 MSSVs Open	204.0
Indicated Power Reaches VHP Setpoint	208.7
Pressurizer Pressure Reaches TM/LP Setpoint	208.7
Peak Core Power Occurs	209.0
VHP Signal Initiates Reactor Trip	209.1
Scram CEA Insertion Begins, and Turbine Trips	209.9
MDNBR Occurs	210.0
Pressurizer PORVs Close	214.0

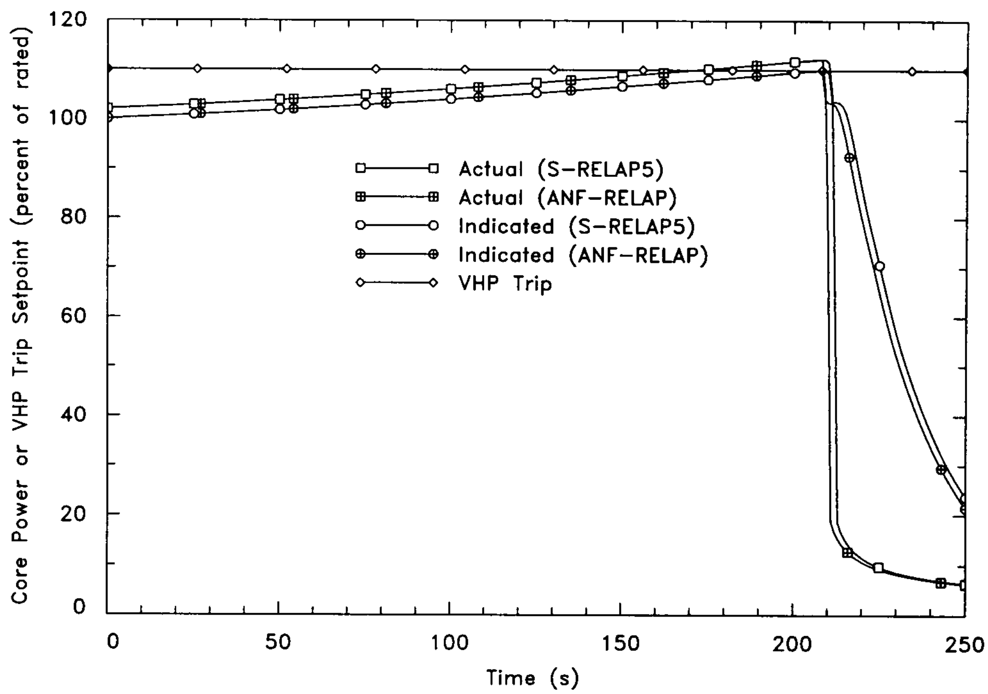

Figure 6.46 DNB-Limiting UCBW Core Power and VHP Trip Setpoint

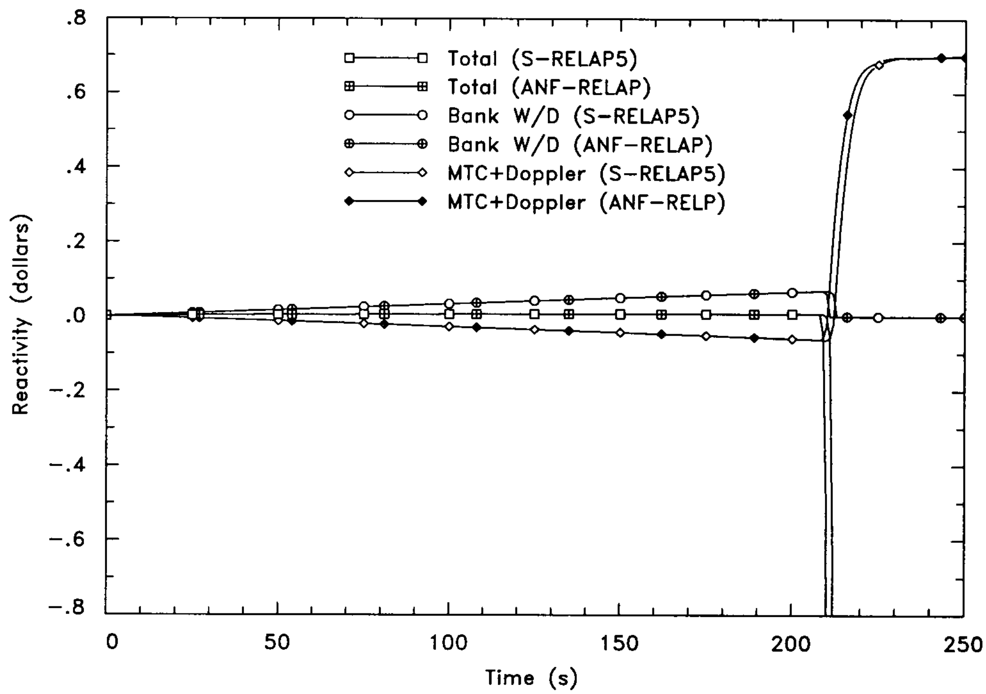

Figure 6.47 DNB-Limiting UCBW Reactivity

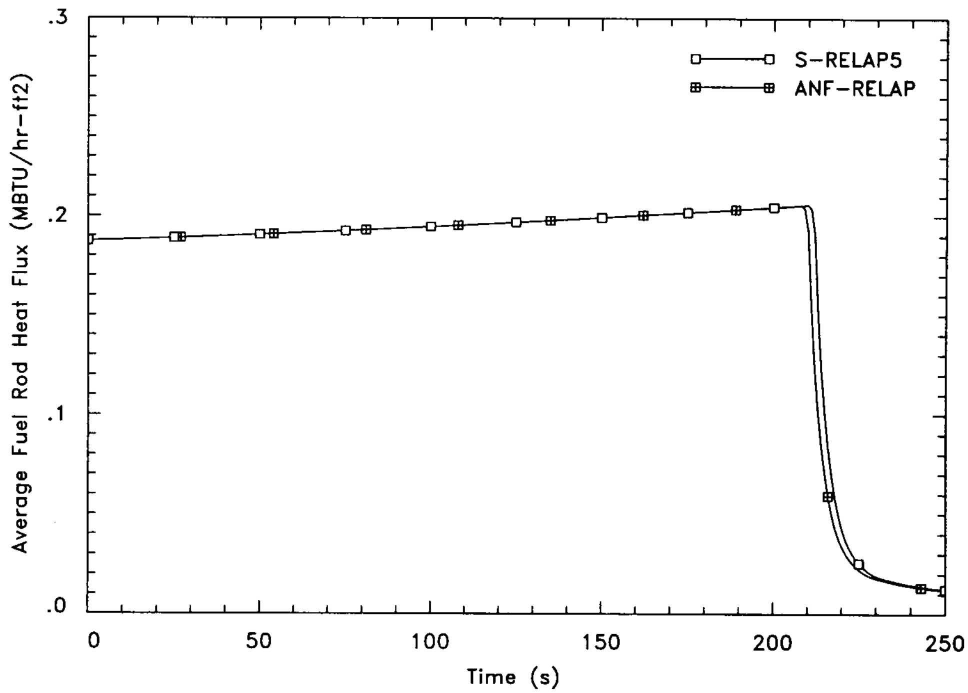

Figure 6.48 DNB-Limiting UCBW Average Fuel Rod Heat Flux

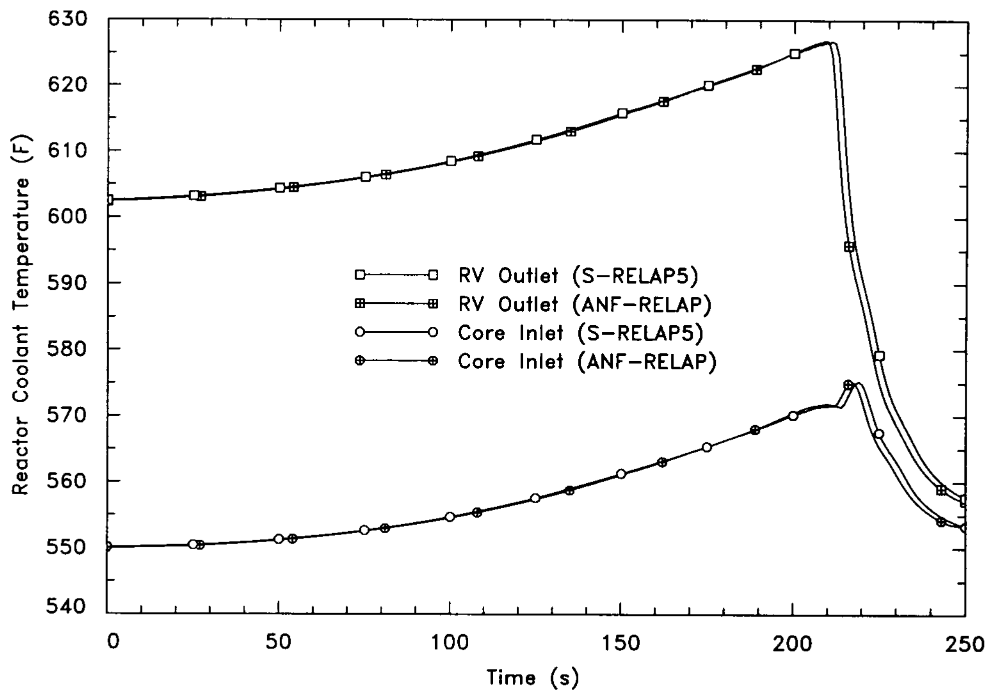

Figure 6.49 DNB-Limiting UCBW RCS Temperatures

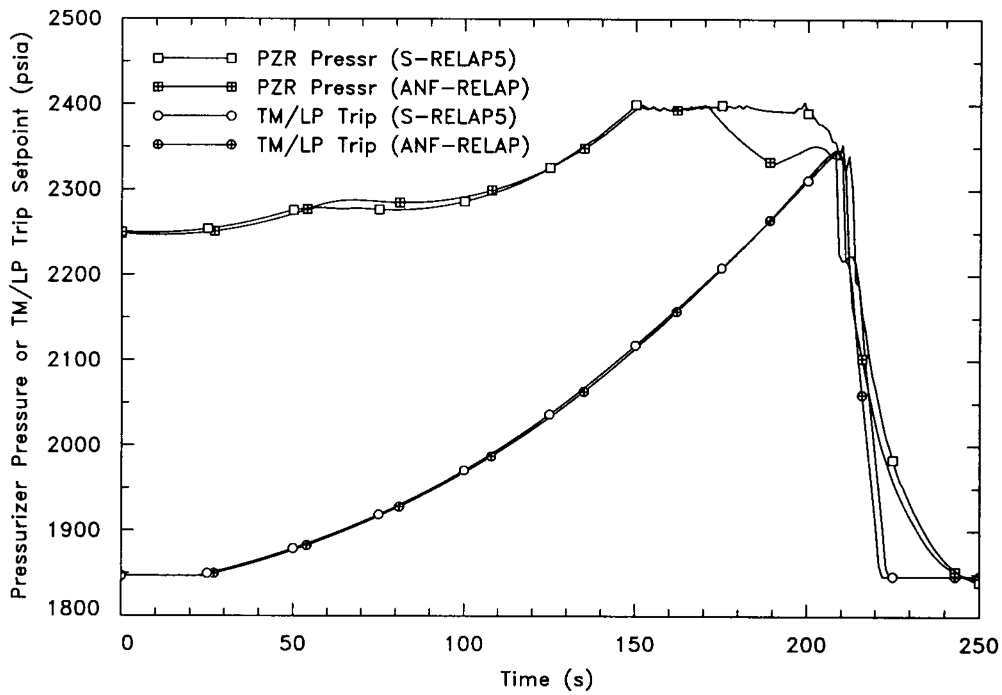

Figure 6.50 DNB-Limiting UCBW Pressurizer Pressure and TM/LP Trip Setpoint

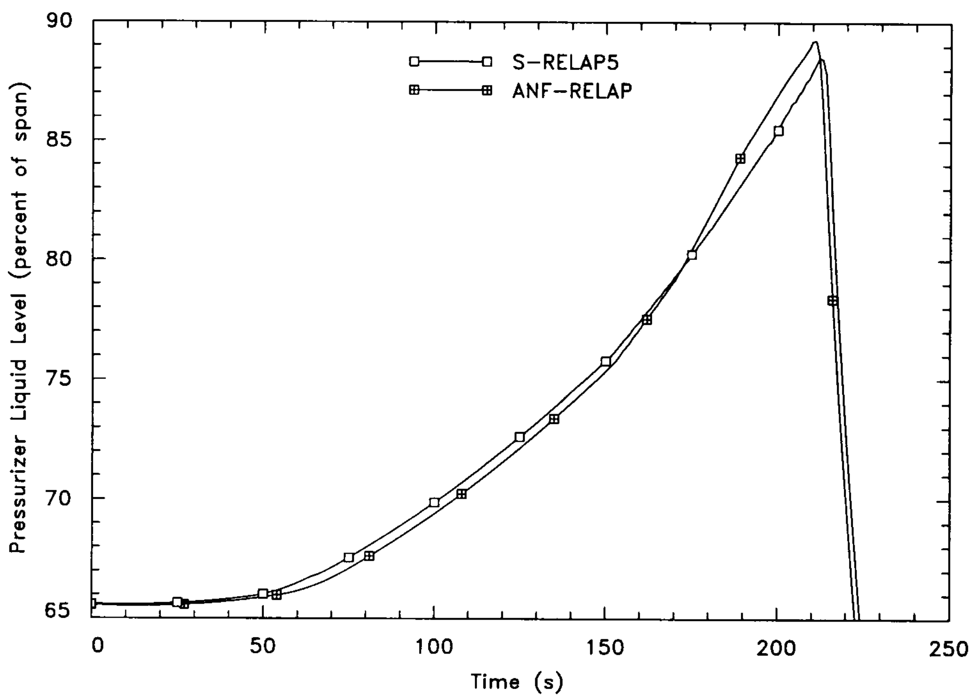

Figure 6.51 DNB-Limiting UCBW Pressurizer Liquid Level

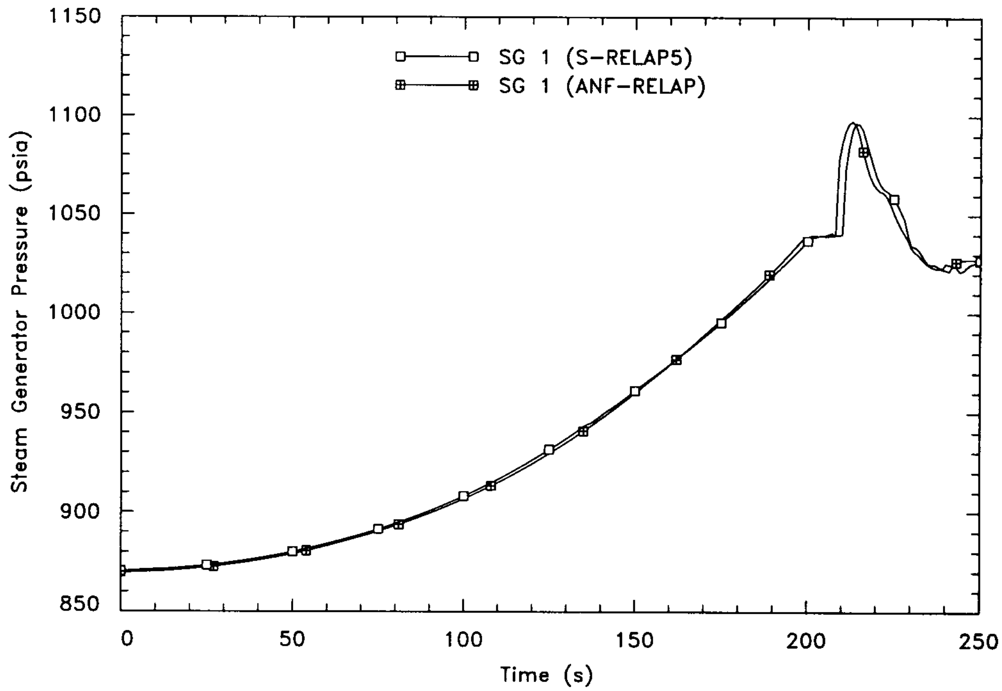

Figure 6.52 DNB-Limiting UCBW Steam Generator Pressure

6.7 *Steam Generator Tube Rupture (SGTR)*

Event Description

The SGTR event is initiated by a break of a single steam generator U-tube. RCS inventory begins to flow into the SG secondary side due the pressure differential. The break flow exceeds the make-up capacity of the charging pump causing the pressurizer pressure and level to decrease, leading to a reactor trip on the low pressure setting of the TM/LP trip. The trip of the reactor is followed by a turbine/generator trip so that the secondary side pressurizes and inventory from the RCS and the SG is released by the MSSVs. HPSI flow is initiated by the low-pressurizer-pressure signal. The reactor coolant pressure falls to saturation and a quasi-static relief of decay heat by steam through the MSSVs occurs until the operators intervene.

In this sample calculation, operator actions (including a 30 minute delay for operator identification of event) were assumed for a typical CE 2x4 plant. These actions included isolating the AFW system and closing the MSIV of the ruptured SG. Then, the operators used the ADVs and pressurizer PORVs to reduce the RCS pressure. Finally, the PORVs were cycled to regain control of the plant.

Events Analyzed

The initiator for this event is a double-ended break of a single steam generator U-tube in the downstream side just above the tube sheet. The event analyzed is initiated at HFP without offsite power available. This leads to a loss of power to the bus upon turbine trip. The initial plant state is also biased, based on technical specification limits and instrumentation uncertainties, to maximize the releases. The event summary, Table 6.8, describes the operator actions assumed.

For the purposes of this analysis, the SGTR event is considered terminated at 8000 seconds with the plant fully under operator control.

Analysis Results

Figure 6.53 through Figure 6.65 present the S-RELAP5 predicted response for key plant parameters. For code-to-code comparison, the ANF-RELAP predictions are also included on these figures. The sequence of events for the SGTR even is given in Table 6.8.

The key parameters affecting the radiological release are the cycling of the MSSVs and ADV for the affected steam generator. Upon turbine trip, the turbine admission valves are closed and the steam dump system is unavailable due to the loss of condenser vacuum. The result is a rapid increase in SG pressures up to the MSSV setpoints. The MSSVs cycle, releasing heat and inventory to the atmosphere. After 1800 seconds, the operator is assumed to take action to isolate the affected steam generator and begin a cooldown of the RCS. This cooldown includes opening the ruptured SG ADV in an effort to limit further actuation of its MSSVs.

For the SGTR event, the results of the system thermal-hydraulic code are used as boundary conditions for an analysis of the radiological consequences. The purpose of this analysis is to compare the predicted response of S-RELAP5 to that of ANF-RELAP for the parameters that are input to the radiological release model. Specifically, the parameters of interest are the total break flow and steam release for the ruptured SG. The total steam release from the ruptured steam generator is predicted to be 101,000 lb_m and the integrated break flow (for the entire 8000 seconds transient) is 168,000 lb_m. For these parameters, the agreement between S-RELAP5 and ANF-RELAP is excellent. The integrated break flow is within 1.5 percent and the total steam release from the ruptured SG (ADVs and MSSVs) is within 0.5 percent.

There are a number of small differences in the calculated values of the other system variables (e.g., the RCS temperatures) that did not have a significant impact on the course of the transient. The largest difference shows up in the predictions for the inventory for the unaffected SG, see Figure 6.59. For the unaffected SG, the total steam release for the ADVs and MSSVs is about 2.7 percent greater for ANF-RELAP. This magnitude is within the difference expected in the calculated critical flow, as described in Section 3.3.

Conclusion

The results demonstrate that S-RELAP5 provides a satisfactory representation of the SGTR event. Furthermore, the S-RELAP5 results were generally in close agreement with the

ANF-RELAP results for the response of key system variables. In particular, the predicted total steam release from the affected steam generator was within one percent for the two codes.

Table 6.8 SGTR Event Summary

Event	Time (s)
Double-Ended Rupture of SG Tube	0.0
Reactor Trips on TM/LP Signal	673.8
Turbine Trips, Loss of Offsite Power, RCP Coastdown, and MFW Trips	674.5
Low Pressurizer Pressure Trip of SIS	689.2
HPSI Flow Begins	719.2
AFW Flow Begins (Low SG Level Trip)	1031.6
Operator Action to Isolate AFW and MSIV to Ruptured SG	1800
Operator Opens ADVs on Both SGs	1800
Operator Isolates Ruptured SG ADV	3000
Operator Opens Pressurizer PORV	5000
Operator Closes Pressurizer PORV	5035
Operator Terminates HPSI and Charging Flow	6000
Operator Re-opens Pressurizer PORV	6000
Operator Closes Pressurizer PORV	6035
Operator has Full Control of Plant	8000

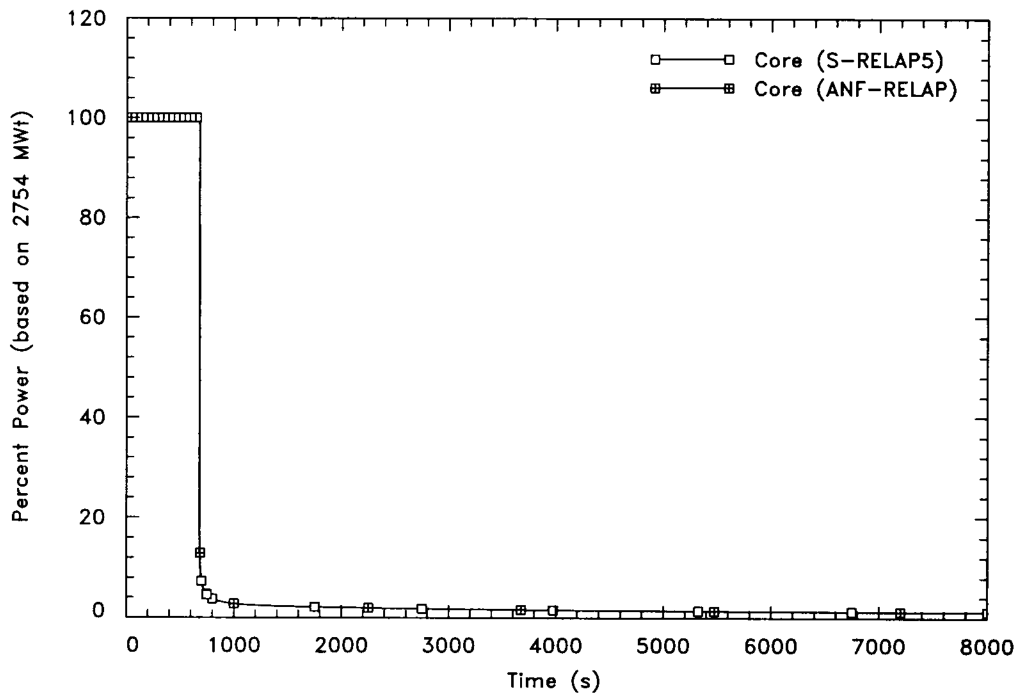

Figure 6.53 SGTR Reactor Power

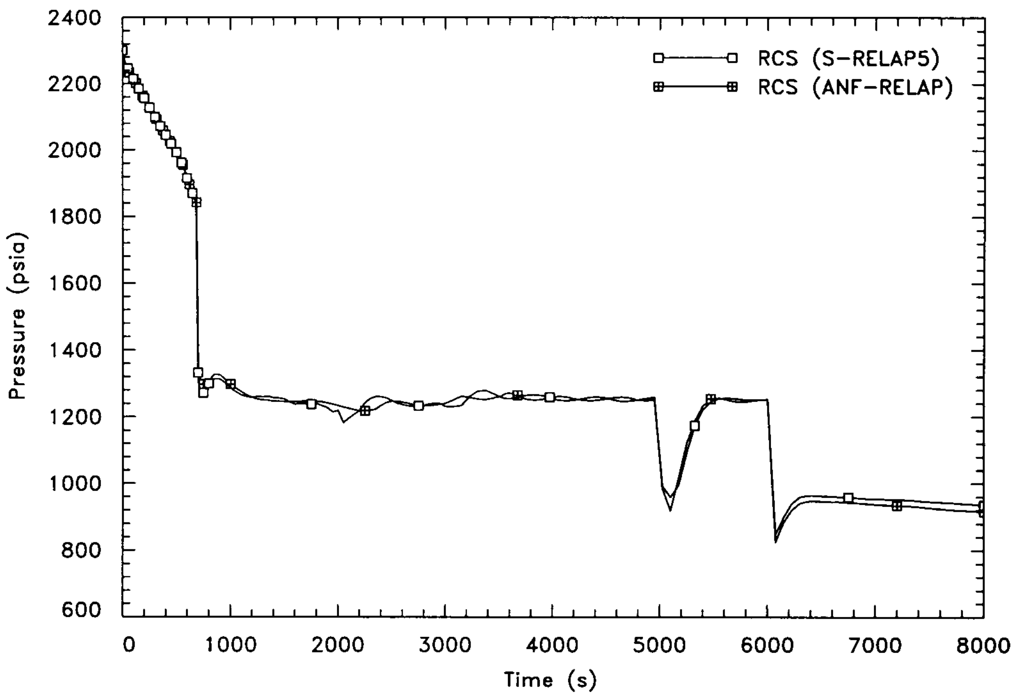

Figure 6.54 SGTR Pressurizer Pressure

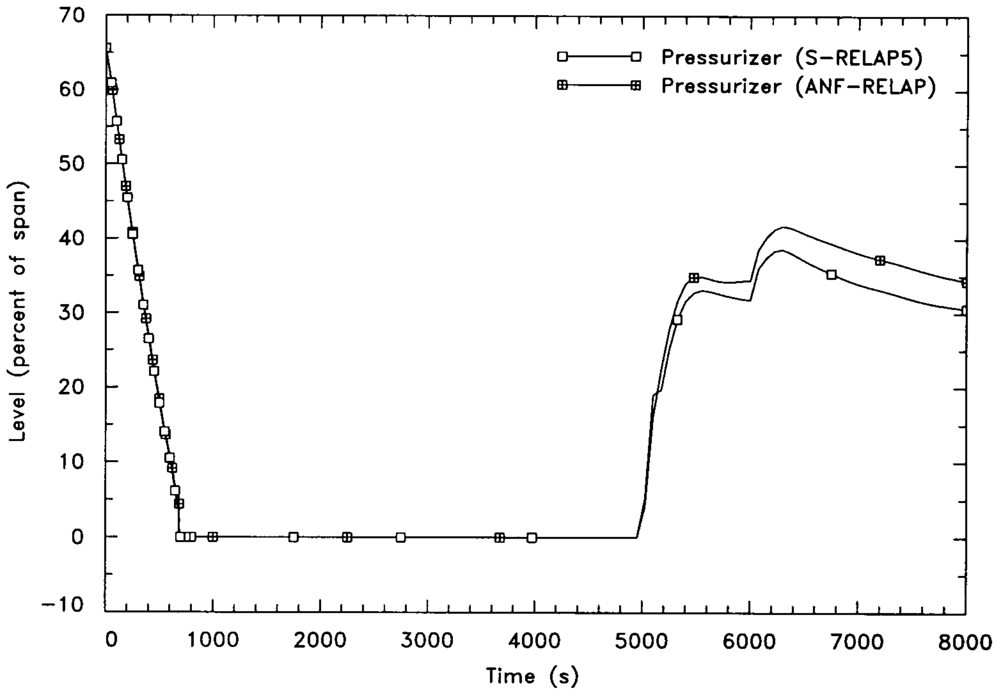

Figure 6.55 SGTR Pressurizer Liquid Level

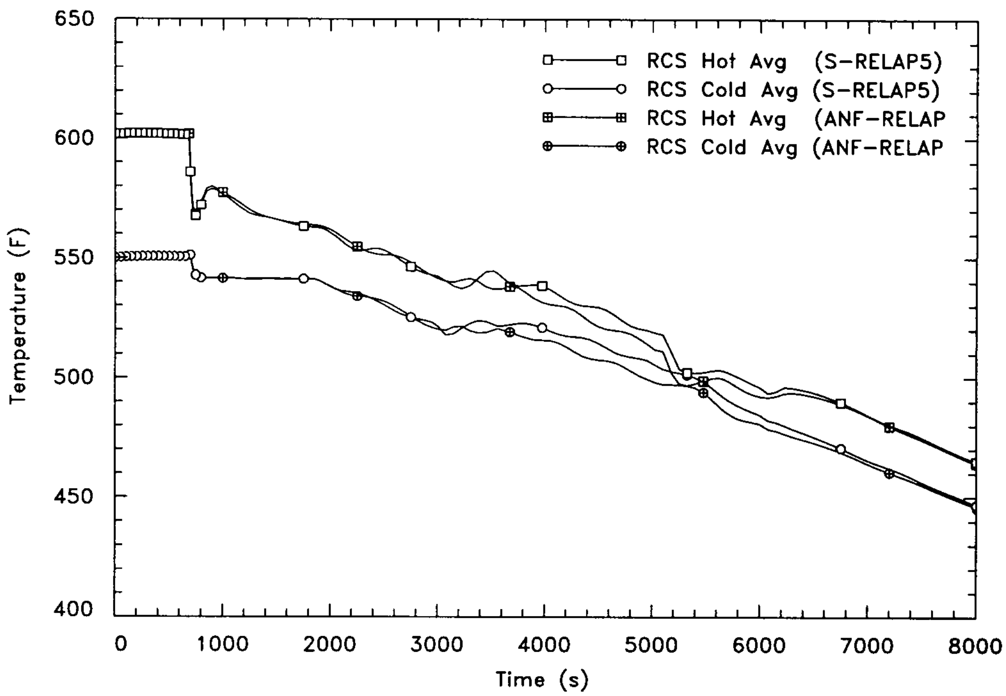

Figure 6.56 SGTR RCS Temperatures

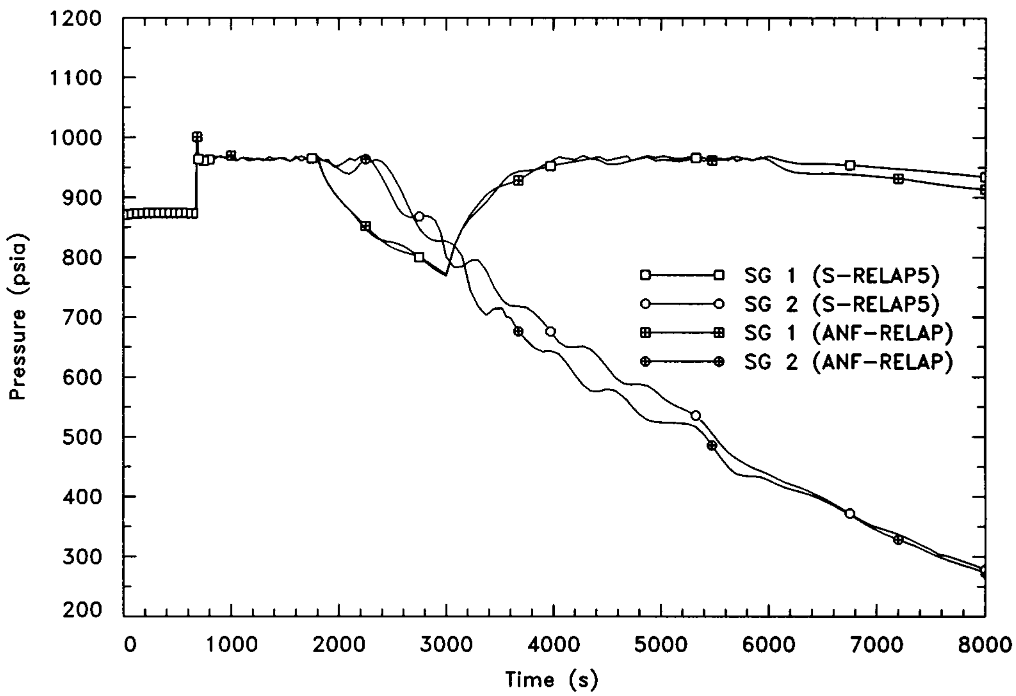

Figure 6.57 SGTR Steam Generator Pressures

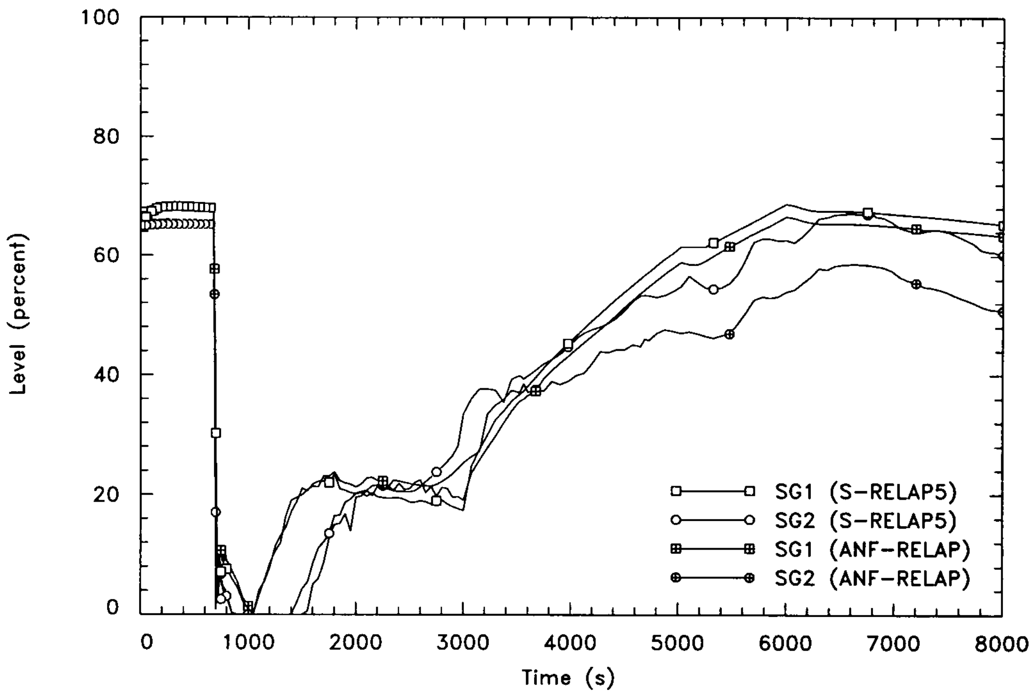

Figure 6.58 SGTR Steam Generator Levels

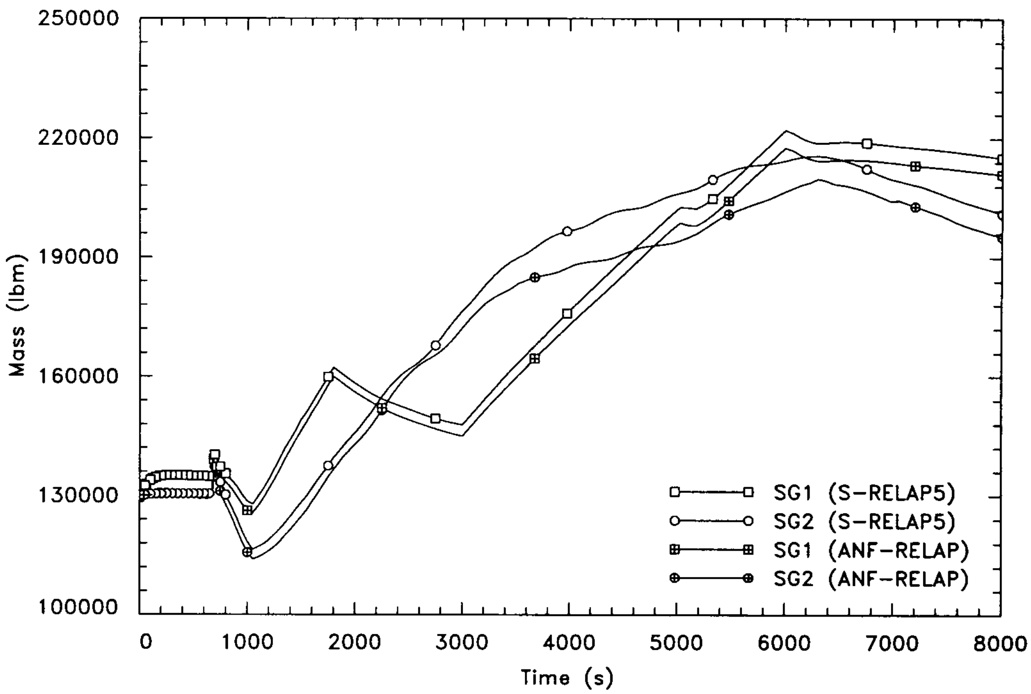

Figure 6.59 SGTR Steam Generator Mass

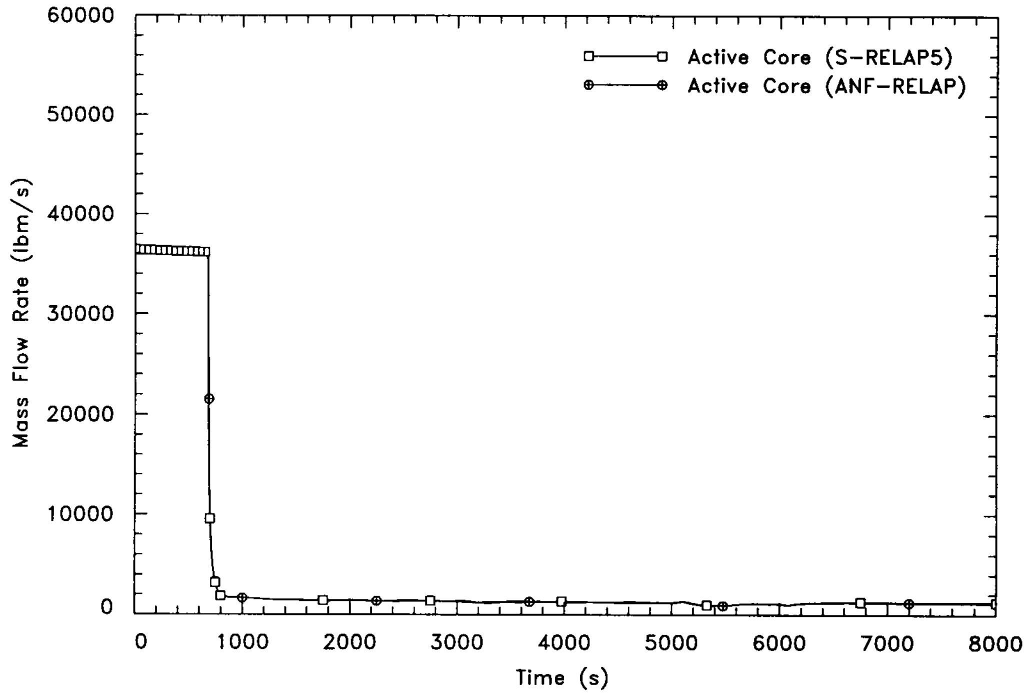

Figure 6.60 SGTR Active Core Mass Flow Rate

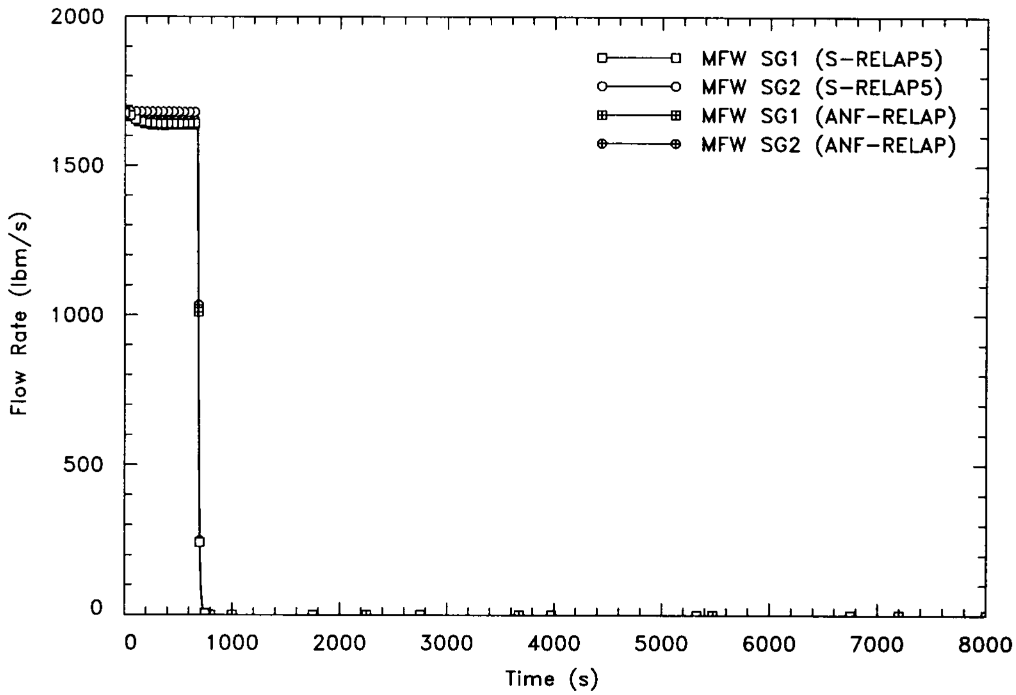

Figure 6.61 SGTR MFW Flow Rates

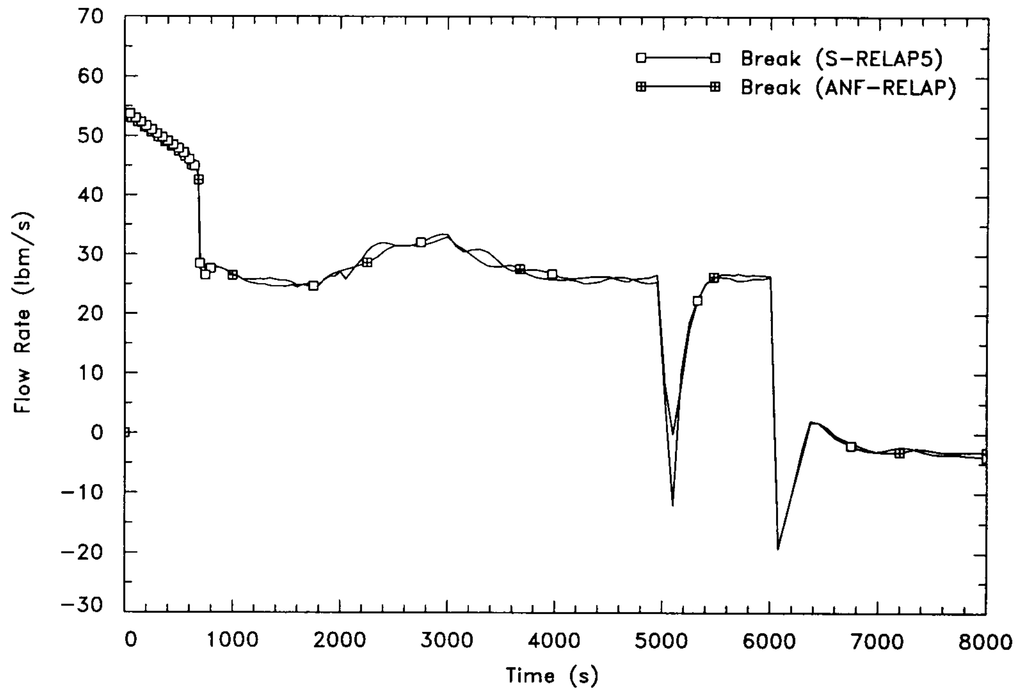

Figure 6.62 SGTR Break Flow Rate

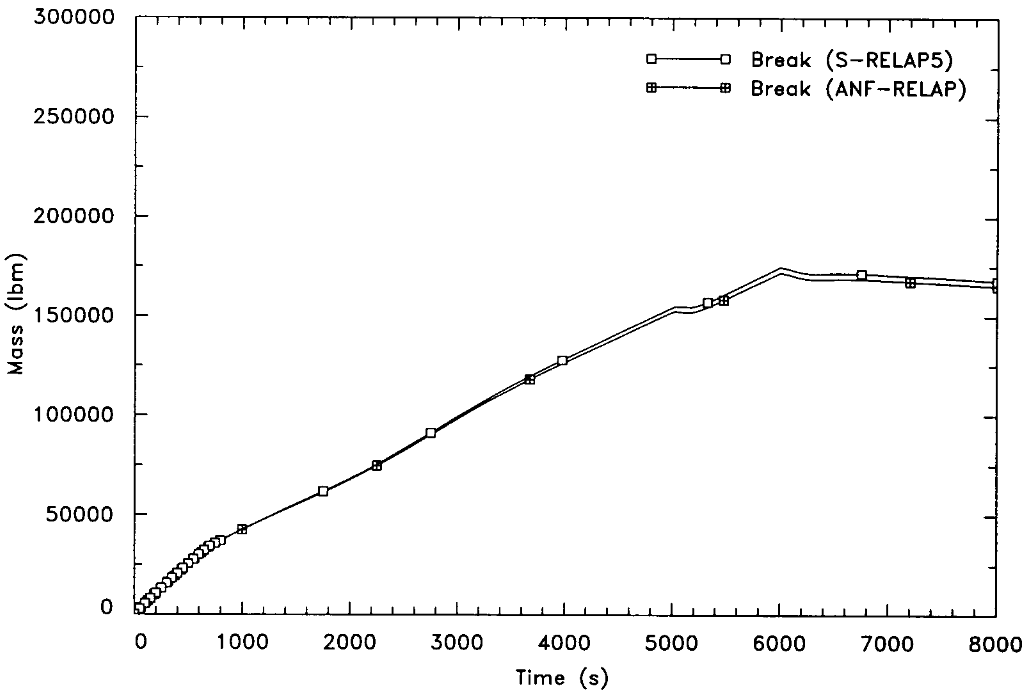

Figure 6.63 SGTR Integrated Break Flow

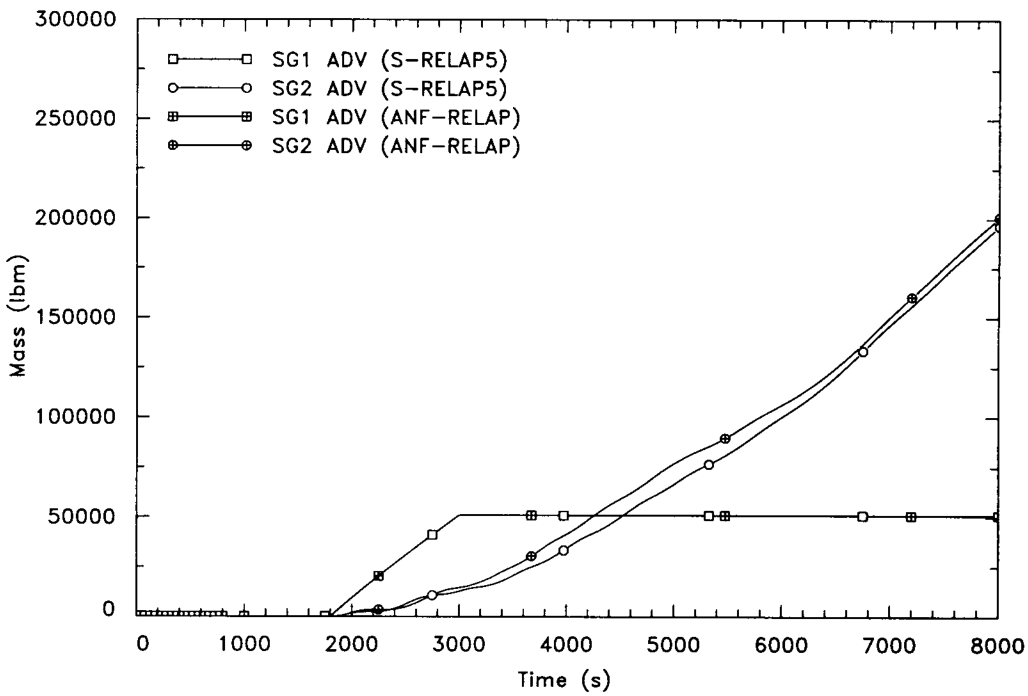

Figure 6.64 SGTR Integrated ADV Flows

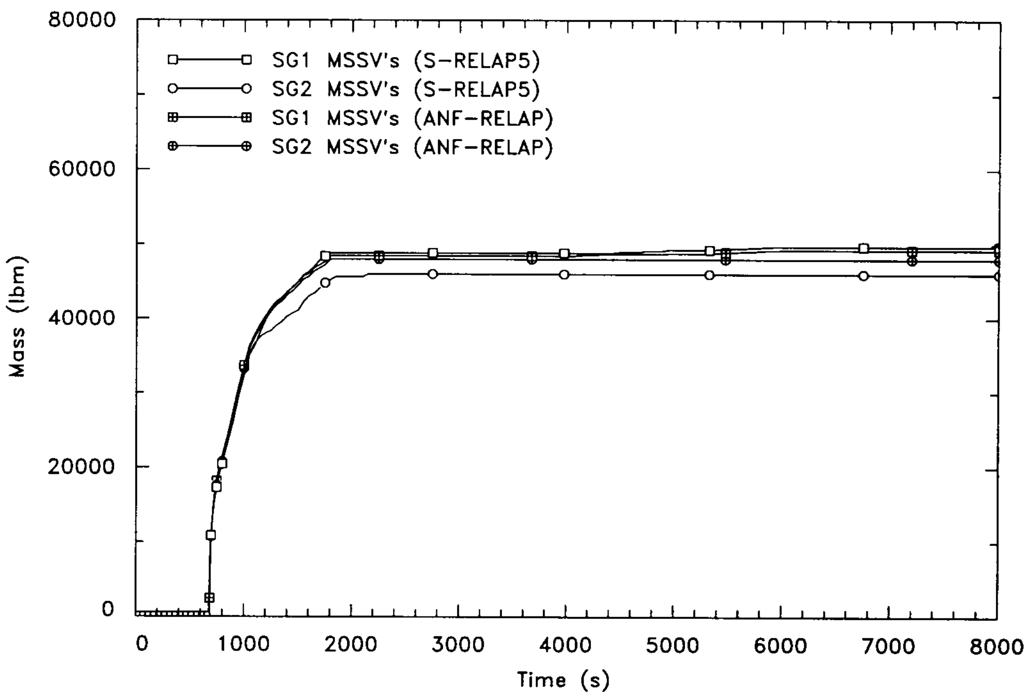

Figure 6.65 SGTR Integrated MSSV Flows

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Revised Pages

The following page was revised in content as a result of NRC review.

]

5.6 *CVCS Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)*

The Boron Dilution event does not require an S-RELAP5 based system analysis. The methodology for performing Boron Dilution analyses is described in this section.

Identification of Causes and Event Description

One means of positive reactivity insertion to the core is the addition of unborated, primary grade coolant from the demineralized and reactor makeup coolant systems. This coolant is introduced to the RCS through the reactor charging/makeup portion of the CVCS.

The most limiting event resulting in an inadvertent boron dilution is typically a malfunction of the CVCS valve which causes pure coolant to be delivered to the RCS by all available charging/makeup pumps. The CVCS and makeup coolant systems are designed to limit, even under various postulated failure modes, the potential rate of dilution to values which will allow sufficient time for automatic or operator response to terminate the dilution. Typically, the sources of dilution may be terminated by closing isolation valves in the CVCS. The lost shutdown margin may be regained by the opening of isolation valves to the RWST, thus allowing the addition of highly borated coolant to the RCS.

The acceptance criteria includes SRP requirements in Section 5.1 for a Condition II event. If operator action is required to terminate the transient, the minimum time intervals to respond are:

- 30 minutes (during refueling); and
- 15 minutes (for all other modes).

These times apply between either (a) the time when an alarm announces an unplanned moderator dilution, or (b) the initiation of the dilution, and the time of loss-of-shutdown margin.

The choice of (a) or (b) is determined from the plant licensing basis.

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