WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
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March 14, 2007
ET 07-0004

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

Reference: 1) Letter ET 06-0032, dated August 2, 2006, from T. J. Garrett, WCNOC, to USNRC

2) NRC letter dated November 6, 2006, from J. L. Funches, USNRC, to T. J. Garrett, WCNOC

Subject: Docket No. 50-482: Revision to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)"

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Operating License NPF-42 for the Wolf Creek Generating Station (WCGS) to incorporate changes to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)." This amendment application proposes to incorporate changes to these specifications based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. Revisions to TS 3.7.3 are made to add the Main Feedwater Regulating Valves (MFRVs) and their associated bypass valves. Additionally, Surveillance Requirement (SR) 3.7.2.1 and SR 3.7.3.1 are revised to relocate the isolation time limits from the SRs to the TS Bases. This change is consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times."

Reference 1 provided notification of WCNOC's plan to replace the MSIVs, MFIVs, and the MSFIS controls during Refueling Outage 16. Reference 1 requested a fee exemption for review fees associated with the specific review of the field programmable gate array (FPGA) technology to be utilized in the replacement MSFIS controls. Reference 2 granted a partial fee exemption.

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Attachments I through V provide the Evaluation, Markup to Technical Specification Pages, Retyped Technical Specification Pages, Proposed TS Bases Changes (for information only), and List of Regulatory Commitments, respectively, in support of this amendment request. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification Bases Control Program," at the time the amendment is implemented.

Enclosure I provides the proprietary CS Innovations LLC Report 9100-00003-P, "Wolf Creek Generating Station Main Steam and Feedwater Isolation System (MSFIS) Controls Summary." As Enclosure I contains information proprietary to CS Innovations LLC, it is supported by an affidavit signed by CS Innovations LLC, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to CS Innovations, be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. This affidavit, along with a CS Innovations LLC authorization letter, 9100-00001, "Application for Withholding Proprietary Information from Public Disclosure," is contained in Enclosure III.

Enclosure II provides non-proprietary CS Innovations LLC Report 9100-00003-NP, "Wolf Creek Generating Station Main Steam and Feedwater Isolation System (MSFIS) Controls Summary." Enclosure IV provides the WCGS System Verification and Validation Plan for the Advanced Logic System MSFIS controls. Enclosure IV defines the procedures and requirements for the comprehensive evaluation of the Advanced Logic System MSFIS controls. Enclosure V provides the Nutherm Qualification Report, WCN-9715R that documents the qualification of the CS Innovations LLC replacement MSFIS controls. The Appendices associated with the Nutherm Qualification Report, which encompasses the test procedures and test data, are not included with this application.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. The amendment application was reviewed by the WCNOC Plant Safety Review Committee. In accordance with 10 CFR 50.91, a copy of this application is being provided to the designated Kansas State official.

WCNOC requests approval of this proposed license amendment by December 31, 2007, to support the preparations for Refueling Outage 16, which is scheduled to start in March 2008. Once approved, the amendment will be implemented prior to startup from Refueling Outage 16.

If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachments:
- I - Evaluation
- II - Markup of Technical Specification Pages
- III - Retyped Technical Specification Pages
- IV - Proposed TS Bases Changes (for information only)
- V - List of Regulatory Commitments

Enclosure:
- I - CS Innovations LLC Report 9100-00003-P, "Wolf Creek Generating Station Main Steam and Feedwater Isolation System (MSFIS) Controls Summary."
- II - CS Innovations LLC Report 9100-00003-NP, "Wolf Creek Generating Station Main Steam and Feedwater Isolation System (MSFIS) Controls Summary."
- III - CS Innovations LLC letter 9100-00001, "Application for Withholding Proprietary Information from Public Disclosure,"
- IV - WCGS System Verification and Validation Plan for the Advanced Logic System MSFIS controls
- V - NUTHERM Qualification Report for CS Innovations Replacement MSFIS System

cc: T. A. Conley (KDHE), w/a, w/e
J. N. Donohew (NRC), w/a, w/e
V. G. Gaddy, NRC, w/a, w/e
B. S. Mallett (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

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

By____________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 14th day of March, 2007.

____
Notary Public

NOTARY PUBLIC STATE OF KANSAS | CINDY NOVINGER | My Appt. Exp. 7/8/10

Expiration Date 7/8/10

1.0 SUMMARY DESCRIPTION

Wolf Creek Nuclear Operating Corporation (WCNOC) requests an amendment to Operating License NPF-42 for the Wolf Creek Generating Station (WCGS) to incorporate changes to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)." This amendment application proposes to incorporate changes to these specifications based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. Revisions to TS 3.7.3 are made to add the Main Feedwater Regulating Valves (MFRVs) and their associated bypass valves.

Additionally, Surveillance Requirement (SR) 3.7.2.1 and SR 3.7.3.1 are revised to relocate the isolation time limits from the SRs to the TS Bases. This change is consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times."

2.0 DETAILED DESCRIPTION

This amendment application proposes to incorporate changes to these specifications based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. The modification is planned for installation in Refueling Outage 16, which is scheduled to start in March 2008. The following changes are being proposed to the WCGS TSs:

- TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," Table 3.3.2-1 is revised by adding the MSFIS automatic actuation logic and actuation relays to Function 4, Steam Line Isolation (new Function 4.c), and Function 5, Turbine Trip and Feedwater Isolation (new Function 5.b).

    - New Function 4.c will require 2 trains of automatic actuation logic and actuation relays (MSFIS) to be OPERABLE in MODE 1, 2, or 3 except when the MSIVs are closed in MODE 2 or 3. Condition G will apply and allows 24 hours to restore one train to OPERABLE status. Surveillance Requirement (SR) 3.3.2.3 is applicable to new Function 4.c. Existing Functions 4.c. and 4.d. are re-lettered based on new Function 4.c.

    - New Function 5.b will require 2 trains of automatic actuation logic and actuation relays (MSFIS) to be OPERABLE in MODE 1, 2, or 3 except when the MFIVs are closed in MODE 2 or 3. New Note (k) is included to address the specific Applicability for new Function 5.b. Condition G will apply and allows 24 hours to restore one train to OPERABLE status. Surveillance Requirement (SR) 3.3.2.3 and SR 3.3.2.6 is applicable to new Function 5.b. Existing Functions 5.b. and 5.c. are re-lettered based on new Function 4.b.

    - Function 5.a is revised to include MODE 3 (including footnote (j)) to address the expanded Applicability in TS 3.7.3. The applicable Condition is revised from Condition H to Condition G and Condition H is deleted. Function 4.b and 5.a are clarified to indicate that the automatic actuation logic and actuation relays are applicable to the Solid State Protection System.

- SR 3.7.2.1 and SR 3.7.3.1 are revised to relocate the isolation times from the SRs to the TS Bases. This change is consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times."

- TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)," is revised to add the main feedwater regulating valves and their associated bypass valves. The following changes are proposed:

  - TS 3.7.3 is retitled to "Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves." TS 3.7.3 LCO, Applicability, ACTIONS, and Surveillance Requirements are revised to incorporate MFRVs and MFRV bypass valves.

  - The Applicability is revised so that WCGS valve specific configurations are addressed.

  - The Completion Time of Condition F is revised to allow 72 hours rather than 4 hours.

  - Current Condition G is re-lettered to Condition J with the addition of new Conditions G, H, and I, to address the addition of the MFRVs and MFRV bypass valves to TS 3.7.3. The default Condition J is revised to include Conditions G, H, or I.

  - SR 3.7.3.1 is revised to include the verification of the isolation time for the MFRVs and MFRV bypass valves. The isolation times for the MFIVs, MFRVs, and MFRV bypass valves are relocated from the TSs to the TS Bases.

  - New SR 3.7.3.3 is added for verifying each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal.

Proposed revisions to the TS Bases are also included in this application. The changes to the affected TS Bases pages will be incorporated in accordance with TS 5.5.14, "Technical Specifications (TS) Bases Control Program."

3.0 TECHNICAL EVALUATION

3.1 <u>System Description</u>

3.1.1 Main Steam Isolation Valves (MSIVs)

One MSIV is installed in each of the four main steam lines outside the containment and downstream of the main steam safety valves. The MSIVs are 28-inch gate valves with hydraulic actuators. The MSIVs prevent uncontrolled blowdown from more than one steam generator (SG) in the event of a postulated steam system piping failure. The valves are bidirectional, double disc, parallel slide gate valves. The valves are designed to close between 1.5 to 5 seconds against the flows associated with line breaks on either side of the valve, assuming the most limiting normal operating conditions prior to occurrence of the break. Each MSIV is equipped with two redundant actuator trains such that either actuator train can independently perform the safety function to fast-close the valve on demand. (Reference 1)

An actuator train consists of a hydraulic accumulator controlled by solenoid valves on the associated MSIV. For each MSIV, one actuator train is associated with separation group 4 (yellow), and one actuator train is associated with separation group 1 (red). The MSIVs are operated by hydraulic actuators. These actuators are controlled by a combination of hydraulic fluid and/or compressed nitrogen gas accumulators, which are controlled by solenoid valves. Each main steam isolation valve has one actuator with two separate nitrogen accumulators. Each accumulator is controlled from a separate Class IE electrical system, and each is capable of closing the valve independently of the other. (Reference 3)

<u>MSIV and Actuator Replacement</u>

The existing hydraulic actuators for the MSIVs have a poor maintenance history. The valve actuators are complex and have numerous O-rings under high pressure. The history of these valves includes leaks that have resulted in loss of generation capacity, and delays in starting up the plant following refueling outages, as well as increased personnel exposure to hazardous materials (use of the hazardous material, Fyrquel, associated with the hydraulic actuators). During Refueling Outage 16, the existing hydraulic actuators will be replaced with system-medium actuators along with the valve bodies.

The new MSIV actuators are shown in Figure 1. They are simple steam pistons, with the piston shaft attached directly to the valve stem. The new MSIV actuators are operated by system-medium (process fluid) to close the valve and utilize instrument air or process fluid to open the valve. Energy for closing an MSIV is provided by the system-medium (steam), which is admitted to the volume above the actuator piston (upper piston chamber) to close the valve. The MSIV actuators utilize six solenoid valves, three solenoids per each train, to perform their safety design functions. Each train is capable of closing the valve independently of each other. Steam will be directed to the actuator upper piston chamber (to close the valve) by two parallel trains consisting of one two-way solenoid valve and one three-way solenoid valve in series. For emergency closure, both upper piston chamber solenoid valves within an actuation train must be de-energized. Once the two upper piston chamber solenoids within an actuation train de-energize, they open to admit steam from the valve bonnet chamber to the actuator upper piston chamber. The actuator lower piston chamber is vented through a two-way solenoid valve and a three-way solenoid valve connected in parallel to the condenser (normal vent path) or a floor drain (backup vent path). The two lower piston chamber solenoid valves are normally de-energized and open to the vent path. After a 30-second time delay, both actuator lower piston chamber solenoid valves will energize, isolating the lower piston chamber. Isolating the lower piston chamber will prevent any leakage of process fluid from either the piston rings of the stem seal from venting through the lower piston chamber to the condenser.

The replacement of the MSIV hydraulic actuators with system-medium actuators will result in a system pressure dependent valve closure time of between 6 seconds to 33 seconds for a steam pressure ranging from 1100 psig to 100 psig. SR 3.7.2.1 currently requires verifying the isolation time of each MSIV is $\leq$ 5 seconds at all system pressures. The isolation time limit is being relocated to the TS Bases consistent with NRC approved TSTF-491 (Reference 4). The replacement of the actuators and the increase in MSIV isolation time has been evaluated for impact on the accident analyses and is further discussed in Section 4.0.

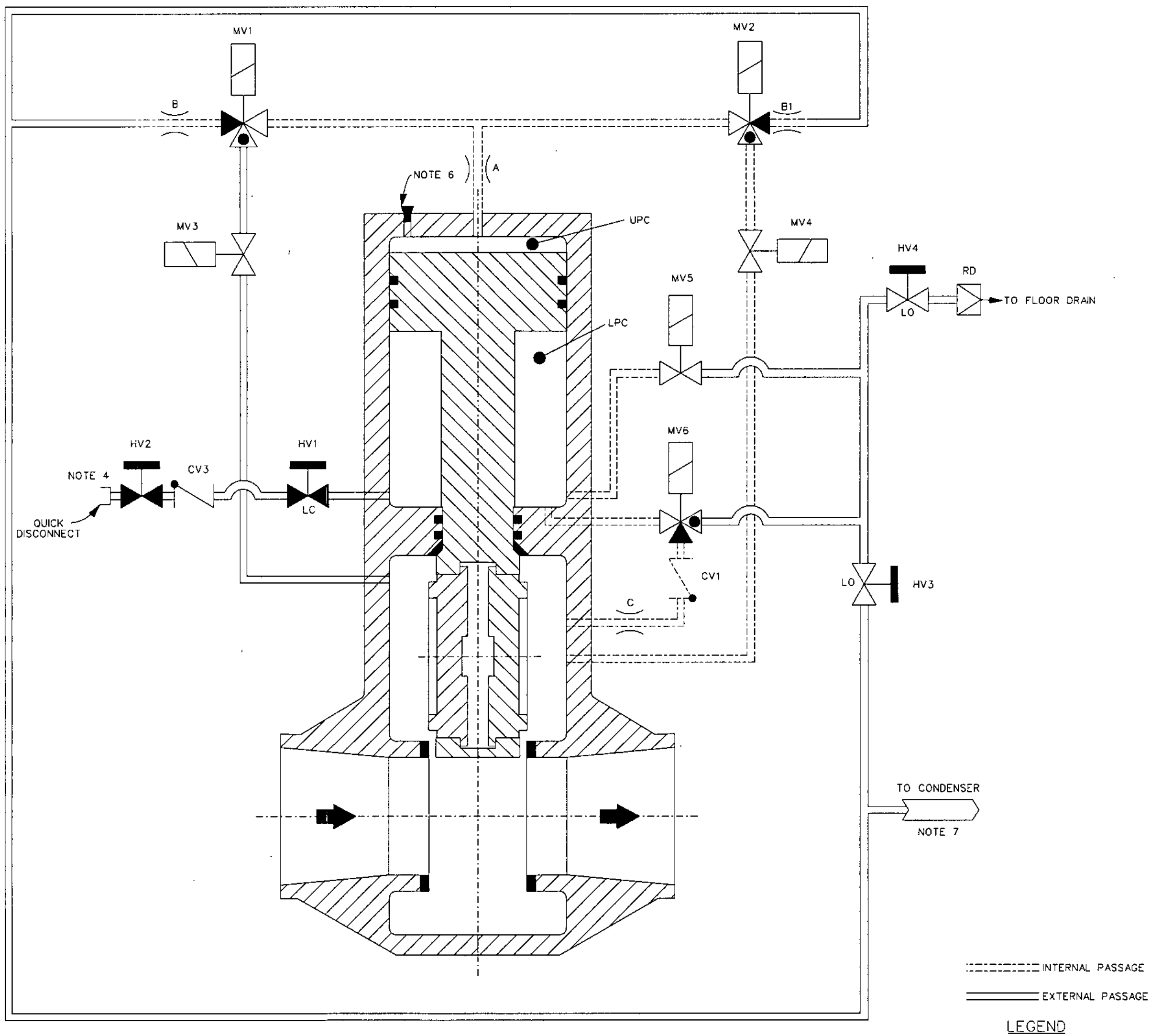


Figure 1
MSIV/MFIV Actuator

3.1.2 Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and MFRV Bypass Valves

One main feedwater isolation valve (MFIV) is installed in each of the four main feedwater lines outside the containment and downstream of the MFRVs. The MFIVs are 14-inch gate valves with hydraulic actuators. The valves are bi-directional, double disc, parallel slide gate valves. The MFIVs are installed to prevent uncontrolled blowdown from more than one SG in the event of a feedwater pipe rupture in the turbine building. The main feedwater check valve provides backup isolation. The MFIVs isolate the non-safety related portions from the safety-related portions of the system. In the event of a secondary cycle pipe rupture inside the containment, the MFIV limits the quantity of high-energy fluid that enters the containment through the broken loop and provides a pressure boundary for the controlled addition of auxiliary feedwater to the three intact loops. (Reference 2)

Each MFIV is equipped with two redundant actuator trains such that either actuator train can independently perform the safety function to fast-close the valve on demand. An actuator train consists of a hydraulic accumulator controlled by solenoid valves on the associated MFIV. For each MFIV, one actuator train is associated with separation group 4 (yellow), and one actuator train is associated with separation group 1 (red). The MFIVs are operated by hydraulic actuators. These actuators are controlled by a combination of hydraulic fluid and/or compressed nitrogen gas accumulators, which are controlled by solenoid valves. Each MFIV has one actuator with two separate nitrogen accumulators. Each accumulator is controlled from a separate Class IE electrical system, and each is capable of closing the valve independently of the other. (Reference 3)

The MFRVs are air-operated angle valves that control feedwater flow to the SGs between approximately 30% and full power. The MFRV bypass valves are air-operated globe valves used to control flow to the SG up to approximately 30% power. The MFRVs and MFRV bypass valves function to control feedwater flow to the SGs. The safety function of the MFRVs and MFRV bypass valves is credited in the accident analyses to provide a backup to the MFIVs for the potential failure of an MFIV to close.

<u>MFIV and Actuator Replacement</u>

The existing hydraulic actuators for the MFIVs have a poor maintenance history. The valve actuators are complex and have numerous O-rings under high pressure. The history of these valves includes leaks that have resulted in loss of generation capacity, and delays in starting up the plant following refueling outages, as well as increased personnel exposure to hazardous materials (use of the hazardous material, Fyrquel, associated with the hydraulic actuators). During Refueling Outage 16, the existing hydraulic actuators will be replaced with system-medium actuators along with the valve bodies.

The new MFIV actuators are shown in Figure 1. The operation of the MFIV actuators is the same as described in Section 3.1.1 with the system-medium being feedwater.

The replacement of the MFIV hydraulic actuators with system-medium actuators will result in a system pressure dependent valve closure time of between 6 seconds to 50 seconds for a system pressure ranging from 1100 psig to 0 psig. SR 3.7.3.1 currently requires verifying the isolation time of each MFIV is $\leq$ 5 seconds at all system pressures. The isolation time limit is being relocated to the TS Bases consistent with NRC approved TSTF-491 (Reference 4). The replacement of the actuators and the increase in MFIV isolation time has been evaluated for impact on the accident analyses and is further discussed in Section 4.0.

3.1.3 Main Steam and Feedwater Isolation System (MSFIS)

The MSFIS consists of two independent actuation trains. Each of the actuation trains monitors system inputs and by means of logic matrices, drive actuation relays that energize or de-energize the solenoids required to the appropriate MSIV or MFIV operation. Each MSIV or MFIV has redundant hydraulic actuator trains, one side active and the other side standby. Both trains operate the same, except they have different control signals. The active sides have Fast Close, Slow Open, Slow Close, 10% Close, Exercise, Engineered Safety Feature Actuation System (ESFAS) and Accumulator Pre-charge for control signals. The standby sides do not have Slow Open and Slow Close sequences. The current logic circuitry is of a discrete solid-state design, except for the electromechanical relays used as the final output devices.

The MSFIS accepts input signals for the MSIVs and MFIVs in the form of contact conditions from the Main Control Board switches (such as 10% close exercise, Slow Open, Slow Close, Accumulator Test, and Manual Fast Close), MSFIS Test Panel rotary switches, the ESFAS output relays, and MSIV or MFIV position limit switches. The input signals are processed through an input buffer for isolation, into a valve control logic card that provides the necessary timing intervals for valve operations, and to an output relay driver to energize or de-energize the actuation relays and solenoids for each valve. The MSFIS Test Panel is used to verify proper operation of the selected valve control logic circuitry and the fast close signal from the input buffer through to the relay output driver. The operation of an eight position switch for each valve provides the testing functions.

MSFIS processes the safety inputs from ESFAS and Manual Fast Close to produce the desired safety signals. However, failures have occurred on the printed circuit boards that produce false actuation signals that close either an MSIV or MFIV. Closure of either valve at power has caused plant trips and reduced plant availability.

<u>Description of MSFIS Modification</u>

The MSFIS was identified as a multiple single point failure system that could cause plant trips. In August 2003 a plant trip occurred due to a failed circuit card in the system and circuit failures have caused valves to stroke partially closed. The existing MSFIS contains obsolete components and the current logic will not operate the replacement valve actuators. The MSFIS controls are being replaced since the existing MSFIS controls are not compatible with the replacement MSIV and MFIV system-medium actuators.

As a result, WCNOC developed Specification J-105A for the MSFIS replacement. WCNOC is replacing the existing MSFIS with a control system based on the Advance Logic System (ALS) technology. The ALS technology incorporates distributed controls such that no single failure will result in an untimely closure of a MSIV or MFIV. The distributed control is achieved by having multiple autonomous boards in the system, each controlling a part of the system. The intelligence on each board is implemented using a standard logic based architecture implemented in field programmable gate arrays to ensure a deterministic and reliable behavior.

The MSFIS consists of two independent actuation trains that monitor system inputs and by means of advanced logic matrices, drive actuation relays that energize or deenergize the solenoids required for the appropriate MSIV or MFIV operation. The modified MSFIS performs the same as the current design except that the system is comprised of advanced logic technology. The Solid State Protection System and Reactor Protection System inputs the ESFAS signals, the Main Control Board handswitches input to the MSFIS cabinets, and the same actuation output relays are utilized in the new design.

Enclosure I provides the CS Innovations LLC document, "Wolf Creek Generating Station Main Steam and Feedwater Isolation System (MSFIS) Controls Summary." This document provides the system level design associated with the replacement Advanced Logic System MSFIS controls. Enclosure IV provides the WCNOC System Verification and Validation Plan. This document defines the procedures and requirements for a comprehensive evaluation that will assure that the Advanced Logic System MSFIS controls meet the requirements for a safety related Class 1E system. Enclosure V provides the Nutherm International report, "Nutherm Qualification Report for CS Innovations Replacement MSFIS System." This report documents the results of the qualification program and established that the MSFIS controls will provide the required safety function in the specified mild environment, seismic and electromagnetic/radio frequency conditions. The Appendices associated with the Nutherm Qualification Report, which encompasses the test procedures and test data, are not included with this application.

3.2 Revisions to the Technical Specifications

3.2.1 TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation"

TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," Condition H is deleted and indicated as "Not Used." Condition G is applicable to Function 5.a, Automatic Actuation Logic and Actuation Relays – Turbine Trip and Feedwater Isolation, with the addition of MODE 3 to the Applicable MODES or Other Specified Conditions.

TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," Table 3.3.2-1 is revised by adding the MSFIS Automatic Actuation Logic and Actuation Relays to Function 4, Steam Line Isolation (new Function 4.c), and Function 5, Turbine Trip and Feedwater Isolation (new Function 5.b).

New Function 4.c will require 2 trains of automatic actuation logic and actuation relays (MSFIS) to be OPERABLE in MODE 1, 2, or 3 except when the MSIVs are closed in MODE 2 or 3. Condition G will apply and allows 24 hours to restore one train to OPERABLE status. Surveillance Requirement (SR) 3.3.2.3 and SR 3.3.2.6 are applicable to new Function 4.c.

Function 5.a is revised to include MODE 3 (including footnote (j)) and the applicable Condition is revised to Condition G to address the expanded Applicability in TS 3.7.3. Function 4.b and 5.a are clarified to indicate that the automatic actuation logic and actuation relays are applicable to the Solid State Protection System.

New Function 5.b will require 2 trains of automatic actuation logic and actuation relays (MSFIS) to be OPERABLE in MODE 1, 2, or 3 except when the MFIVs are closed in MODE 2 or 3. New Note (k) is included to address the specific Applicability for new Function 5.b. Condition G will apply and allows 24 hours to restore one train to OPERABLE status. Surveillance Requirement (SR) 3.3.2.3 and SR 3.3.2.6 are applicable to new Function 5.b.

Justification of 24 hour Completion Time for new Function 4.c and Function 5.b

The existing MSFIS logic is considered part of the Solid State Protection System in TSs. Currently, if one train of MSFIS is inoperable, then one train of Solid State Protection System is inoperable. The addition of MSFIS automatic actuation logic and actuation relays to TS Table 3.3.2-1 allows one train of MSFIS to be declared inoperable or placed in test without declaring the corresponding train of Solid State Protection System inoperable.

As noted above, Condition G will apply to new Functions 4.c and 5.b with a Completion Time of 24 hours. In March 1991, Amendment No. 43 approved a 6 hour Allowed Outage Time or Completion for the Automatic Actuation Logic and Actuation Relays (Steam Line Isolation Function and Turbine Trip and Feedwater Isolation Function) based on the qualitative analysis in WCAP-10271, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System." In January 2005, Amendment No. 156 approved a 24 hour Completion Time for Condition G which applied to Function 4.b, Steam Line Isolation – Automatic Actuation Logic and Actuation Relays, based on the analysis in WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times." WCNOC considered the MSFIS cabinets, in addition to the Solid State Protection System to be included in the Automatic Actuation Logic and Actuation Relays Function. Subsequent to the issuance of Amendment No. 156, WCNOC determined that further evaluation was necessary to justify a 24 hour Completion Time for the MSFIS actuation logic and actuation relays.

In order to maximize the applicability of the generic WCAP-14333 evaluation to most if not all of the Westinghouse plants, relatively conservative assumptions were made regarding duration of testing, maintenance frequency and duration, and impact of component failure.

Assumptions relative to the WCAP-14333 evaluation include an actuation signal maintenance interval of once every 18 months, with a maintenance activity outage time of 24 + 6 hours (Section 5.2 and Table 5.1). In addition, the WCAP-14333 evaluation assumes a test interval of once every two months, with a test duration of four hours (36 hours in 18 months). In other words, WCAP-14333 demonstrates an acceptable increase in plant risk based on the impact of change in annual Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) risk given a mean actuation signal test and maintenance unavailability of 60 hours in an 18 month period.

As long as it can be shown that the combined mean actuation signal unavailability for the Solid State Protection System and MSFIS remains bounded by the WCAP-14333 assumptions; the acceptable increase in annual risk determined in WCAP-14333 will also be bounding. As long as the increase in annual risk as determined in WCAP-14333 is bounded, application of the 24 hour Completion Time for both the Solid State Protection System and MSFIS is acceptable. The increase in annual risk, on which the NRC approval of the WCAP-14333 changes was based, remains valid.

A review of WCGS Maintenance Rule unavailability data (2001 through 2004) was performed for both the Solid State Protection System and MSFIS portions of the TS 3.3.2 Function 4.b and 5.a signals. This review determined a mean annual unavailability for these functional signals of approximately 2.9 hours per year, with a maximum for any given year of 5.6 hours. This would equate to a mean unavailability of approximately 4.3 hours in an 18 month period. The TS 3.3.2 Function 4.b and 5.a signals are also impacted when performing STS IC-211A/B. The WCGS Maintenance Rule unavailability data indicates an additional unavailability of approximately 9 hours per year for performance of this STS. For an 18 month period, the total unavailability for these functional signals would be approximately 22 hours {[5.6 hrs (max per year) + 9 hours] * 18/12}. For the most part, the unavailability time for TS 3.3.2 Function 4.b and 5.a for this time period is due to testing. This historical unavailability was for a period during which the Completion Time for the TS 3.3.2 Function 4.b and 5.a signals was 6 hours.

A total of 25 Westinghouse units (16 plants) provided input to a survey performed for WCAP-14333 indicating the expected increase in unavailability time associated with Completion Time extension. Nearly all units indicated that they expected no increase in unavailability time due to Completion Time extension. Three units indicated an estimated unavailability increase of 25%, with two units indicating an estimated unavailability increase of 50%, due to Completion Time extension.

If the maximum estimated unavailability increase of 50% is applied, the unavailability for an 18 month period would be approximately 33 hours. If it were conservatively assumed that the Completion Time increase would result in a doubling of the overall test and maintenance unavailability, the unavailability for an 18 month period would be approximately 44 hours.

An assumed increase in mean unavailability time due to Completion Time extension of either 50%, or an increase by a factor of two, is considered bounding. In either case, the mean unavailability still remains well below the mean test and maintenance unavailability of 60 hours in an 18 month period assumed in the WCAP-14333 evaluation.

Utilization of an assumption of mean unavailability of 24 hours in an 18 month period in the WCAP-14333 evaluation results in a CDF and LERF risk increase within the acceptance criteria of Regulatory Guide 1.174 and Regulatory Guide 1.177. Even if bounding assumptions of the degree of combined Solid State Protection System and MSFIS unavailability for the TS 3.3.2 Function 4.b and 5.a signals are applied, the mean unavailability resulting from an increase in Completion Time to 24 hours remains well below that assumed in the WCAP-14333 evaluation.

Based on the additional evaluation performed by WCNOC, it is acceptable, from a plant risk increase perspective, to apply the 24 hour Completion Time justified in WCAP-14333 to both the Solid State Protection System and MSFIS portions of the TS 3.3.2 Function 4.b and 5.a signals. As such, the application of a 24 hour Completion Time for new Function 4.c and Function 5.b is appropriate.

Surveillance Requirements for new Function 4.c and Function 5.b

Surveillance Requirement (SR) 3.3.2.3 and SR 3.3.2.6 are applicable to new Function 4.c and Function 5.b. SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST. The MSFIS reliability analysis report "System Reliability Analysis for Advanced Logic System," calculated a Mean Time Between Failure for a single separation group as 28,881 hours, which is equivalent to 3.28 years. However, WCNOC will maintain a surveillance frequency for the MSFIS actuation logic of 31 days on a STAGGERED TEST BASIS (using the MSFIS automatic tester), which is consistent with the frequency of the ACTUATION LOGIC TEST for the BOP ESFAS. WCNOC will continue to perform a SLAVE RELAY TEST (SR 3.3.2.6) on the associated MSFIS slave relays.

3.2.2 TS 3.7.2, "Main Steam Isolation Valves (MSIVs)"

SR 3.7.2.1 is revised to relocate the isolation times from the SRs to the TS Bases. This change is consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times." The availability of this TS improvement was announced in the Federal Register on December 29, 2006 (71 FR 250, page 78472) as part of the consolidated line item improvement process. TSTF-491, Revision 2, proposed to relocate the required isolation times for the MSIVs to a licensee controlled document this is referenced in the TS Bases. WCNOC is relocating the isolation times to the TS Bases in lieu of a different licensee controlled document. Changes to the TS Bases are subject to the 10 CFR 50.59 process. TS 5.5.14, "Technical Specification (TS) Bases Control Program," provides adequate assurance that prior NRC review and approval will be requested for changes to the TS Bases requirements in accordance with 10 CFR 50.59. Furthermore, the MSIVs are subject to the periodic testing and acceptance criteria in accordance with the Inservice Testing Program. Compliance with the Inservice Testing Program is required by 10 CFR 50.55a and TS 5.5.8. As such, the TS provides multiple requirements to assure the MSIVs are maintained OPERABLE.

The impact of an MSIV closure time as a function of SG pressure on the accident analyses was evaluated and is discussed in Section 4.0. The evaluation concluded that a variable MSIV closure time is acceptable with respect to the accident analyses. A curve of the MSIV closure time limit as a function of SG pressure is incorporated into the TS 3.7.2 Bases. The single value closure time limit is being replaced with a closure time test acceptance criteria more appropriate to the MSIV system-medium actuator design.

3.2.3 TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)"

- TS 3.7.3 is retitled to "Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves." TS 3.7.3 LCO, Applicability, ACTIONS, and Surveillance Requirements are revised to incorporate MFRVs and MFRV bypass valves.

- The Applicability is revised so that WCGS valve specific configurations are addressed.

- The Completion Time of Condition F is revised to allow 72 hours rather than 4 hours.

- Current Condition G is re-lettered to Condition J with the addition of new Conditions G, H, and I, to address the addition of the MFRVs and MFRV bypass valves to TS 3.7.3. The default Condition J is revised to include Conditions G, H, or I.

- SR 3.7.3.1 is revised to include the verification of the isolation time for the MFRVs and MFRV bypass valves. The isolation times for the MFIVs, MFRVs, and MFRV bypass valves are relocated from the TSs to the TS Bases.

- New SR 3.7.3.3 is added for verifying each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal.

<u>Addition of MFRVs and MFRV Bypass Valves to TS 3.7.3</u>

The MFIVs isolate main feedwater flow to the secondary side of the SGs following a high energy line break (HELB). The MFRVs and MFRV bypass valves function to control feedwater flow to the SGs. The safety function of the MFRVs and MFRV bypass valves is to provide backup isolation of the main feedwater flow to the secondary side of the SGs following an HELB. Because an earthquake is not assumed to occur coincident with a spontaneous break of safety related secondary piping, loss of the non-safety grade MRFVs and MFRV bypass valves is not assumed. If the single active failure postulated for a secondary pipe break is the failure of a safety grade MFIV to close, then credit is taken for closing or isolating the non-safety grade MFRVs or MFRV bypass valves. The MFRVs and MFRV bypass valves are highly reliable backups to the MFIVs.

This is supported by NUREG-0138, "Staff Discussion on Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR to NRC Staff." It states the following:

> Consistent with the lesser safety importance of the secondary system boundary, staff does not require that an earthquake be assumed to occur coincident[ly] with a postulated spontaneous break of the steamline piping; i.e., loss of equipment not designed to withstand a SSE (Safe Shutdown Earthquake) is not assumed coincident with an assumed spontaneous steamline break accident.
>
> Continued reliability of these components over the life of the plant is assured by frequency (generally weekly) [of] in-service tests. Thus, the staff believes that it is acceptable to rely on these non-safety grade components in the steam and feedwater systems because their design and performance are compatible with the accident conditions for which they are called upon to function. It is the staff position that utilization of these components as a backup to a single failure in safety grade components adequately protects the health and safety of the public.

Closure of the MFIVs or the MFRVs and MFRV bypass valves terminates flow to the SGs, terminating the event for feedwater line breaks occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the mean steam lines or in the main feedwater lines downstream of the MFIVs are mitigated by their closure. Closure of the MFIVs or MFRVs and MFRV bypass valves, effectively terminates the addition of feedwater to an affected SG, limiting the mass and energy release from steam line breaks.

The MFIVs and the main feedwater check valves isolate the non-safety related portions from the safety related portions of the system. In the event of a feedwater pipe rupture in the non-safety related portion of the system, the check valves close to terminate loss of fluid from the secondary side. In the event of a secondary side pipe rupture inside containment, the MFIVs limit the quantity of high energy fluid that enters containment through the break. The main feedwater check valves provide a pressure boundary for the controlled addition of auxiliary feedwater to the intact loops.

The MFIVs or the MFRVs and the MFRV bypass valves close on receipt of any safety injection (SI) signal, a Tavg-Low coincident with reactor trip (P-4), a Low-Low SG Water Level signal, or SG Water Level-High High signal. The MFIVs may also be actuated manually. Credit is taken in the accident analyses for the MFIVs to close on demand. However, the MFRVs and MFRV bypass valves are provided as a highly reliable backup in the unlikely event a mechanical failure prevented the primary isolation valves from fully closing. Therefore, the MFRVs and MFRV bypass valves are fully capable of mitigating the design basis event. Section 4.0 provides further discussion of the accident analyses.

The proposed LCO requires that four MFIVs and their associated actuator trains, four MFRVs, and four MFRV bypass valves be OPERABLE. The MFIVs and MFRVs and MFRV bypass valves are considered OPERABLE when isolation times are within limits when given an isolation actuation signal and they are capable of closing on an isolation actuation signal. The availability of the MFRVs and MFRV bypass valves to perform the backup isolation function is assured by the new requirements contained in the proposed TS change. Because the TS requirements provide assurance that MFRVs and MFRV bypass valves can perform the required isolation function, a 72 hour Completion Time for one or more MFIVs inoperable is warranted.

The Completion Times (72 hours for one or more MFIVs, one or more MFRVs, or one or more MFRV bypass valves inoperable and 8 hours for two valves in the same flow path inoperable) are reasonable, based on operating experience and the low probability of an event occurring during this time period that would require isolation of the main feedwater flow paths. The extension of the Completion Time for inoperable MFIVs could prevent an unnecessary plant shutdown transient or prevent a feedwater transient due to a less than adequate time allowed for a repair.

Relocation of Isolation Times from SR 3.7.3.1

SR 3.7.3.1 is revised to relocate the isolation times from the SRs to the TS Bases. This change is consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times." The availability of this TS improvement was announced in the Federal Register on December 29, 2006 (71 FR 250, page 78472) as part of the consolidated line item improvement process. TSTF-491, Revision 2, proposed to relocate the required isolation times for the MSIVs to a licensee controlled document that is referenced in the TS Bases. WCNOC is relocating the isolation times to the TS Bases in lieu of a different licensee controlled document. Changes to the TS Bases are subject to the 10 CFR 50.59 process. TS 5.5.14, "Technical Specification (TS) Bases Control Program," provides adequate assurance that prior NRC review and approval will be requested for changes to the TS Bases requirements in accordance with 10 CFR 50.59. Furthermore, the MFIVs are subject to the periodic testing and acceptance criteria in accordance with the Inservice Testing Program. Compliance with the Inservice Testing Program is required by 10 CFR 50.55a and TS 5.5.8. As such, the TS provides multiple requirements to assure the MFIVs are maintained OPERABLE.

The impact of an MFIV closure time as a function of SG pressure on the accident analyses was evaluated and is discussed in Section 4.0. The evaluation concluded that a variable MFIV closure time is acceptable with respect to the accident analyses. A curve of the MFIV closure time limit as a function of SG pressure is incorporated into the TS 3.7.3 Bases. The single value isolation time limit is being replaced with an isolation time test acceptance criteria more appropriate to the MFIV system-medium actuator design.

4.0 Evaluation of Accident Analyses

The replacement of the MSIVs and MFIVs and associated actuators will result in an increase in the valve closure time. Depending on the system pressure, the closure time for the MFIV varies from approximately 6 seconds to approximately 50 seconds for a system pressure ranging from 1100 psig to 0 psig. For the MSIV, the closure time varies from approximately 6 seconds to approximately 33 seconds for a steam pressure ranging from 1100 psig to 100 psig. Consequently, the accident analyses were evaluated or reanalyzed to account for the increase in the valve closure time. For analysis purposes, a conservative yet bounding closure time of 15 seconds has been assumed in the evaluation or analyses discussed in Section 4.0. This closure time was determined based on the steam generator pressures expected to remain higher than 400 psig for all affected design basis accidents when the valves are approaching the closed position.

The safety implications associated with the replacement of the MSIVs and MFIVs and associated actuators have been evaluated, based on reanalysis or engineering evaluation for the accident scenarios in USAR Chapter 3, Appendix 3B, Chapter 6, and Chapter 15.

A review of the current licensing basis accident analyses presented in USAR Chapter 15 reveals that the MSIV and MFIV isolation time acceptance criteria are neither specifically nor implicitly modeled or credited in the following Chapter 15 design basis accidents. As such, the replacement of the MSIVs and MFIVs and associated actuators has no impact on the following USAR accident analyses conclusions:

- Feedwater system malfunctions that result in a decrease in feedwater temperature (USAR Section 15.1.1)
- Excessive increase in secondary steam flow (USAR Section 15.1.3)
- Loss of external electrical load (USAR Section 15.2.2)
- Turbine trip (USAR Section 15.2.3)
- Loss of condenser vacuum and other events resulting in turbine trip (USAR Section 15.2.5)
- Partial loss of forced reactor coolant flow (USAR Section 15.3.1)
- Complete loss of forced reactor coolant flow (USAR Section 15.3.2)
- Reactor coolant pump shaft seizure (locked rotor) (USAR Section 15.3.3)
- Reactor coolant pump shaft break (USAR Section 15.3.4)
- Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (USAR Section 15.4.1)
- Uncontrolled rod cluster control assembly bank withdrawal at power (USAR Section 15.4.2)
- Rod cluster control assembly misoperation (USAR Section 15.4.3)
- Startup of an inactive reactor coolant pump at an incorrect temperature (USAR Section 15.4.4)
- Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant (USAR Section 15.4.6)
- Inadvertent loading and operation of a fuel assembly in an improper position (USAR Section 15.4.7)
- Spectrum of rod cluster control assembly ejection accidents (USAR Section 15.4.8)
- Inadvertent operation of the emergency core cooling system during power operation (USAR Section 15.5.1)
- Chemical and volume control system malfunction that increases reactor coolant inventory (USAR Section 15.5.2)
- Inadvertent opening of a pressurizer safety or relief valve (USAR Section 15.6.1)

- Break in instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment (USAR Section 15.6.2)

- Radioactive release from a subsystem or component (USAR Section 15.7)

The following design basis accidents do not model the MSIVs, but model MFIVs to ensure that the applicable acceptance criteria are satisfied. Impact of the proposed MFIV replacement was reviewed and qualitatively evaluated in subsequent sections.

- Feedwater system malfunctions that result in an increase in feedwater flow (USAR Section 15.1.2)

- Loss of non-emergency AC power to the station auxiliaries (USAR Section 15.2.6)

- Loss of normal feedwater flow (USAR Section 15.2.7)

The following design basis accidents model the MSIVs and/or MFIVs. Consequences due to the valve and actuator replacement were obtained by reanalysis of the accidents with the results discussed below.

- Steam system piping failure (USAR Section 15.1.5)

- Feedwater system pipe break (USAR Section 15.2.8)

- Steam Generator Tube Rupture (USAR Section 15.6.3)

- Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (USAR Section 15.6.5)

In addition to the Chapter 15 design basis accidents, the USAR Section 6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary Pipe Rupture Inside Containment, USAR Chapter 3, Appendix 3B, and Main Steam Line Break (MSLB) Mass and Energy Release Analysis Outside Containment, were reanalyzed for the replacement of the valves and actuators. The results of the reanalyses are discussed below.

4.1 Feedwater system malfunctions that result in an increase in feedwater flow (USAR Section 15.1.2)

This event is analyzed at hot full power and hot zero power conditions for automatic and manual rod control feedwater malfunction cases. Feedwater isolation is assumed to occur from a SG High-High Water Level signal. To assess the effect of an increase in the MFIV closure time from the current licensing basis value of 5.0 seconds to 15 seconds, due to the installation of new MFIVs and actuators, both the automatic and manual rod control feedwater malfunction cases were evaluated with an increased MFIV closure time of 15 seconds.

In the hot full power condition, results from the automatic and manual rod control cases show very little variation on the core conditions, including core inlet enthalpy and core exit pressure, at the time of peak core average heat flux. The differences in the core conditions are small and would result in a small difference in the calculated minimum Departure from Nucleate Boiling Ratio (DNBR). The minimum DNBR, with greater than a 6.9% margin to the DNBR limit documented in the current licensing basis accident analysis would have negligible impact, due to the reanalysis modeling with an increased MFIV closure time of 15 seconds.

Similarly, for the hot zero power condition, results from the automatic and manual rod control cases show very little core condition variation, including core inlet enthalpy and core exit pressure, at the time of peak core average heat flux. The differences in the core conditions are small and would result in a small difference in the calculated minimum DNBR. Therefore, the minimum DNBR with greater than 10% margin to the DNBR limit documented in the current licensing basis accident analysis, would be negligibly impacted, due to the reanalysis modeling with an increased MFIV closure time of 15 seconds.

As such, it is concluded that the acceptance criteria for the hot full power and hot zero power cases will be met and the USAR accident analysis conclusions remain valid.

4.2 Loss of non-emergency AC power to the station auxiliaries/Loss of normal feedwater flow (USAR Section 15.2.6 and 15.2.7)

In the Loss of Non-emergency AC Power to the Station Auxiliaries (LOAC) and Loss of Normal Feedwater Flow (LONF) events, the loss of main feedwater results in a reactor trip on a Low-Low SG Water Level trip signal. This signal initiates the sequence of events that provide auxiliary feedwater (AFW) flow to the SGs. The LOAC/LONF transients analyzed for the USAR are classified as Condition II events, "Incidents of Moderate Frequency," as defined by the American Nuclear Society's "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973.

In accordance with USAR Section 10.4.7.2, the feedwater check valves are downstream of the AFW injection point. Thus, before AFW can be delivered to the SG, the MFIVs must be fully closed to enable the AFW to fill the piping volume (purge the relatively hot feedwater) from the MFIVs to the SG. Although main feedwater isolation is not explicitly modeled as such in the LOAC/LONF analyses, the assumed AFW purge volume implies that main feedwater isolation occurs. Thus, based on the current analysis of record for these events, main feedwater isolation must occur prior to crediting AFW initiation. In the LOAC/LONF analyses, the total delay to the start of AFW from the Low-Low SG Water Level signal is 394 seconds, which includes a 2 second delay from the Low-Low SG Water Level signal to reactor trip (rod motion) and an additional delay of 392 seconds after reactor trip. In USAR Sections 15.2.6.2 and 15.2.7.2, the 392 second delay accounts for the 60 second delay for diesel generator and AFW pump start and allows for the filling of the associated feedwater piping. The MFIVs are designed such that they will close using the highest system pressure from either side of the valve. It is assumed that during LOAC/LONF events the highest system pressure will not fall below 515 psia. Thus, the maximum MFIV closure time to be evaluated is bounded by 15 seconds.

The 392 second delay described above sufficiently accommodates the increased MFIV closure time of 15 seconds. Thus, there is no effect on the results of the LOAC/LONF analyses and the acceptance criteria for the LOAC/LONF events continue to be met. As such, the conclusions for the LOAC/LONF events presented in the USAR remain valid. The radiological consequences of the LOAC/LONF events are not of concern, as the DNB and overpressurization criteria are satisfied.

4.3 Steam system piping failure (USAR Section 15.1.5)

Introduction

The rupture of the main steam line accident has been reanalyzed to support the MSIV and MFIV and associated actuator replacement, in which the closure characteristics of the valves are different compared to the current licensing basis analysis of record. Currently, it is assumed that these valves begin closing following a 2 second time delay, and ramp closed linearly in 5 seconds. For reanalysis purposes, a 15 second isolation time for valve closure, consistent with other related analyses, has been assumed.

Transient Description

The rupture of a main steam line results in increased steam flow that subsequently enhances primary-to-secondary heat transfer and reduces the Reactor Coolant System (RCS) coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power, a potential problem due to the high peaking factors which exist.

The following functions provide the necessary protection against a steam pipe rupture:

1. SI System actuation.

2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SI signal.

3. Redundant isolation of the main feedwater lines to prevent sustained high feedwater flow, which would cause additional cooldown. Therefore, an SI signal will rapidly close all MFIVs, trip the main feedwater pumps, close the MFRVs and the MFRV bypass valves that provide a diverse backup to the MFIVs. In addition, trip of the main feedwater pumps results in automatic closure of the respective pump discharge isolation valve.

4. Trip of the fast-acting MSIVs (assumed to close with a 15 second closure time) after receipt of a steam line isolation signal.

Following a steam line break, only one SG can blow down completely. Each main steam line is provided with an isolation valve (MSIV) located outside of the containment immediately downstream of the steam line safety valves. The MSIVs are signal-actuated valves that close to prevent flow in the normal (forward) flow direction. The MSIVs on all four steam lines will be driven closed to isolate the respective SGs. Thus, only one SG can blow down, minimizing the potential steam release and resultant RCS cooldown. Redundant isolation of the main feedwater lines is provided by: (1) control actions that close the MFIVs following reactor trip and (2) trip of the main feedwater pumps and closure of pump discharge and MFRVs following receipt of an SI signal. The remaining three SGs will still be available for dissipation of any decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the SG atmospheric relief or safety valves.

Assumptions/Methods

For the most part, the assumptions and methodology used in this reanalysis are consistent with the current licensing basis analysis of record. The transient analysis was performed using the RETRAN-3D computer code operating in the RETRAN-02 mode. A detailed core analysis was then performed using the ANC code to determine if the RETRAN-predicted reactivity feedback model is conservative. The VIPRE code was used in the core thermal-hydraulic analysis to determine if DNB occurs.

Although a major break in a pipeline is classified as an American Nuclear Society (ANS) Condition IV event, the event was analyzed to meet Condition II criteria. The only criterion that may be challenged during this event is the one that states that the critical heat flux should not be exceeded. The evaluation shows that this criterion is met by ensuring that the minimum DNBR does not go below the limit value at any time during the transient.

For the limiting steam line break, with offsite power case evaluation, the following assumptions include:

1. Hot standby initial conditions are assumed with the application of conservative uncertainties.

2. SI is activated on either low pressurizer pressure or low compensated steam line pressure, with a 2 second time delay.

3. A limiting end-of-life shutdown margin of 1.3% Δk is assumed.

4. A limiting negative end-of-life moderator coefficient is assumed.

5. The core power distribution is assumed to be uniform and the two channels have equal reactivity coefficient weighting. This is true despite the faulted loop channel accounting for only ¼ of the core volume. This is conservative because it accentuates the feedback in the cold/faulted loop channel.

6. Minimum boron injection capability is assumed corresponding to the most restrictive single failure in the SI system.

7. SI flow is assumed to be delivered to the RCS 52 seconds after the initiating signal.

8. AFW is activated immediately as the transient begins.

9. Steam line isolation (MSIV closure) is assumed to occur beginning 2 seconds after a low compensated steam line pressure signal is received. The MSIV closes in a 15 second linear ramp.

10. Feedwater is isolated when SI is initiated, with a 2 second time delay.

11. The Doppler feedback curve is based on USAR Figure 15.1-14.

12. A single failure of one SI system train is assumed.

Results

The time sequence of events for the postulated steam line rupture accidents with offsite power is presented in Table 4.3-1.

Figures 4.3-1 through 4.3-12 show the response of pertinent system parameters following a main steam line rupture with offsite power available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one SG.

As shown in Figure 4.3-7, the core attains criticality with the RCCAs inserted, assuming the design shutdown margin and one stuck RCCA. Criticality occurs before boric acid solution enters the RCS from the SI System that is injecting fluid from the refueling water storage tank (RWST). The continued addition of boron results in a peak core power well below the nominal full power value (Figure 4.3-1).

Conclusions

A DNB analysis was performed for the steam line break case described above. The result of the DNBR calculation is depicted in Figure 4.3-13. The analysis demonstrated that the minimum DNBR remains above the limit value of 1.50 and, thus, it is concluded that the DNB design basis is met for the steam line break event initiated from zero power with the replacement MSIVs and MFIVs and associated actuators.

Table 4.3-1 Time Sequence of Events for Rupture of a Main Steam Line

Event	Time (Seconds)
Steam Line Ruptures	0.001
Low Steam Line Pressure Setpoint Reached	0.78
Steam Line Isolation Begins	2.86
Steam Line Isolation Complete, Feed Line Isolation	17.86
SI Begins	52.86
Minimum DNBR Occurs	118.90
Peak Core Thermal Power Occurs	124.00

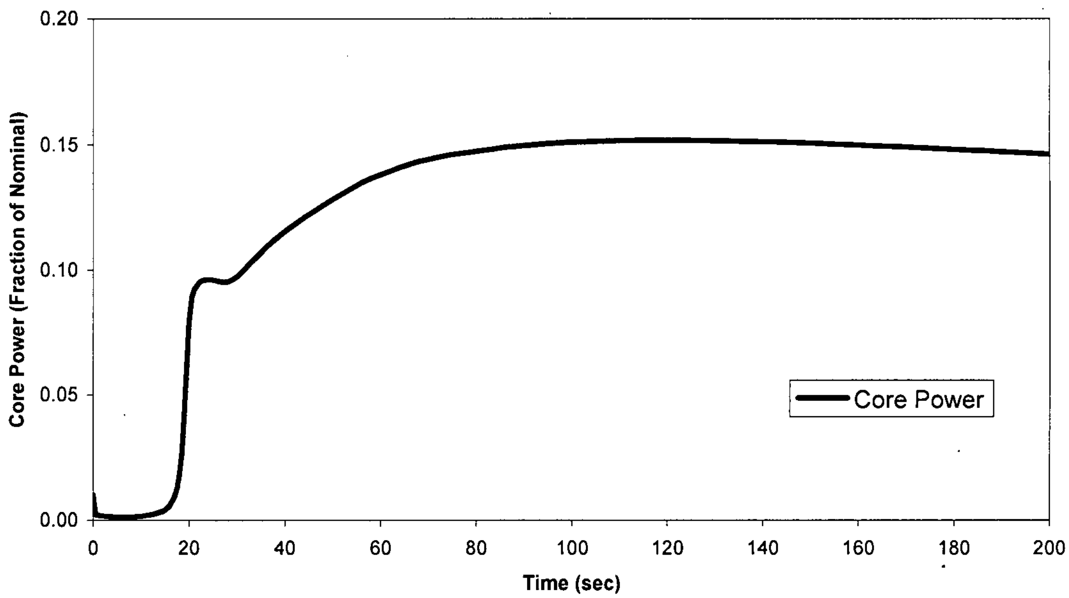


Figure 4.3-1 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Power versus Time

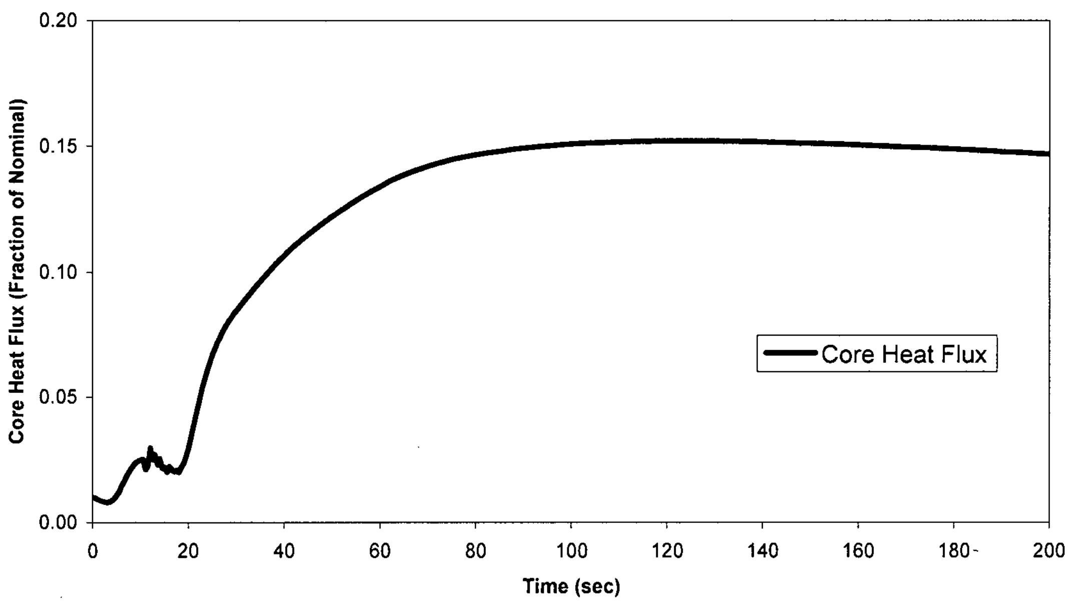


Figure 4.3-2 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture –Core Heat Flux versus Time

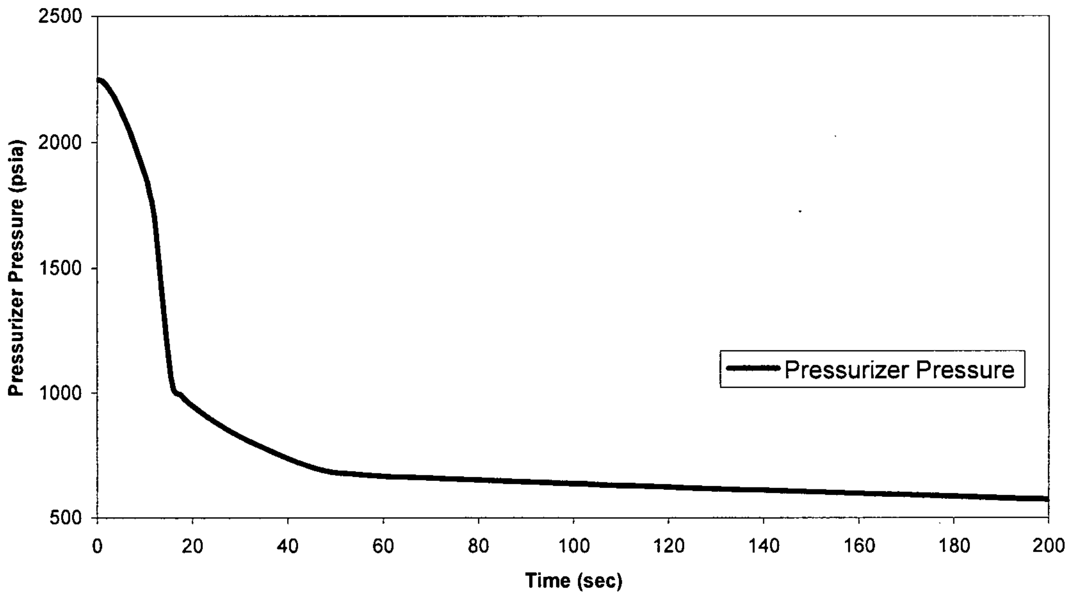


Figure 4.3-3 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Pressurizer Pressure versus Time

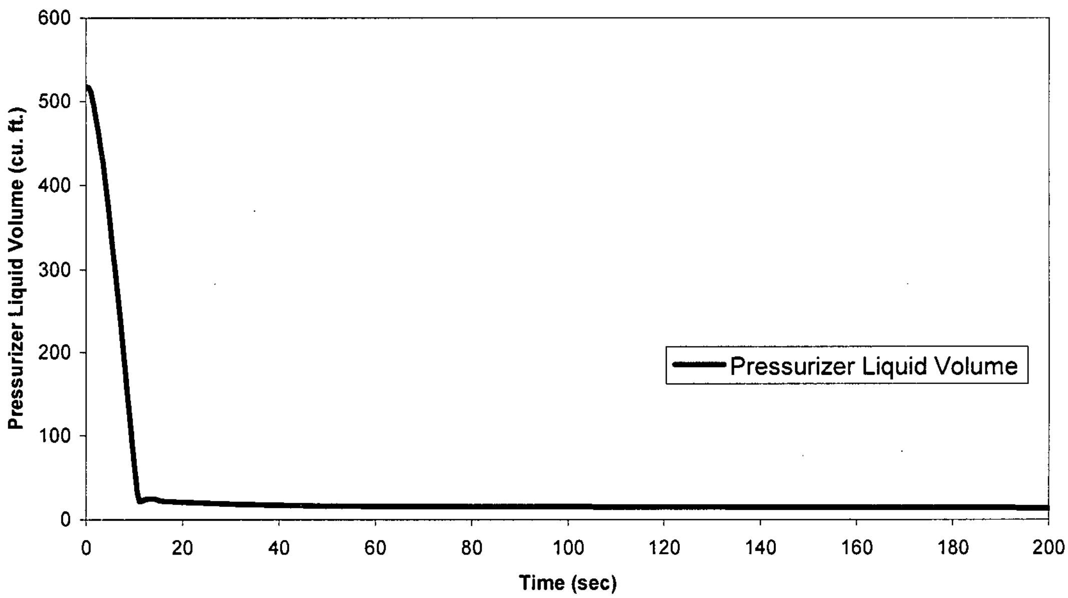


Figure 4.3-4 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Pressurizer Liquid Volume versus Time

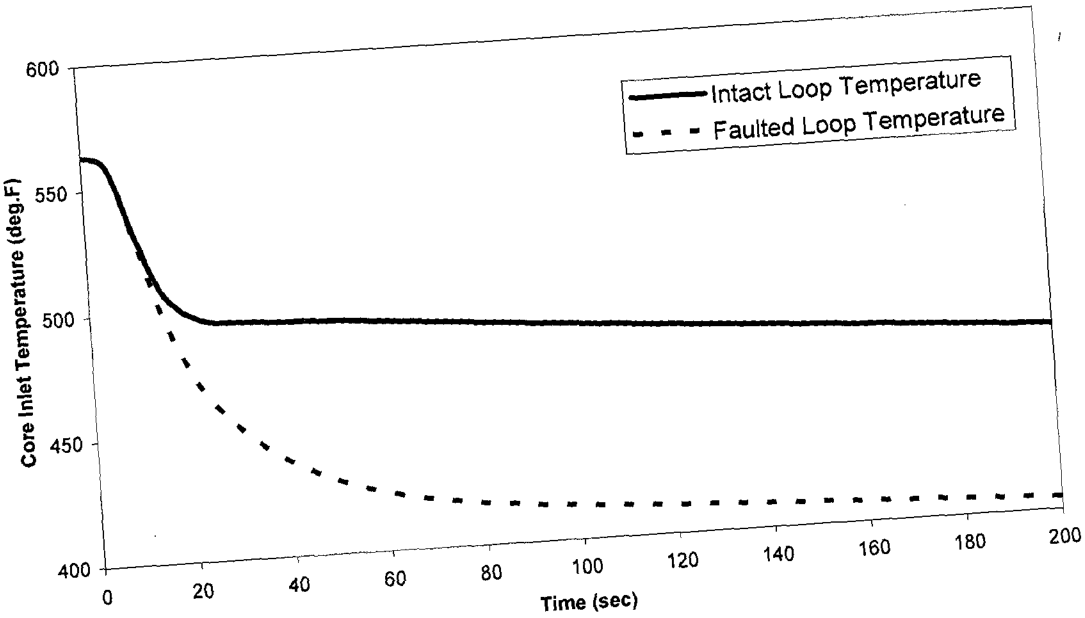


Figure 4.3-5 **Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Inlet Temperature versus Time**

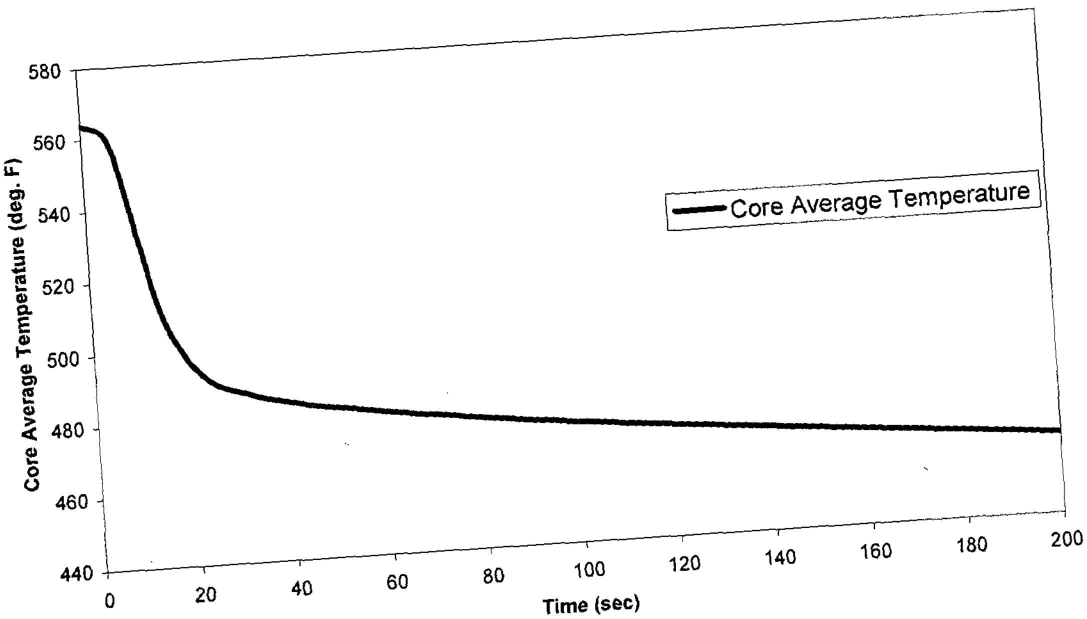


Figure 4.3-6 **Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Average Temperature versus Time**

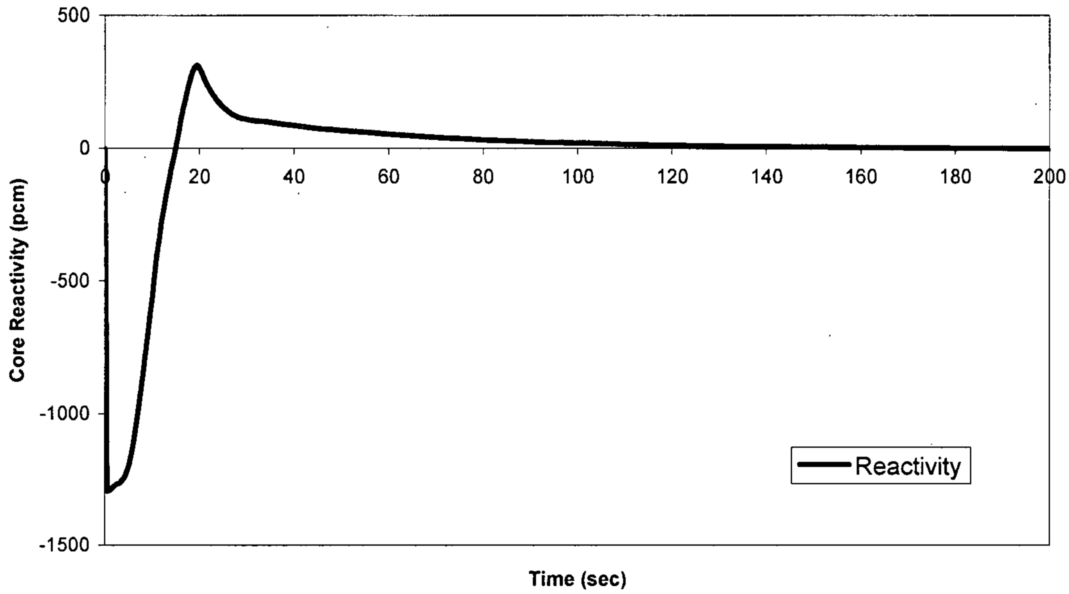


Figure 4.3-7 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Reactivity versus Time

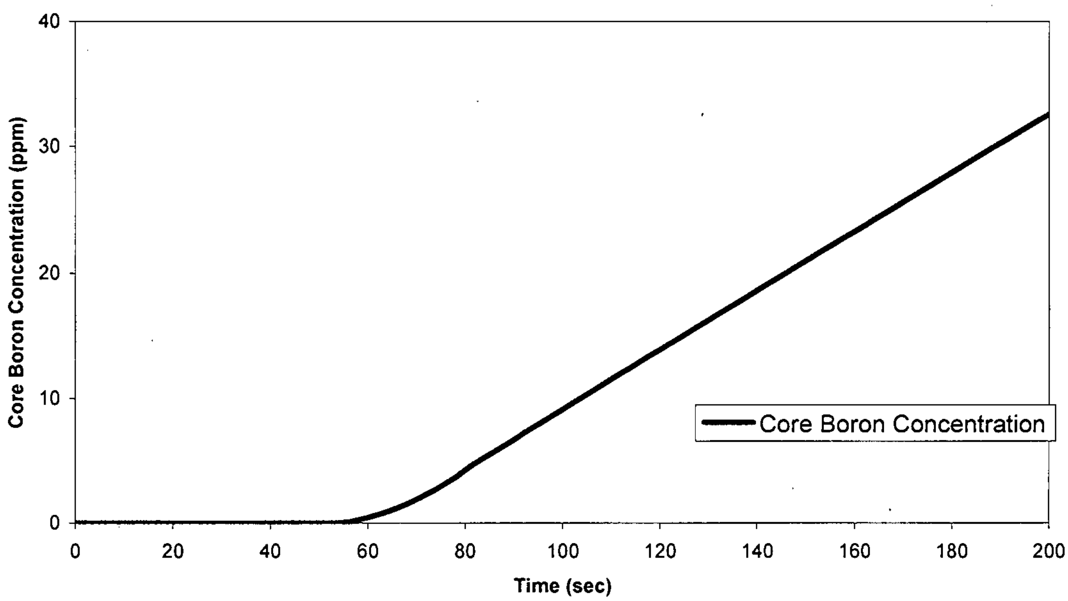


Figure 4.3-8 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture –Core Boron Concentration versus Time

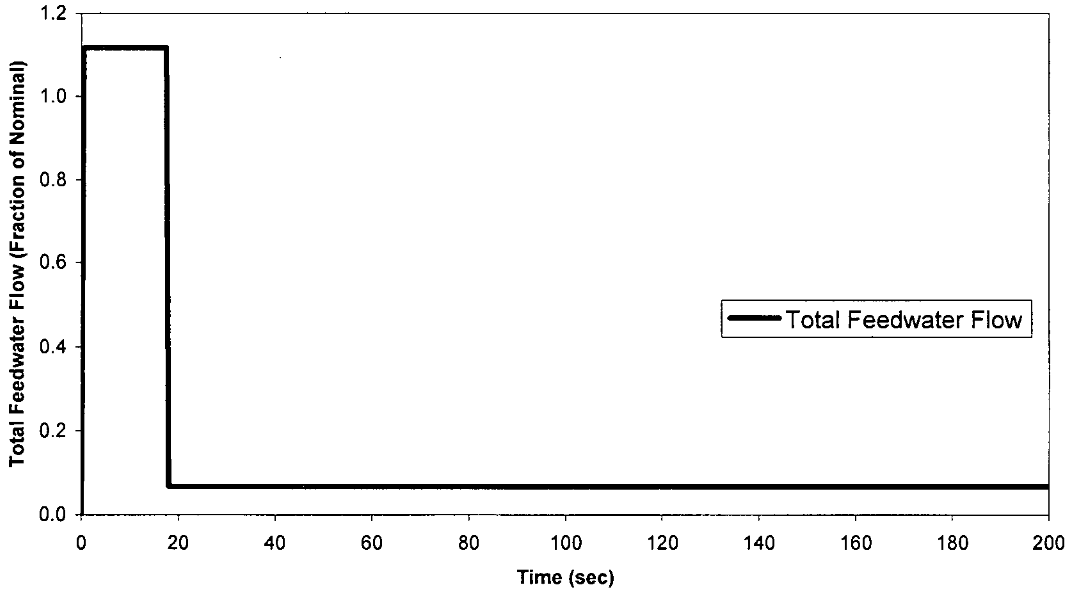


Figure 4.3-9 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – Feedwater Flows versus Time

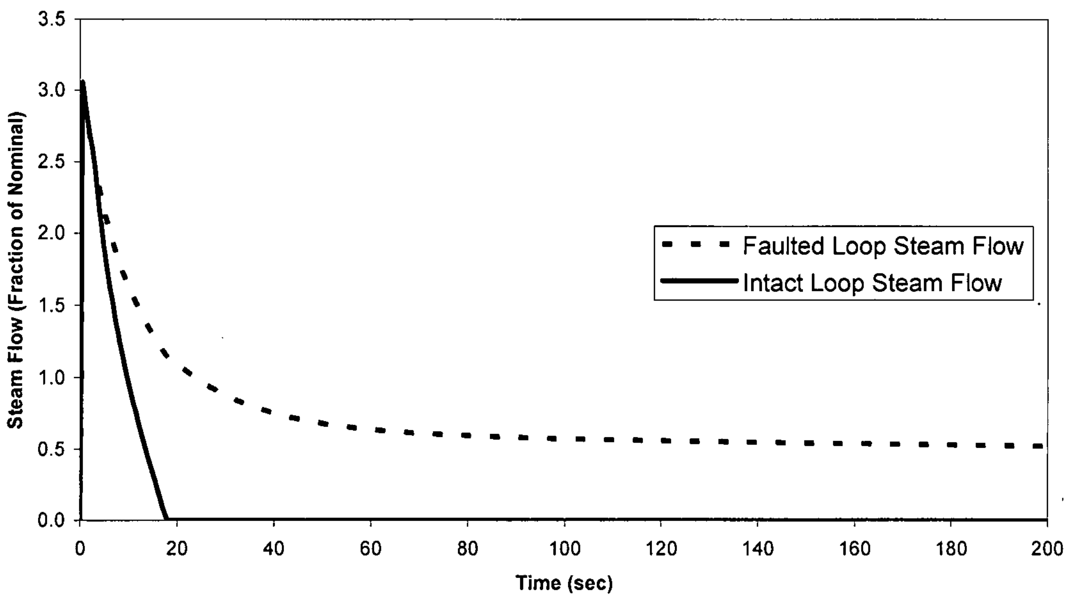


Figure 4.3-10 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture –Steam Flows versus Time

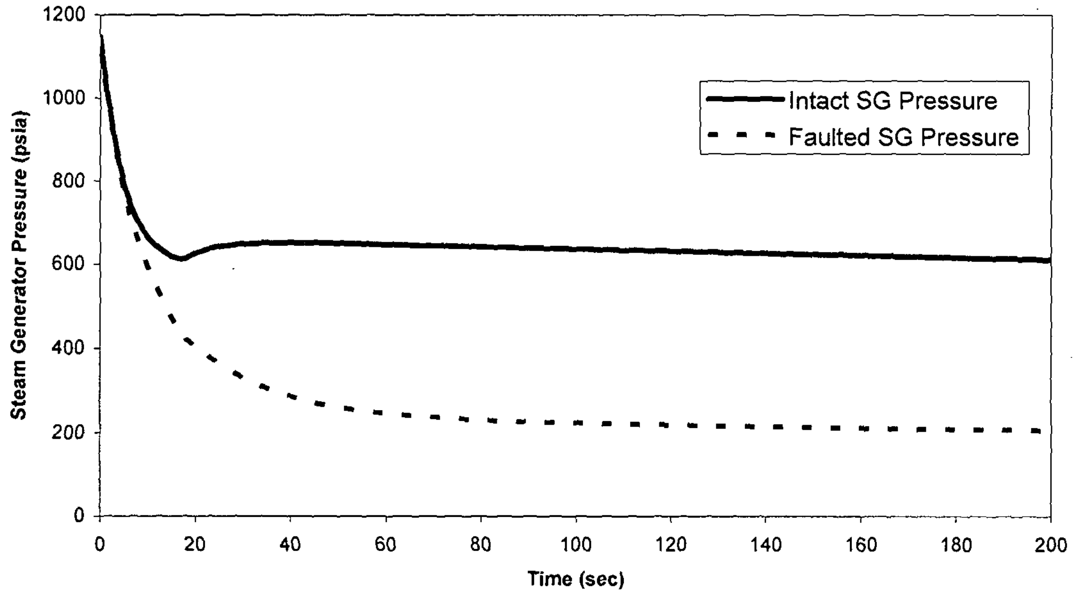


Figure 4.3-11 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture – SG Pressures versus Time

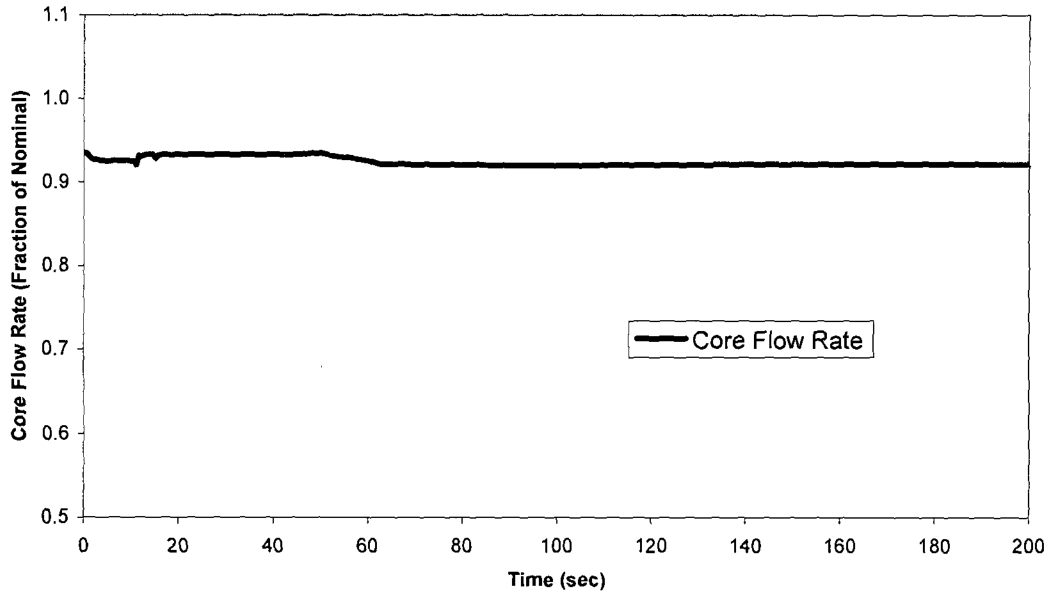


Figure 4.3-12 Hot Zero Power Steam Line Break Transient with Offsite Power Double-Ended Rupture –Core Flow Rate versus Time

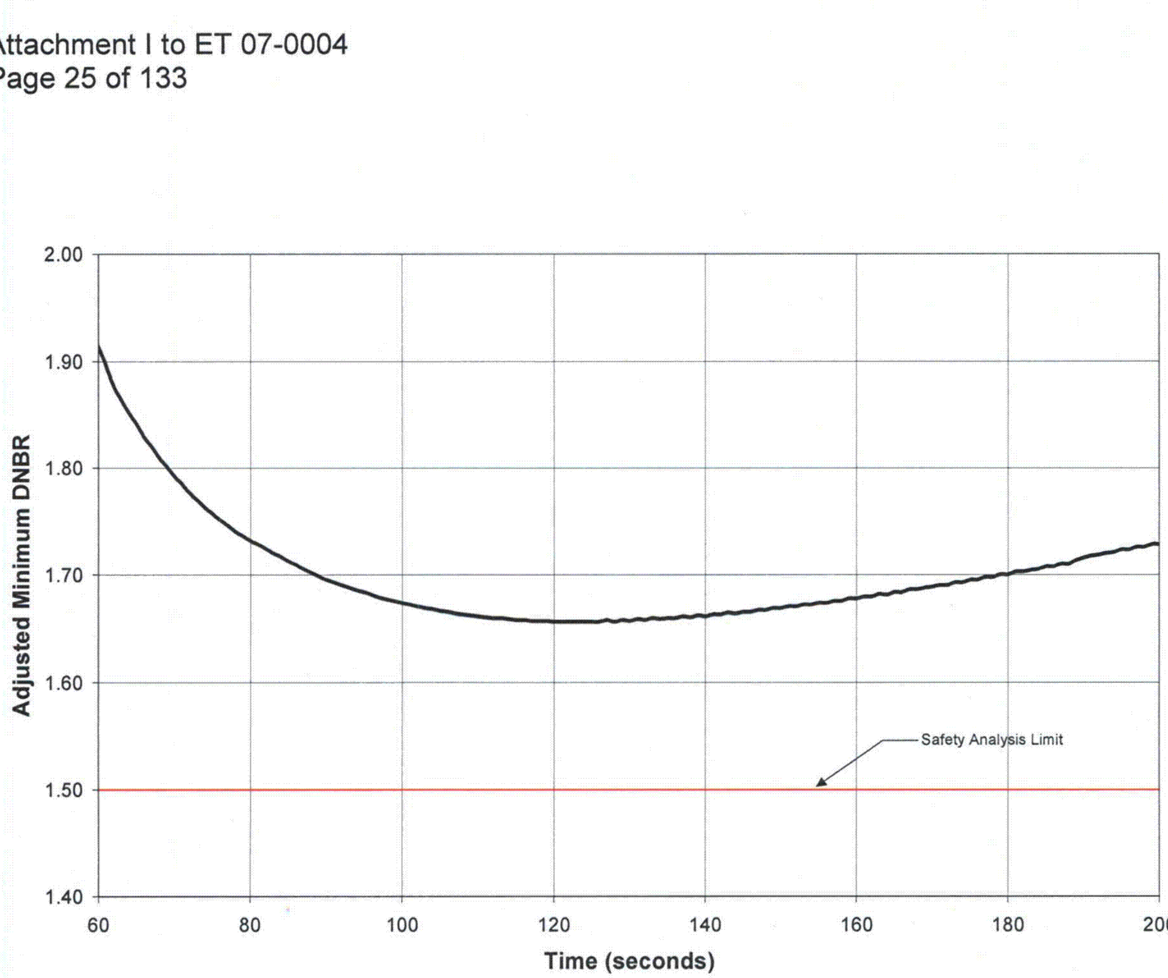


Figure 4.3-13: Adjusted Minimum DNBR for Steam Line Break

4.4 Feedwater system pipe break (USAR Section 15.2.8)

Introduction

The feedwater system pipe break has been reanalyzed to support the MSIV and MFIV and associated actuator replacement, in which the closure characteristics of the valves are different compared to the current licensing basis analysis of record. Currently, it is assumed that these valves begin closing following a 2 second time delay, and ramp closed linearly in 5 seconds. For reanalysis purposes, a 15 second stroke time for valve closing, consistent with other related analyses, has been assumed. For this particular transient, feedwater is terminated as part of the initiating sequence of events so MFIV characteristics are not relevant.

In addition, the reanalysis addressed an issue identified in Reference 5 relating to SG water level uncertainty. The feedwater line break event can potentially be affected by the SG level uncertainty; specifically, if the water level is high, it will delay the trip on Low-Low SG Water Level, prolonging the RCS heat-up and pressurization prior to trip. This will also prolong the moderator feedback induced power increase (Beginning of Life (BOL) moderator temperature coefficient is positive). For this reanalysis, the SG water level is assumed to be 55.7% of the narrow range, a 5.7% increase from nominal, to conservatively delay the reactor trip.

Transient Description

A feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the SGs to maintain shell side fluid inventory. If the postulated break occurs in the feedwater line between the check valve and the SG, then at least one SG will blow down in an uncontrolled manner. Further, a break in this location would prevent the subsequent addition of AFW to the faulted SG because the AFW piping connects to the main feedwater line. Consequently, this location is used in determining the most severe consequences of a feedwater line break accident.

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either RCS cooldown or heatup. Because the consequences of an RCS cooldown resulting from an feedwater line break are bounded by the cooldown consequences of a steam system piping failure (see Section 4.5), the initial and boundary conditions used in this reanalysis are chosen to maximize the RCS pressurization and heatup.

A feedwater line rupture reduces the ability of the secondary system to remove heat generated by the core from the RCS for the following reasons:

1. Feedwater flow to the SGs is reduced. Since the feedwater is subcooled, its loss will cause the SG pressure and temperature to increase and subsequently increases the reactor coolant temperature prior to reactor trip. For this transient, complete feedwater loss is assumed as part of the initiating sequence of events.

2. Mass inventory of the faulted SG may be discharged through the break and therefore would not be available for decay heat removal after reactor trip.

The feedwater line break accident is an ANS Condition IV event and the accident is analyzed to assure that the RCS does not overpressurize and that the plant can be brought to a safe shutdown condition. This is accomplished primarily by the AFW System as the Main Feedwater System is assumed to be unavailable, along with significant SG water inventory. The system must ensure that:

1. No substantial overpressurization of the RCS occurs (RCS pressure does not exceed 110% of design).

2. Sufficient cooling is provided such that the RCS remains subcooled at the turn-around time (time at which the AFW heat removal capacity equals the RCS thermal power).

The severity of the feedwater line break accident depends on a number of system parameters, including break size, initial reactor power level, and credit taken for various control and safety systems. The accident which results in the most severe consequences is the basis for this analysis and is defined below.

To achieve the most limiting accident results, the feedwater line rupture is assumed to occur with the plant at power and SG level at its lowest possible level (Low-Low SG Water Level reactor trip set point). This condition is brought about by first assuming the Main Feedwater System fails, stopping flow to all SGs. This failure occurs when the SG levels are conservatively high. Then, the water level in all SGs decreases equally. This induces RCS heat-up, and positive reactivity feedback (BOL) causes the power to rise. The initially high SG level prolongs this period during which the power rises. The water level decreases until the Low-Low SG Water Level setpoint is reached, at which point the reactor trips and a double-ended rupture of the largest feedwater line is assumed. These assumptions ensure that the most severe feedwater line break accident is analyzed.

Assumptions/Methods

For the most part, the assumptions and methodology used in this reanalysis are consistent with the current licensing basis analysis of record. The accident was analyzed using the RETRAN-3D computer code operating in the RETRAN-02 mode. The RETRAN-3D model addresses overpressurization and overfill concerns and is also used to establish that no bulk boiling occurs prior to the time when the heat removal capability of the SGs, being fed by AFW, exceeds nuclear steam supply system (NSSS) heat generation. This conservatively ensures that the core remains covered with water and thereby will remain in place and geometrically intact with no loss of core cooling capability.

The major assumptions used in the analysis are summarized below:

1. The initial plant power is assumed to be 102% of rated core power.

2. No credit is taken for the pressurizer pressure control (i.e., pressurizer power operated relief valves (PORVs) or pressurizer spray).

3. Initial pressurizer level is at the nominal programmed value plus 5% of span (error).

4. Initial SG water level in all SGs is at the nominal value, plus 5.7% of the narrow range span.

5. Main feedwater is assumed to be lost to all SGs at event initiation due to a malfunction in the feedwater control system under an adverse environment. The feedwater line break, and subsequent reverse blowdown of the faulted SG, is conservatively assumed to occur when the SG inventory reaches 0% narrow range span (i.e., reactor trip on Low-Low SG Water Level of 23.5%, minus environmental allowance and uncertainties). The combination of errors described above yields the most severe feedwater line break transient, with control and protection interaction considered.

6. The AFW System is actuated by the Low-Low SG Water Level signal. A 60 second time delay was assumed following the Low-Low SG Water Level signal to allow time for startup of the standby diesel generators and the AFW pumps. An additional 314 seconds was assumed before the feedwater lines were purged and the relatively cold AFW entered the intact SGs.

7. Failure of one protection train is taken as the worst single failure so that the second motor driven AFW pump is assumed inoperable. The total AFW flow delivered to the three intact SGs is assumed to be 563 gpm. Credit is taken for the discharge flow control device installed on the AFW header common to both the motor driven and turbine driven AFW pumps. Due to this discharge flow control device, the intact SG receiving AFW from both the motor driven and turbine driven AFW pumps is assumed to receive no more than 250 gpm. The remaining two intact SGs which receive AFW from only the turbine driven AFW pump are assumed to receive approximately 157 gpm each.

Two separate cases are analyzed, i.e., with and without offsite power. Where offsite power is available, the reactor coolant pumps continue to operate after the reactor trip. In the case where offsite power is not available, the reactor coolant pumps coast down upon loss of offsite power which is assumed to occur at the reactor trip.

Acceptance Criteria

The acceptance criteria used herein is identical to that used in the current USAR and is consistent with the guidelines provided in the Standard Review Plan, Section 15.2.8. The acceptance criteria is as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures.

2. Any fuel damage that may occur during the transient should be of a sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

3. Any activity released must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

These criteria are conservatively met by insuring that no primary side hot leg boiling occurs prior to "turnaround" following a feedwater line rupture. The "turnaround" time is reached when the heat removal capability of the SGs, being fed by AFW, exceeds the core heat generation and the reactor coolant pump heat addition.

Results

Figures 4.4-1 through 4.4-12 show the significant plant responses following a feedwater line break with offsite power available. Similarly, Figures 4.4-13 through 4.4-24 show the plant responses for the case where offsite power is lost following reactor trip. The calculated sequences of events for both cases are listed in Table 4.4-1. The system responses following the feedwater line break are similar for both cases analyzed. The results show that following reactor trip, the plant remains subcritical and the pressures in the RCS and Main Steam System remain below 110 percent of the respective design pressures.

In both cases, the primary and secondary pressures increase prior to reactor trip. After reactor trip occurs on Low-Low SG Water Level, the pressure decreases sharply, due to the cooldown caused by the break, until steam line isolation occurs. The pressure in the faulted SG continues to decrease, while the pressure in the intact SGs and primary side begins to increase (cooldown ends, heat-up begins) until the safety valve settings are reached.

The primary temperatures increase prior to reactor trip and decrease sharply during cooldown after reactor trip. Once the heat-up begins, the primary temperature increases until the heat removal capability of the intact SGs, with the inventory maintained by the AFW System, equal the decay heat generated in the core plus pump heat ("turnaround" time). The peak primary temperature remains below the saturation temperature although the margin to boiling is decreased. Thus, there is no bulk boiling in the RCS.

In addition, the pressurizer does not fill due to thermal expansion nor does the pressurizer empty. Therefore, the reactor remains covered with water throughout the transient.

Conclusions

The results of the feedwater line rupture analysis show that all applicable acceptance criteria are satisfied. The AFW System is capable of removing the stored energy, residual decay heat, and RCP heat. This prevents overpressurizing the RCS and Main Steam System and uncovering the reactor core.

Table 4.4-1 Time Sequence of Events for a Main Feed Line Break

Actions/Events	With offsite power available	Without offsite power available
Steady State Operation	0.00	0.00
Feedwater Control System Failed	10.00	10.00
Pressurizer Safety Valve Opens	31.74	31.74
Low-Low SG Water Level Setpoint Reached	54.10	54.10
Reactor Trip on Low-Low SG Water Level and Feedwater Line Break Occurs	56.13	56.13
SG Safety Valve #1 Opens	56.30	56.30
SG Safety Valve #2 Opens	56.34	56.34
Pressurizer Safety Valve Closes	60.15	60.84
Steam Line Low Pressure Setpoint Reached in Faulted SG	86.00	83.12
SG Safety Valve #2 Closes	86.65	83.85
SG Safety Valve #1 Closes	88.25	85.22
All MSIVs Closed	103.00	100.12
Centrifugal Charging Pump Begins to Deliver Injection Flow on Low Pressurizer Pressure	138.85	157.48
SG Safety Valve #1 Opens	236.49	236.54
AFW is Delivered to Intact SGs	428.10	428.10
Pressurizer Safety Valve Opens	424.31	427.46
Pressurizer Safety Valve Closes	1792.37	927.10
Core Decay Heat Decreases to AFW Heat Removal Capacity	1775.00	868.00

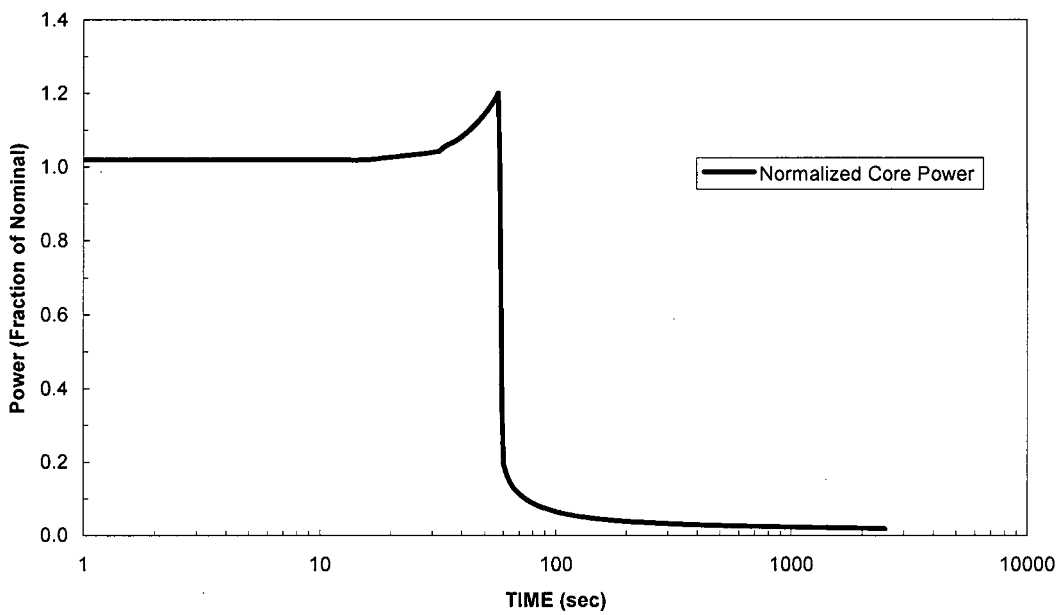


Figure 4.4-1: Normalized Core Power (Case MFLB1 R3D)

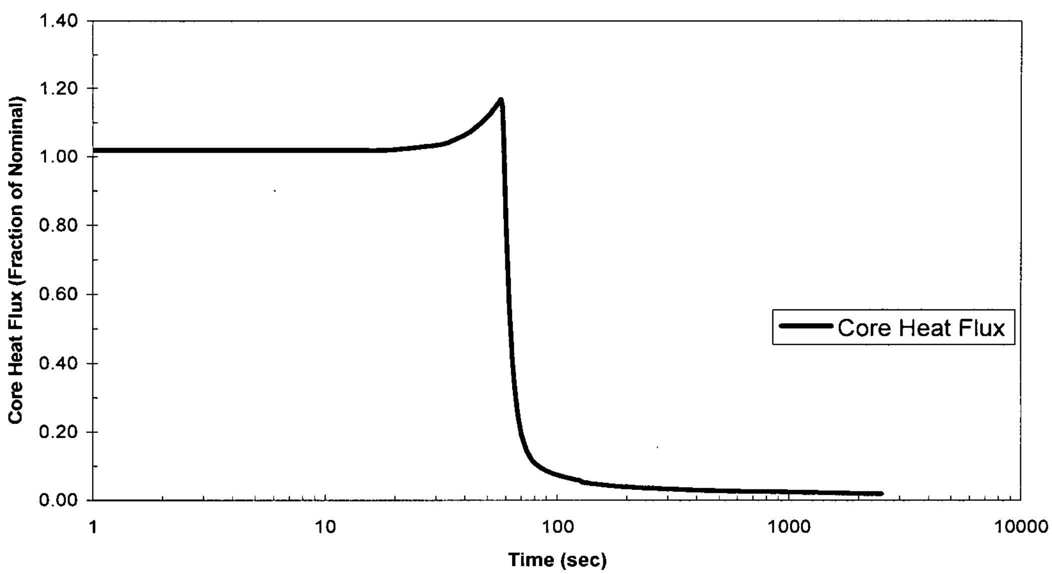


Figure 4.4-2: Core Heat Flux (Case MFLB1 R3D)

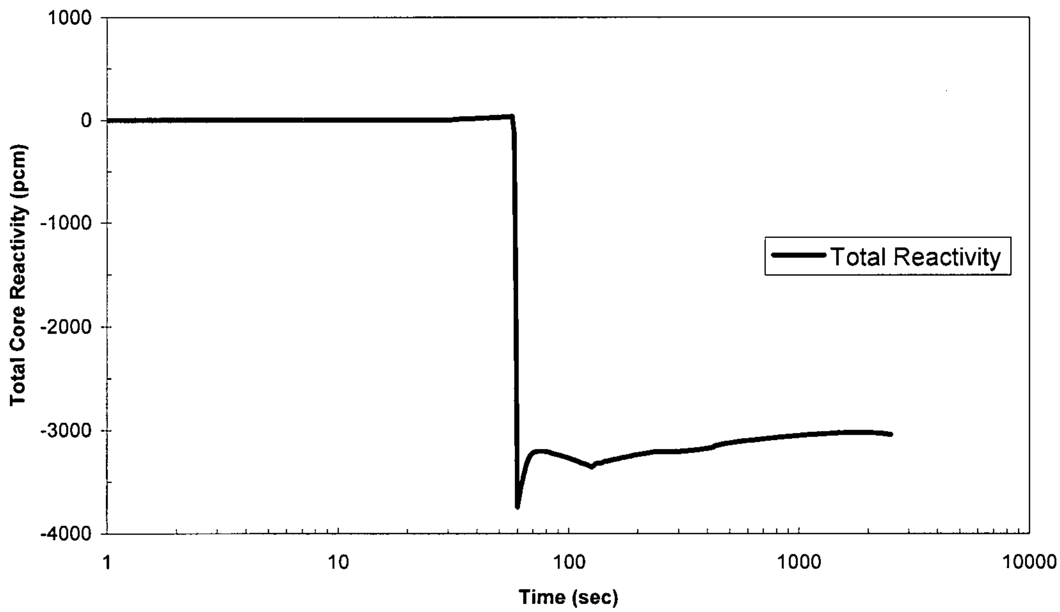


Figure 4.4-3: Total Core Reactivity (Case MFLB1 R3D)

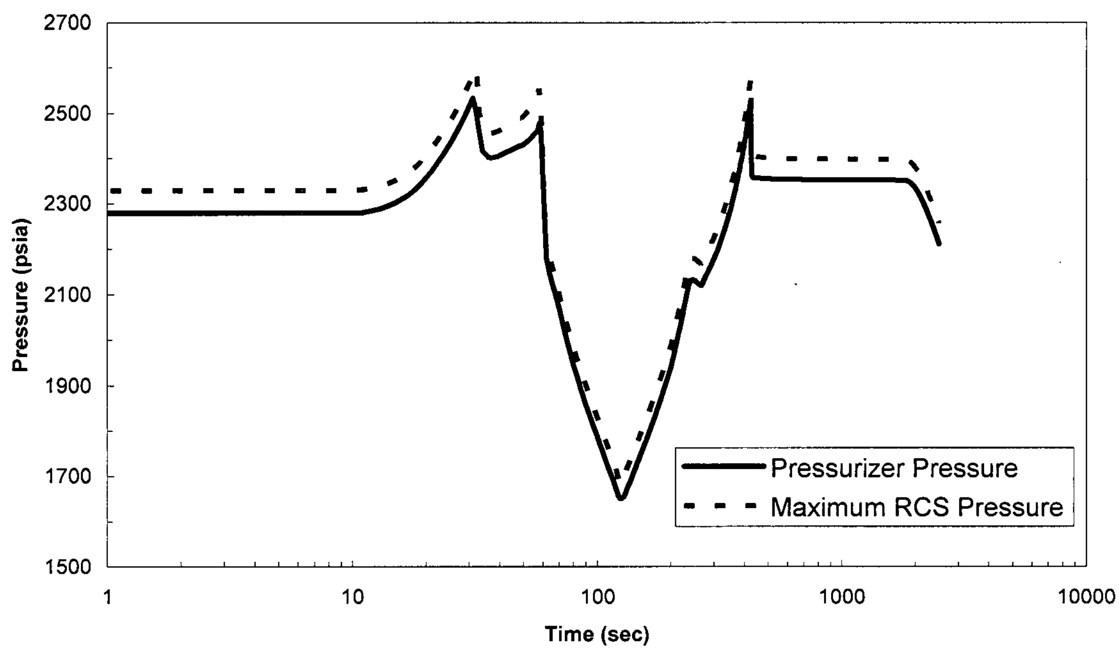


Figure 4.4-4: RCS Pressures (Case MFLB1 R3D)

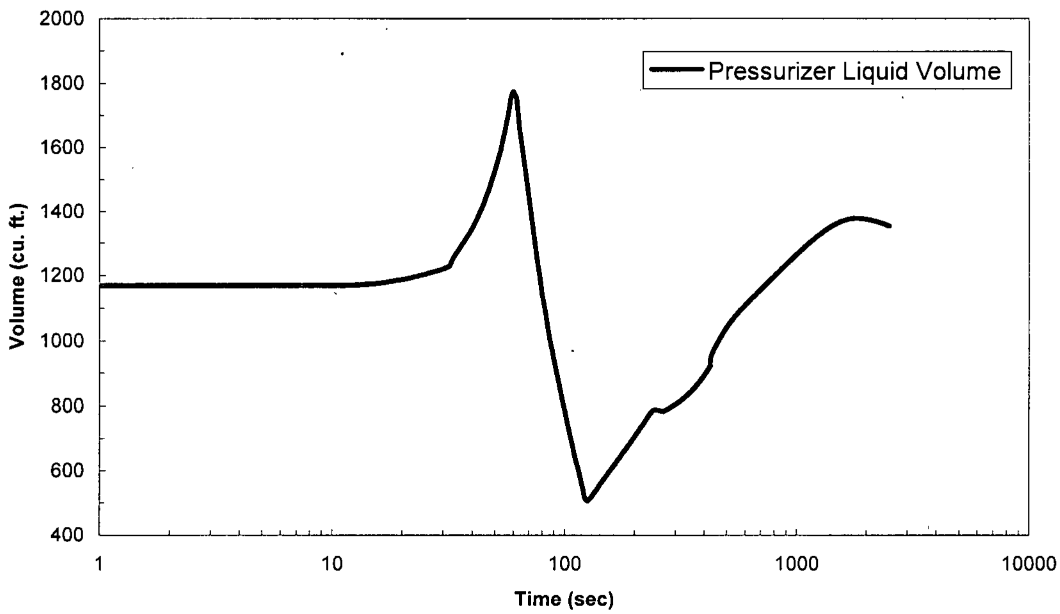


Figure 4.4-5: Pressurizer Liquid Volume (Case MFLB1 R3D)

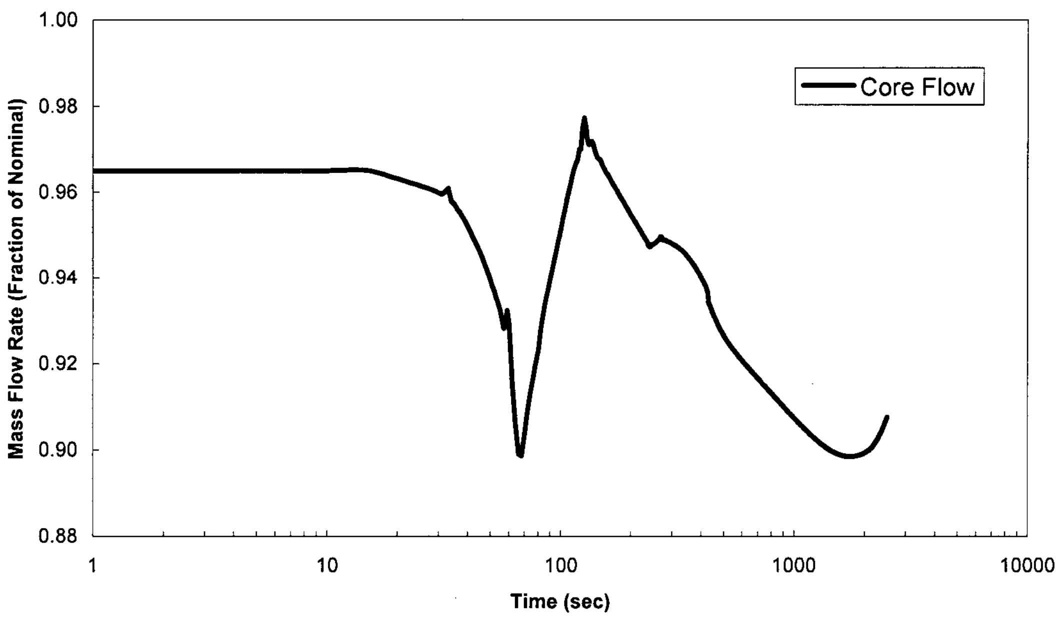


Figure 4.4-6: Core Flow (Case MFLB1 R3D)

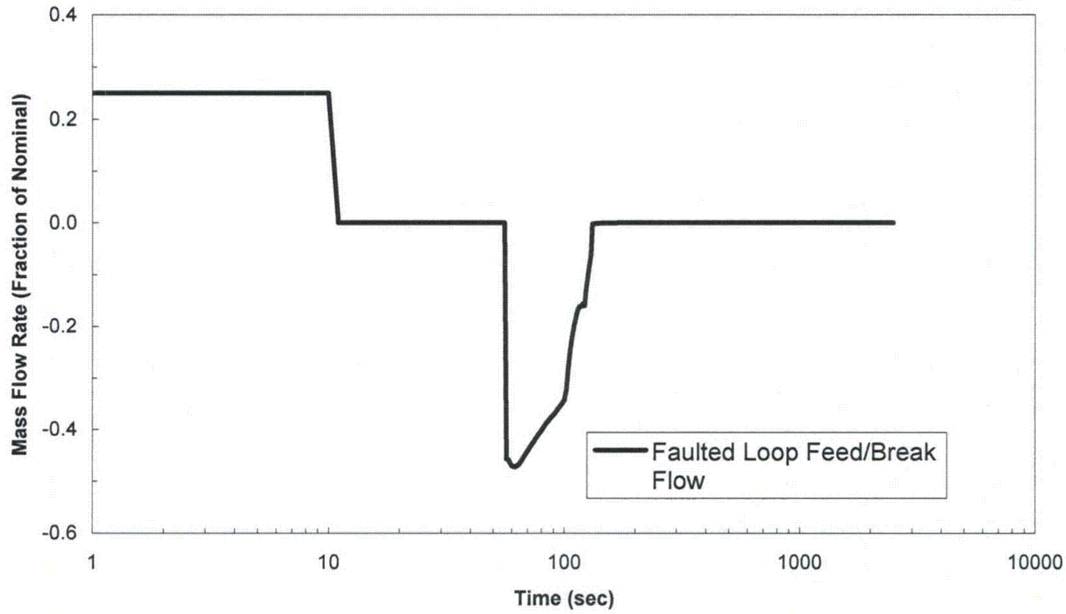


Figure 4.4-7: Faulted SG Feed/Break Flow (Case MFLB1 R3D)

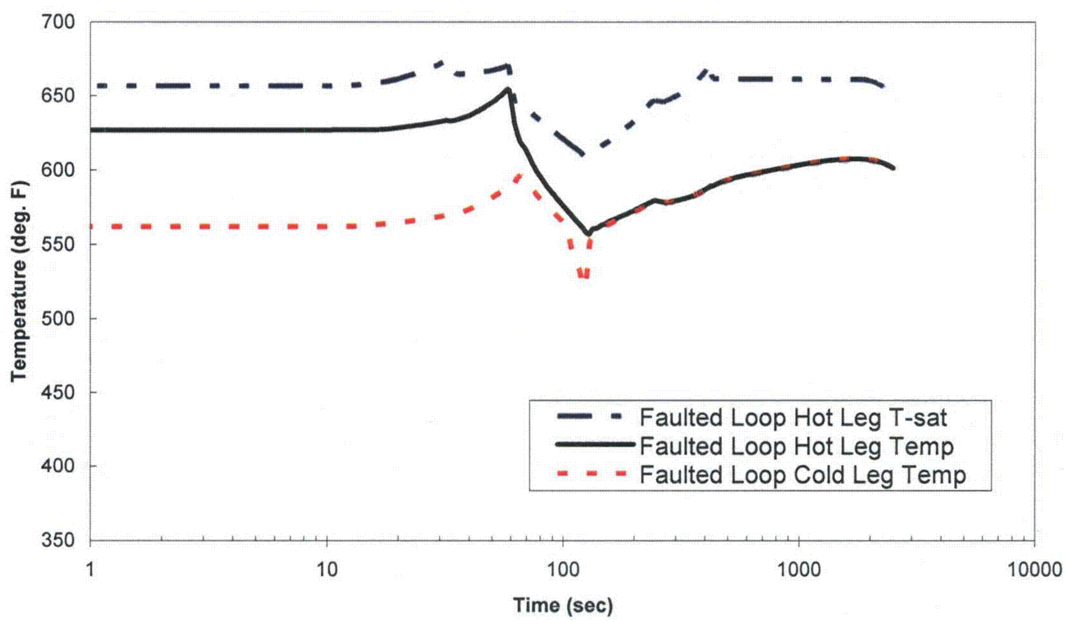


Figure 4.4-8: RCS Faulted Loop Temperatures (Case MFLB1 R3D)

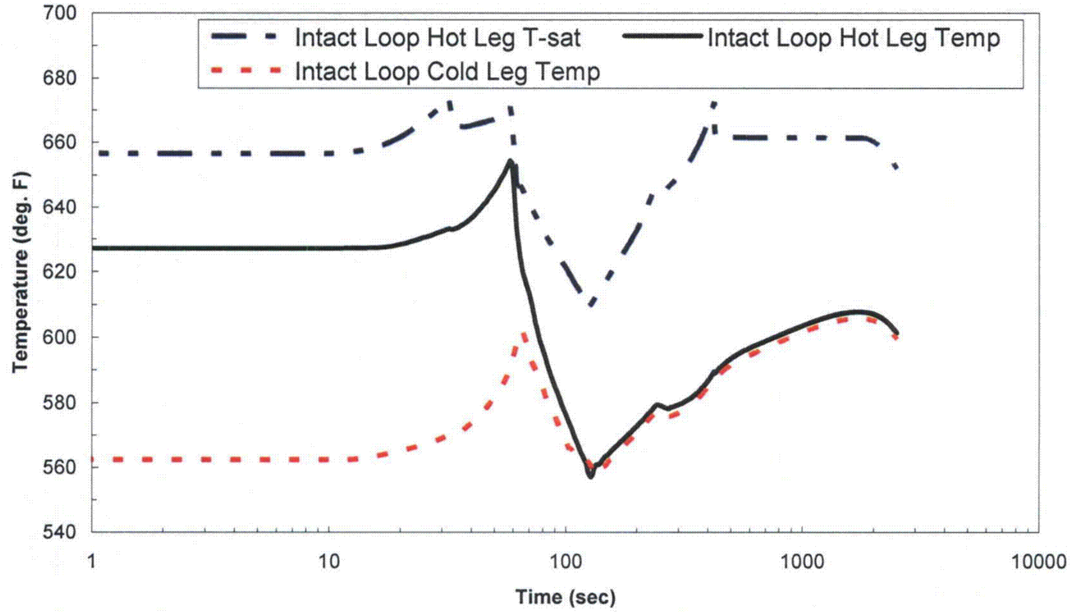


Figure 4.4-9: RCS Intact Loop Temperatures (Case MFLB1 R3D)

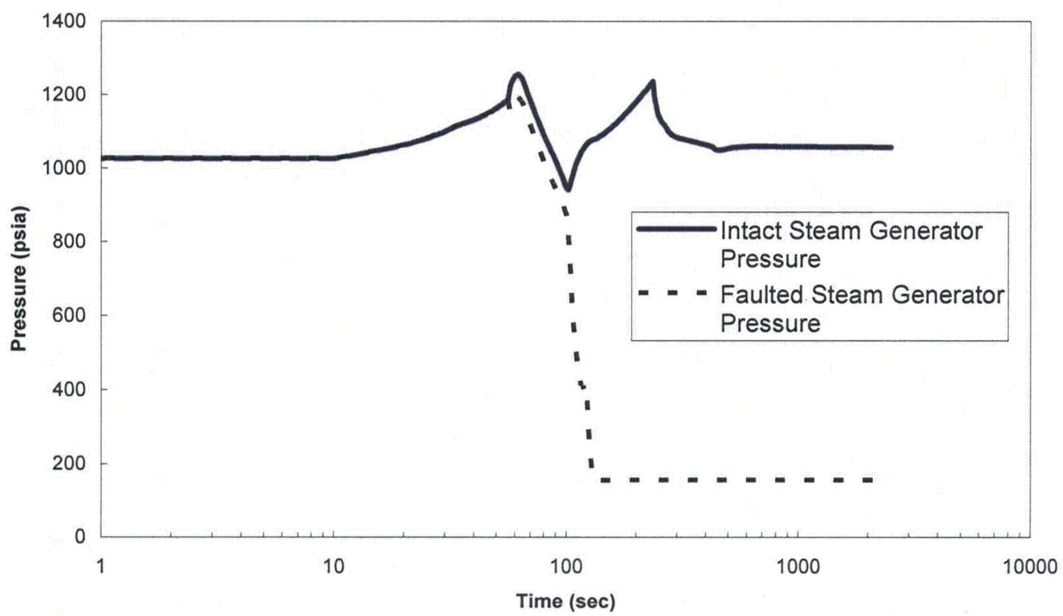


Figure 4.4-10: SG Pressures (Case MFLB1 R3D)

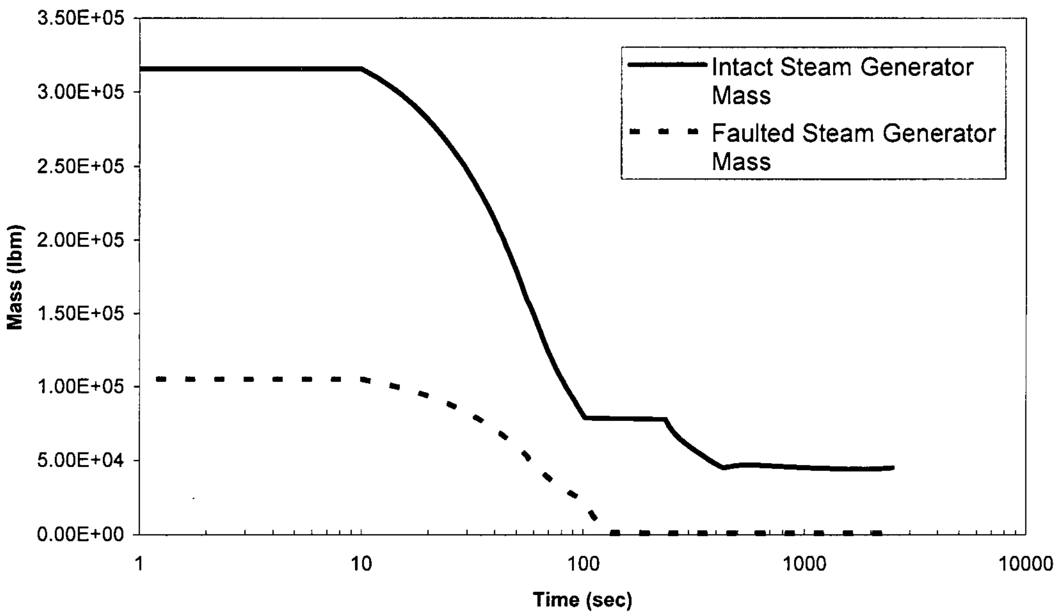


Figure 4.4-11: SG Mass (Case MFLB1 R3D)

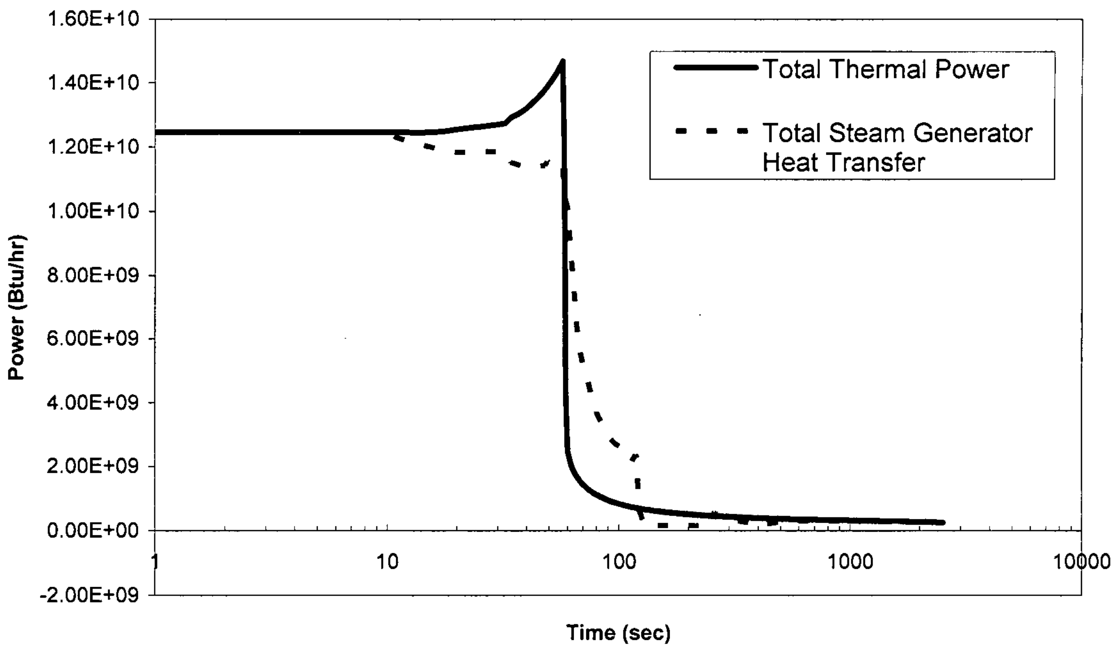


Figure 4.4-12: Energy Balance (Case MFLB1 R3D)

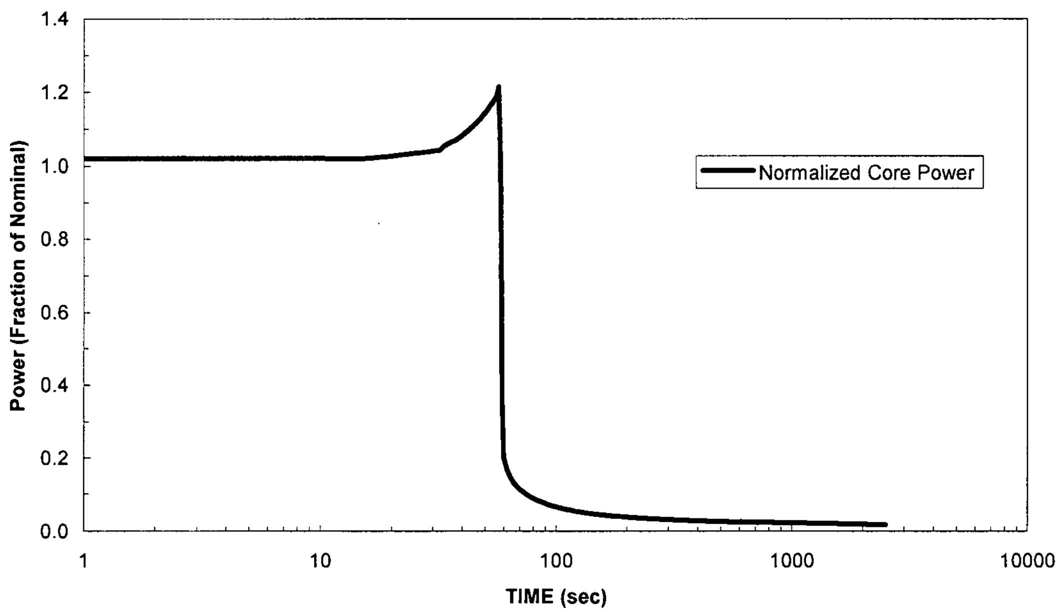


Figure 4.4-13: Normalized Core Power (Case MFLB2 R3D)

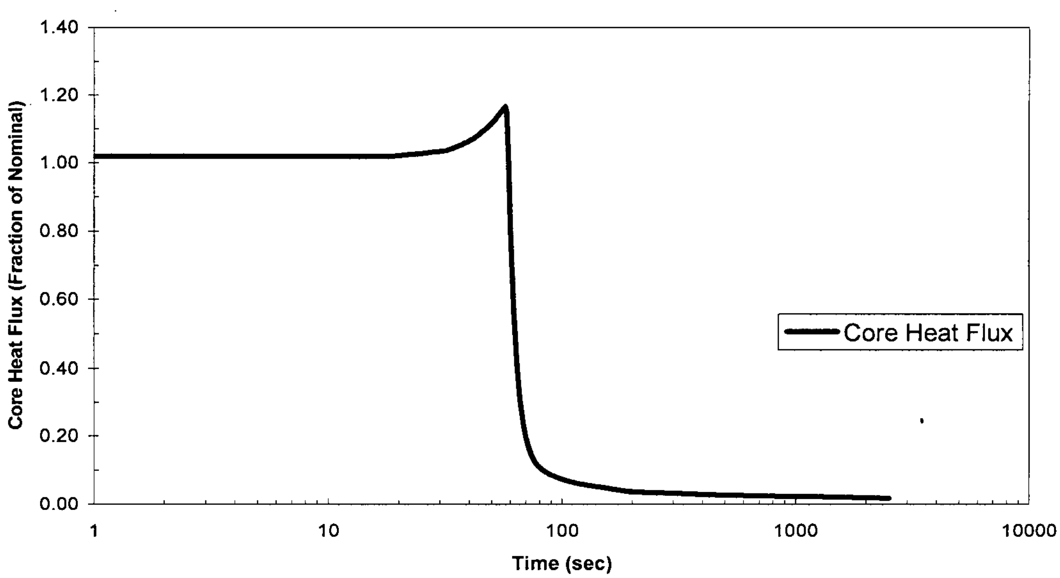


Figure 4.4-14: Core Heat Flux (Case MFLB2 R3D)

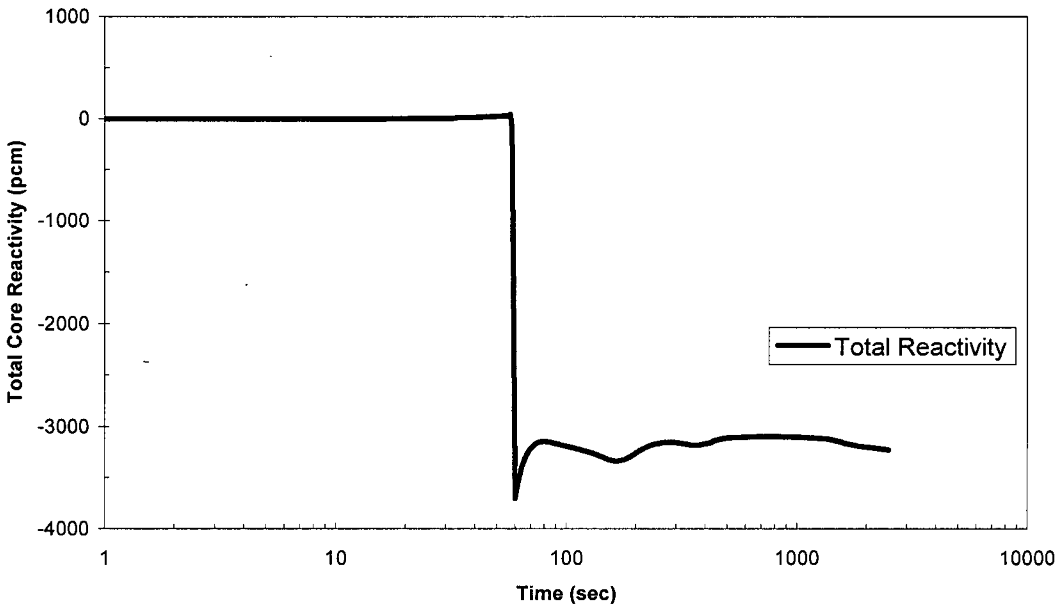


Figure 4.4-15: Total Core Reactivity (Case MFLB2 R3D)

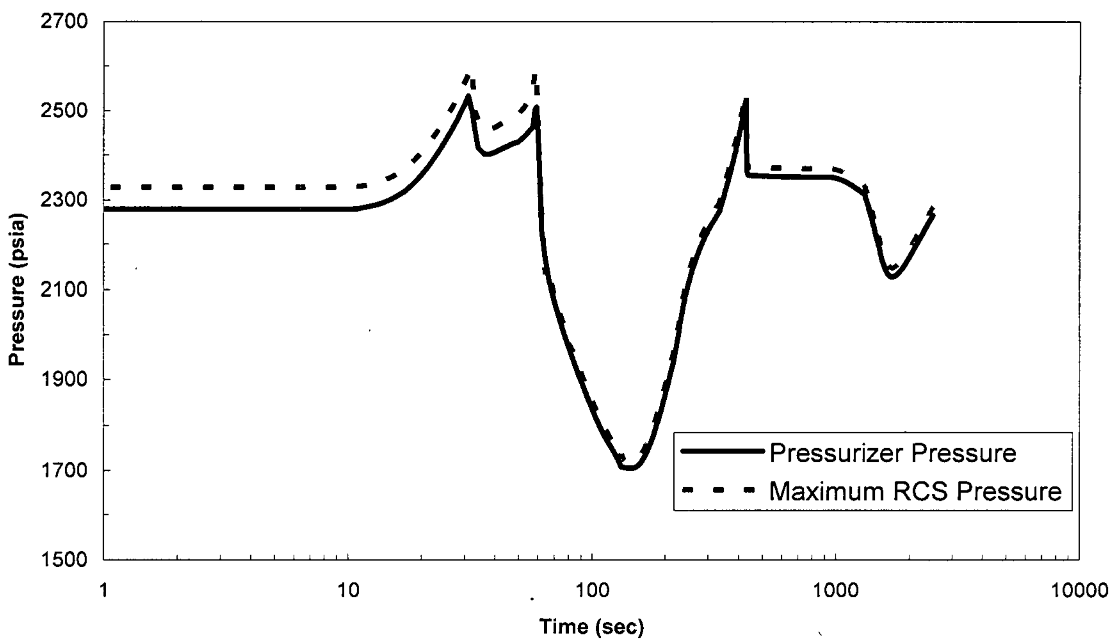


Figure 4.4-16: RCS Pressures (Case MFLB2 R3D)

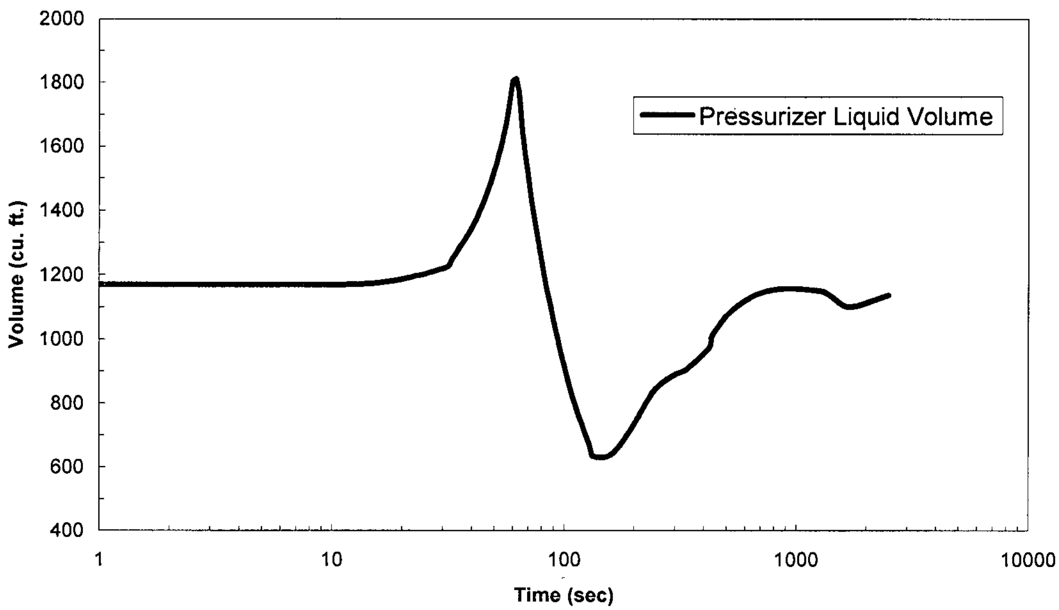


Figure 4.4-17: Pressurizer Liquid Volume (Case MFLB2 R3D)

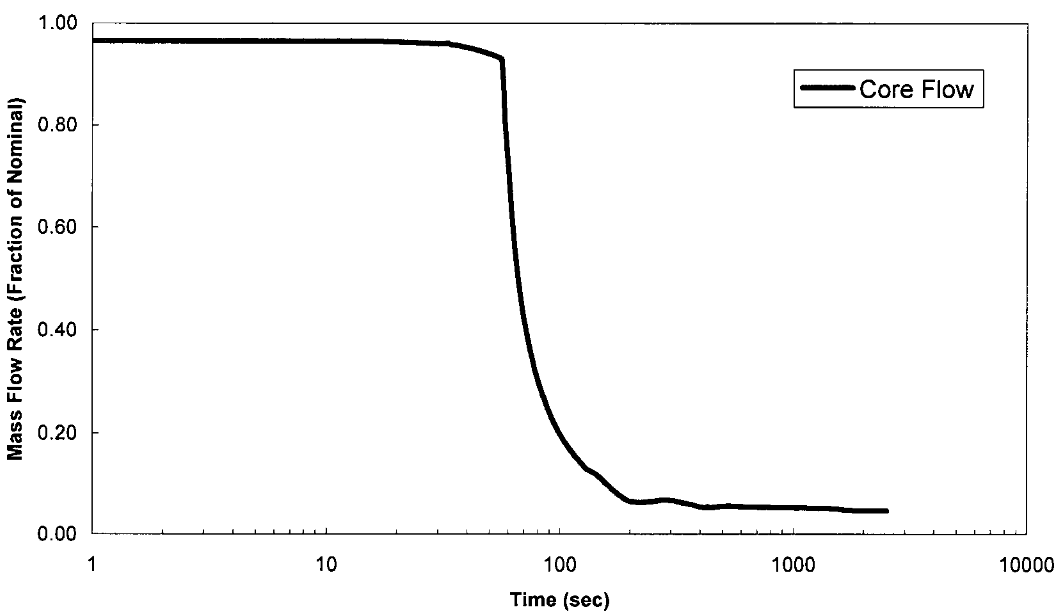


Figure 4.4-18: Core Flow (Case MFLB2 R3D)

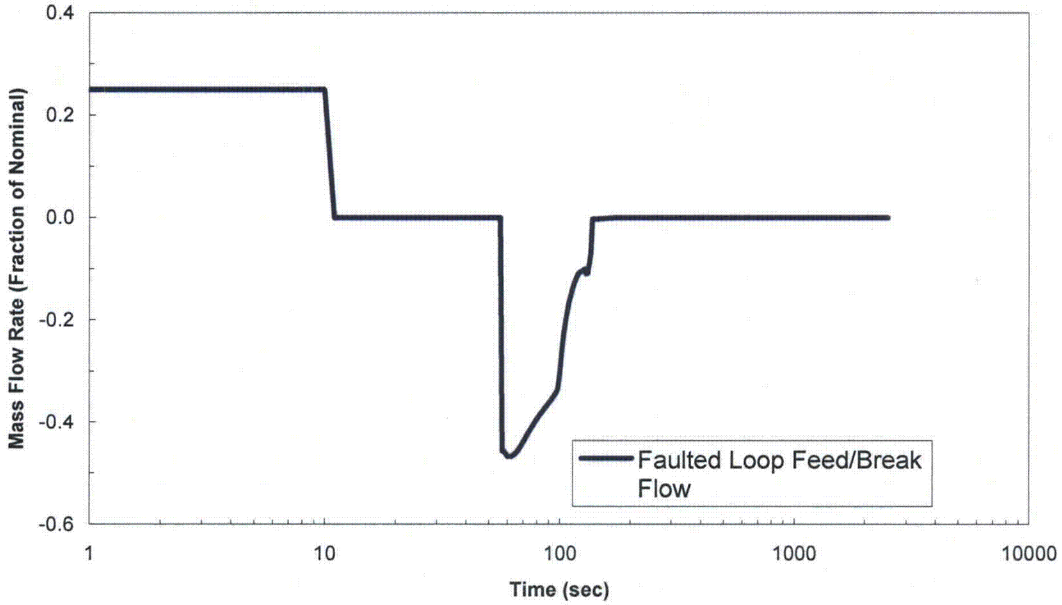


Figure 4.4-19: Faulted SG Feed/Break Flow (Case MFLB2 R3D)

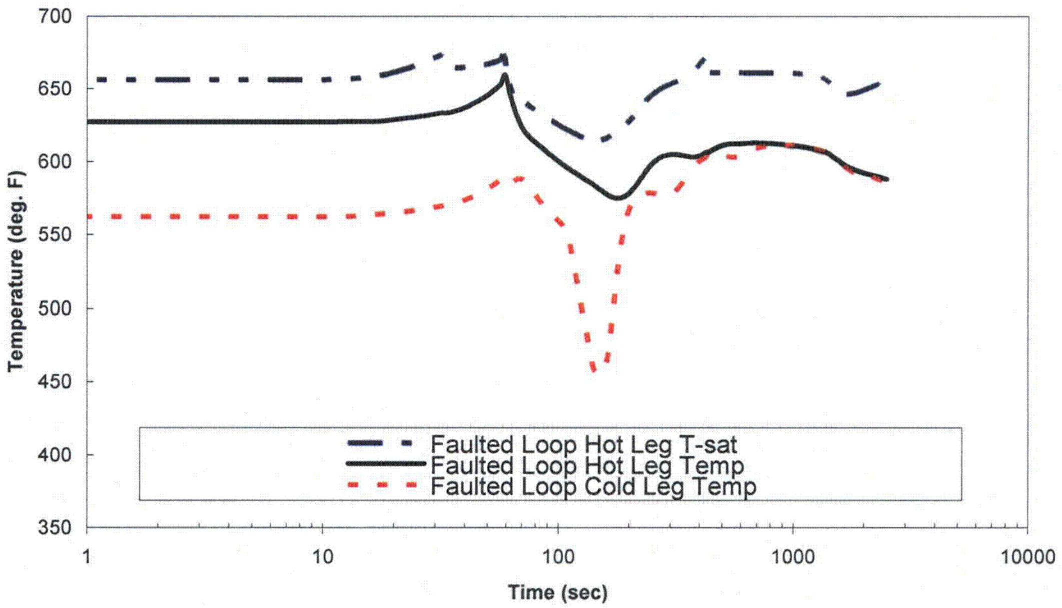


Figure 4.4-20: RCS Faulted Loop Temperatures (Case MFLB2 R3D)

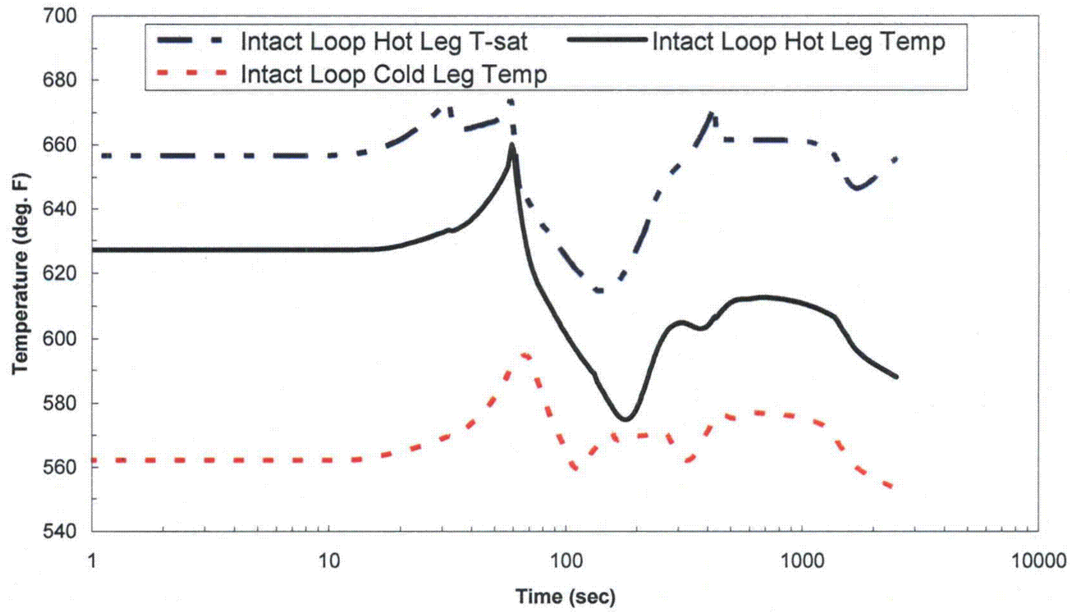


Figure 4.4-21: RCS Intact Loop Temperatures (Case MFLB2 R3D)

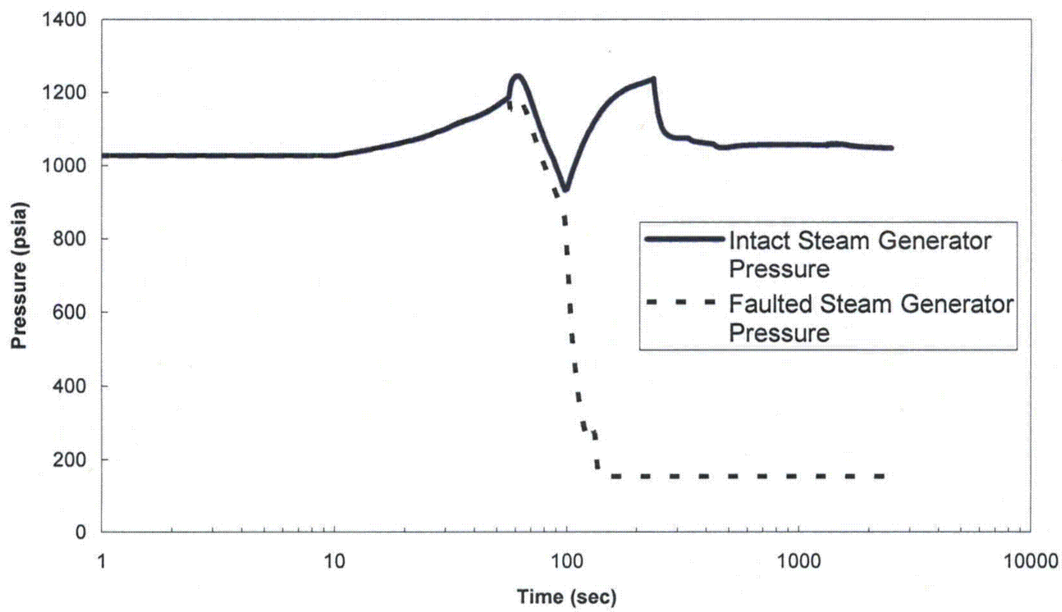


Figure 4.4-22: SG Pressures (Case MFLB2 R3D)

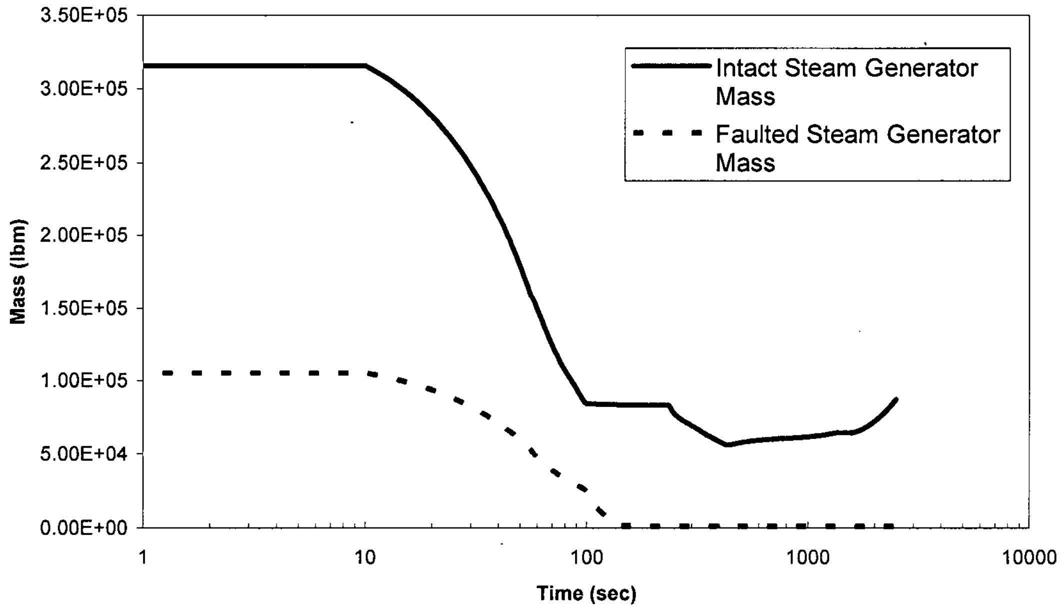


Figure 4.4-23: SG Mass (Case MFLB2 R3D)

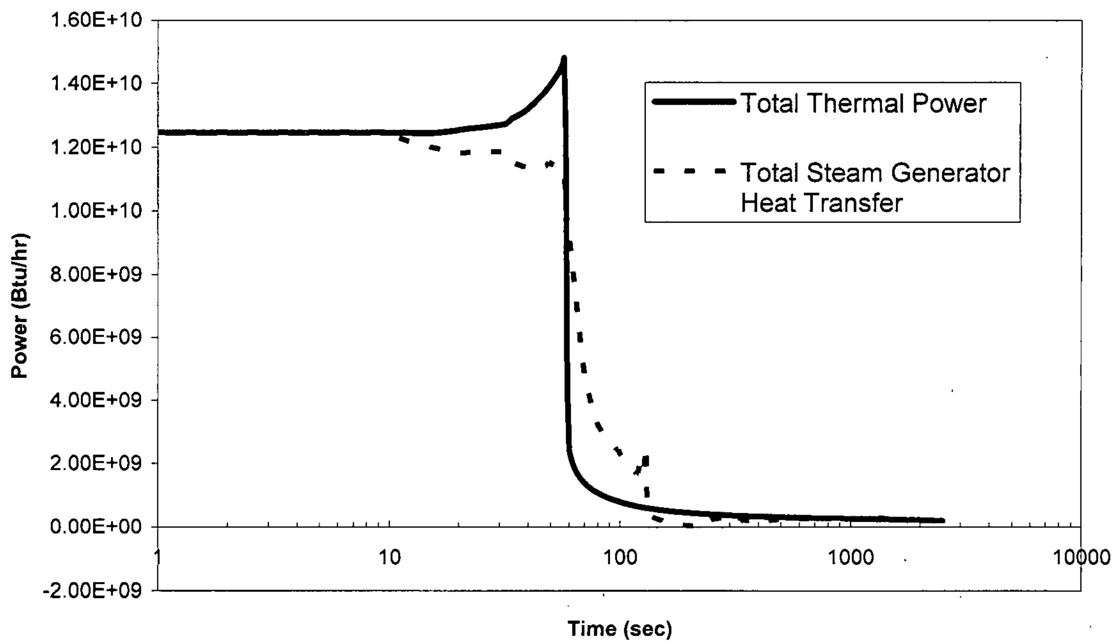


Figure 4.4-24: Energy Balance (Case MFLB2 R3D)

4.5 Steam Generator Tube Rupture (USAR Section 15.6.3)

Introduction

The steam generator tube rupture (SGTR) events have been re-analyzed to assess the impact of the change to the closure characteristics of the MSIVs and MFIVs, resulting from the MSIVs and MFIVs and associated actuator replacement, on the transient response as well as on the radiological consequences (see Section 4.11). In addition, the reanalysis incorporated the revisions to the assumed critical operator action times, reflect operator performance from simulator exercises, based upon operating procedures reflecting current communication protocols.

Transient Description

The accident examined the complete severance of a single SG tube, is considered an ANS Condition IV event, a limiting fault. The accident is assumed to take place at power, with the reactor coolant contaminated with fission products corresponding to continuous operation, due to a limited amount of defective fuel rods. The postulated accident results in an increase in the contamination of the secondary system, due to leakage of radioactive coolant from the RCS. As a coincident loss of offsite power is assumed at the time of reactor trip, discharge of activity to the atmosphere takes place via the SG atmospheric relief valves (ARVs) and/or the steam line safety valves.

Two SGTR scenarios are evaluated in order to ensure that operators can respond to the accident in a timely manner so as to minimize the resulting offsite releases and prevent overfilling of the effected steam line. Those scenarios are described below:

Overfill Scenario:

The overfill SGTR scenario assumes failures in the AFW System such that if operators do not respond quickly enough to terminate feedwater, then the affected steam line may fill with water. The specific failure involves the discharge flow control valve on the discharge side of the motor driven AFW pump feeding the faulted SG. To maximize flow to the faulted SG, it is assumed that this valve fails in the wide-open position. Failure of this valve coupled with the contribution from the turbine driven AFW pump has the potential for overfilling the faulted SG and subsequently, relieve water via a safety valve. The radioactive releases are maximized by assuming the safety valve is stuck-open following water relief, with an effective flow area equal to 5% of the total safety valve flow area.

Stuck-open ARV Scenario:

The stuck-open ARV scenario assumes that when the faulted SG's ARV is initially required to open, that it remains stuck open for twenty minutes. The basis for these twenty minutes includes consideration of the time the valve is stuck-open until the time that an operator manually closes the associated block valve. Failure of the faulted SG's ARV thus maximizes offsite release by assuming a direct path to the atmosphere.

Assumptions/Methods

Mass and energy balance calculations are performed using the RETRAN-3D computer code, operating in the RETRAN-02 mode, to quantify the transient response of both the primary and secondary system to the two SGTR scenarios analyzed. These calculations determine the primary-to-secondary mass release and the amount of steam vented from each of the SGs, from the occurrence of the tube rupture to the time at which the primary and secondary pressures are equalized and the break flow is stopped.

Operator actions in response to a SGTR are assumed to follow emergency operating procedure EMG E-3, "Steam Generator Tube Rupture." The responses of the operators to an SGTR can be considered to be a five-phase response: (1) identify that an SGTR has occurred, identify the faulted SG, and then isolate the faulted SG; (2) prepare to cool the RCS in order to maintain subcooling after subsequent depressurization; (3) cool and then depressurize the RCS to reduce the primary-to-secondary leakage; (4) terminate safety injection in order to prevent repressurization of the RCS; and (5) take the plant to cold shutdown conditions in order to establish cooling by the Residual Heat Removal System. Where applicable, operator response times are "hard-wired" into RETRAN. For example, the time to identify and isolate the faulted SG is manually input to the code, as this value represents a subjective time based on operator cognitive abilities and demonstration in Simulator exercises. Other system response times are calculated by RETRAN. For example, the cooldown is terminated based on subcooling margin which RETRAN can readily calculate. System transient response is explicitly calculated by RETRAN whereas operator response times are implicit input assumptions. Table 4.5-1 lists the operator response times assumed in the analyses.

Consistent with the current analyses presented in USAR, assumptions are conservatively made to increase the probability for faulted SG overfill and to maximize the radioactive releases to the atmosphere for the overfill scenario. Also, conservative assumptions are made to maximize the amount of offsite release for the stuck-open ARV scenario.

In estimating the mass transfer from the RCS through the broken tube, the following assumptions are made:

Overfill Scenario:

1. Reactor trip occurs automatically as a result of low pressurizer or overtemperature delta T (OTΔT). Loss of offsite power occurs at reactor trip.

2. AFW flow rate is allowed to vary with the fluctuation in the faulted SG pressure. This results in higher AFW flow when the faulted SG pressure decreases. Six minutes following the initiation of the SI signal, the AFW from the turbine driven AFW pump to the ruptured SG is terminated by closing the un-failed discharge flow control valve. Eighteen minutes following the initiation of the SI signal, the AFW from the motor driven AFW pump to the ruptured SG is terminated by locally closing the failed discharge flow control valve. AFW flow to the intact SGs maintains the narrow range level between 6% and 50% as indicated in the emergency operating procedure EMG E-3.

3. Cooldown of the RCS is initiated at 30 minutes following the initiation of the SI signal. It is assumed that steam is released through the remaining three OPERABLE SG ARVs in the intact loops until the RCS temperature of the core exit thermal couples corresponds to the ruptured SG pressure as listed in procedure EMG E-3. Note that the analysis assumes that the operators will continue to maintain the plant at that temperature for the duration of the transient, consistent with procedure EMG E-3. TS LCO 3.7.4 requires that four ARV lines shall be OPERABLE. Each ARV line consists of one ARV and an associated manual block valve. With one of the required ARV lines unavailable due to its association with the ruptured SG, the remaining three ARV lines are available to ensure that subcooling can be achieved for the RCS.

4. Following termination of the RCS cooldown, the RCS is depressurized by opening a pressurizer PORV to assure an adequate coolant inventory prior to terminating SI flow. Primary depressurization is initiated at 5 minutes following the termination of the RCS cooldown and continues until the RCS pressure is less than the ruptured SG pressure.

5. Following depressurization, termination of SI is delayed to ensure sufficient liquid enters the ruptured SG steam line to force the safety valve open and cause water relief. It is assumed that 5 minutes following termination of the RCS depressurization that the safety injection flow is reduced to just one centrifugal charging pump (CCP). At 15 minutes following the termination of the RCS depressurization, the one CCP is throttled back to 100 gpm and at 30 minutes following termination, letdown is initiated such that the net flow due to SI and letdown is zero.

Stuck-open ARV Scenario:

1. Reactor trip occurs automatically as a result of low pressurizer pressure or OTΔT. Loss of offsite power occurs at reactor trip.

2. As pressures rise on the secondary side, the SG ARVs open to release excess secondary pressure. Although the ARVs in the unaffected SG close within 7 minutes, the ARV for the faulted SG is assumed to remain open and steam release to continue for 20 minutes until the ARV block valve is manually closed.

3. AFW is initially delivered at a rate of 250 gpm to each SG. AFW is maintained to assure that narrow range level in each SG exceeds 15%.

4. Following ruptured SG isolation, cooldown is initiated when the narrow range in the ruptured SG is greater than 10% and its pressure exceeds 630 psia. Cooldown continues until RCS temperature is reduced to 50°F less than the ruptured SG saturation temperature.

5. RCS depressurization is initiated three minutes after completion of cooldown. This timing is consistent with observed simulator exercises.

6. After primary side depressurization is completed and SI termination criteria are met, a three minutes time delay is assumed prior to SI termination.

7. Following SI termination, the operators equalize pressure in the RCS and faulted SG in 5 minutes. During this time break flow in the faulted SG continues. After pressures are equalized, it is conservatively assumed that the transition to cold shutdown is made utilizing steam release to the atmosphere from the un-faulted SGs.

Results

Table 4.5-2 provides a time sequence of events for both SGTR scenarios. Parameters of primary and secondary systems are plotted as a function of time in Figure 4.5-1 through 4.5-20.

As in indicated on Figure 4.5-10, the faulted SG fills at approximately 1935 seconds. As a result, the overfilled SG water starts to enter the main steam line on the faulted loop and water/vapor mixture relief through the ruptured SG safety valve occurs at approximately 2692 seconds. Eventually, the 682 ft^3 steam line will be filled with water at approximately 3975 seconds. It should be noted that the steam lines have been statically analyzed to withstand all steam lines completely filled with water.

Conclusions

Analysis of the SGTR with postulated failure of the AFW discharge flow control valve indicated that overfilling the SG will occur during the transient. Subsequent accident dose analyses for both SGTR scenarios show that the radiological consequences resulting from a SGTR accident remain well within the limiting values specified in 10 CFR Part 100 and Standard Review Plan, Section 15.6.3. A SGTR will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming a simultaneous loss of offsite power.

Table 4.5-1 Assumed Operator Response Time

Action	Response Time in Minutes	
	Overfill	Stuck-Open ARV
Identify and Isolate faulted SG	T_{SI} + 18	T_{SI} +21
Initiate RCS Cooldown	T_{SI} + 30	T_{SI} + 51
Complete RCS Cooldown	T_{CCld}	T_{CCld}
Initiate RCS Depressurization	T_{CCld} + 8	T_{CCld} + 4.6
Complete RCS Depressurization	T_{CDep}	T_{CDep}
Terminate SI	T_{CDep} + 5	T_{CDep} + 5

Table 4.5-2 Time Sequence of Events for both SGTR scenarios

System Response/Operator Action	Time (seconds)	
	Overfill	Stuck-Open ARV
SGTR Occurs	0.0	0.0
Reactor Trip	154.5	143.0
AFW Injection	182.5	203.0
SI Signal	215.1	300.7
SI Enters RCS	230.1	315.7
Faulted SG Isolated	1295.1	1582.7
Initiate RCS Cooldown	2015.1	3376.7
Terminate RCS Cooldown	2398.6	3923.9
Initiate RCS Depressurization	2878.6	4259.9
Terminate RCS Depressurization	3041.4	4525.5
Terminate SI	3342.5	5113.8
Pressure Equalization	6225.6	~6000

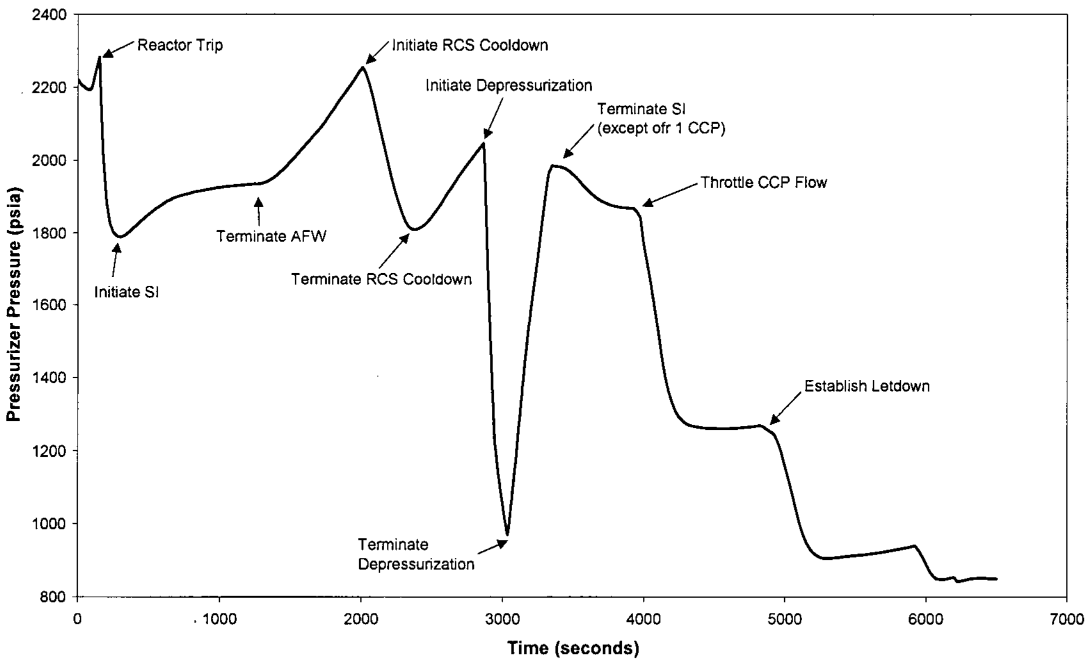


Figure 4.5-1 SGTR Overfill Analysis Pressurizer Pressure

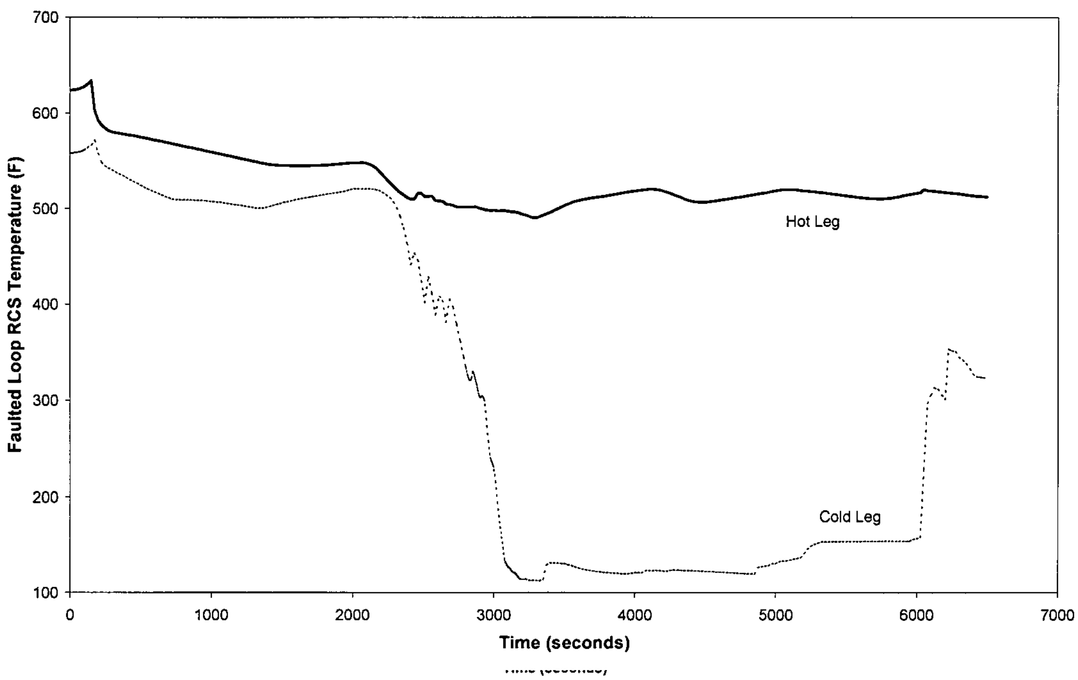


Figure 4.5-2 SGTR Overfill Analysis Faulted Loop RCS Temperatures

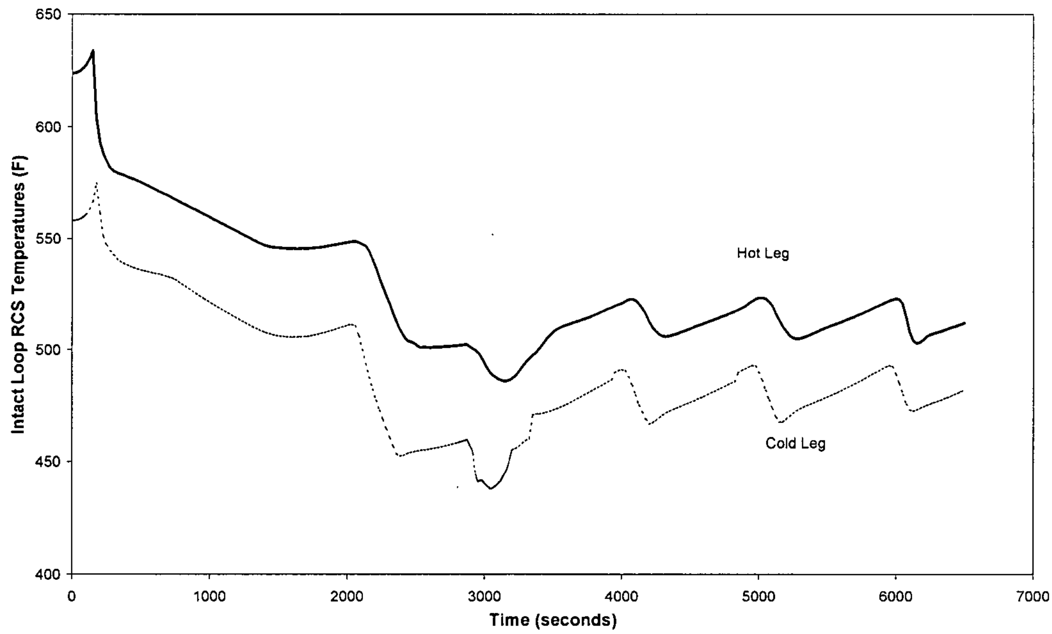


Figure 4.5-3 SGTR Overfill Analysis Intact Loop RCS Temperatures

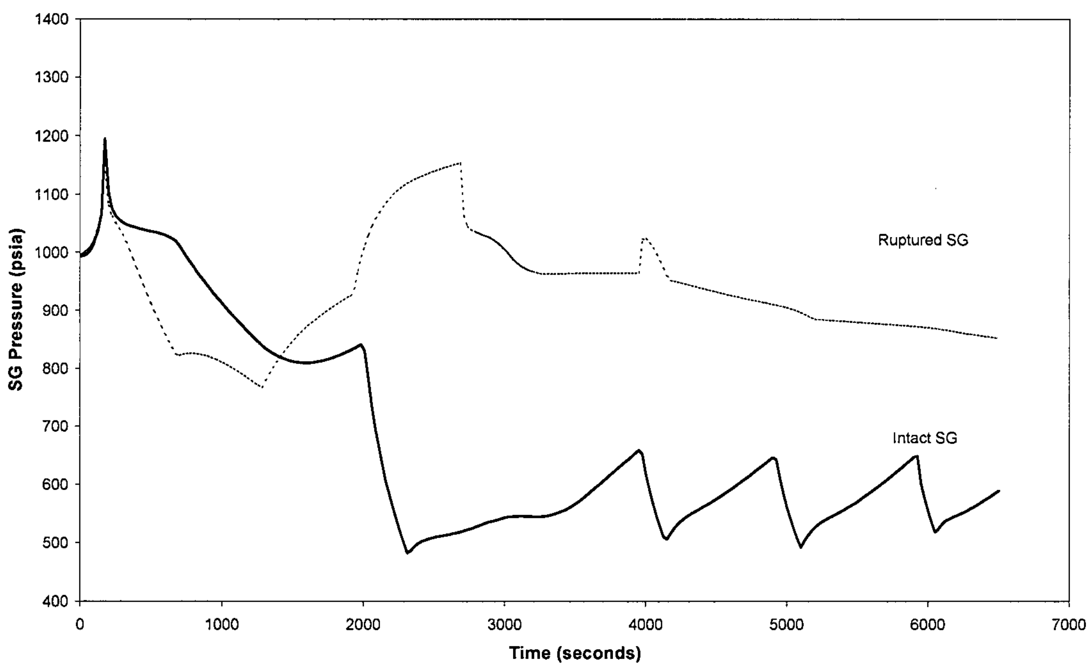


Figure 4.5-4 SGTR Overfill Analysis SG Pressures

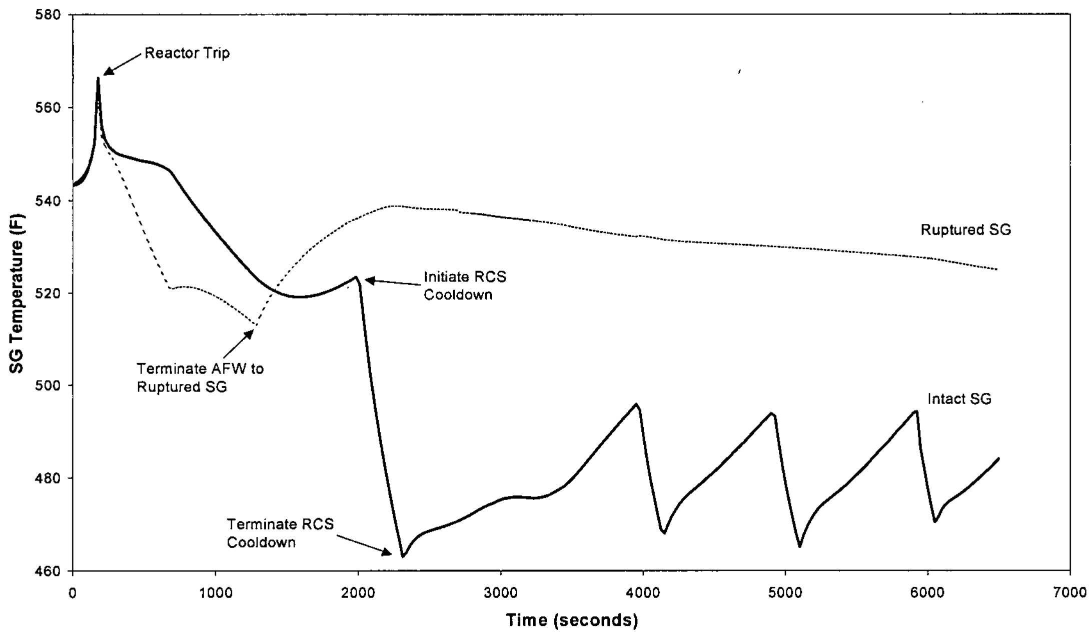


Figure 4.5-5 SGTR Overfill Analysis SG Temperatures

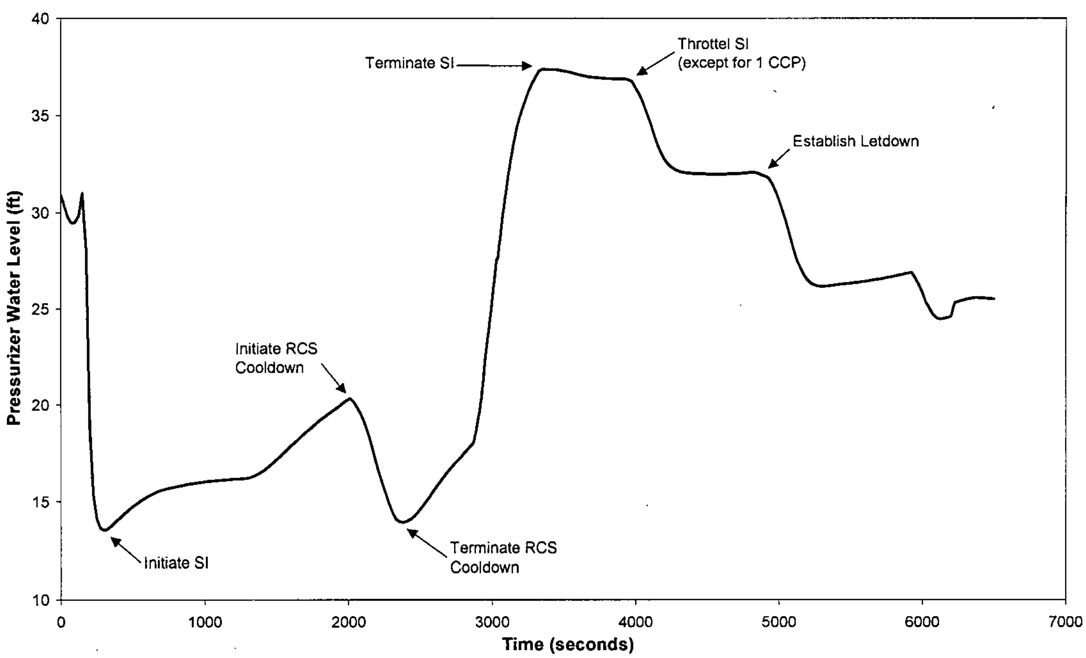


Figure 4.5-6 SGTR Overfill Analysis Pressurizer Water Level

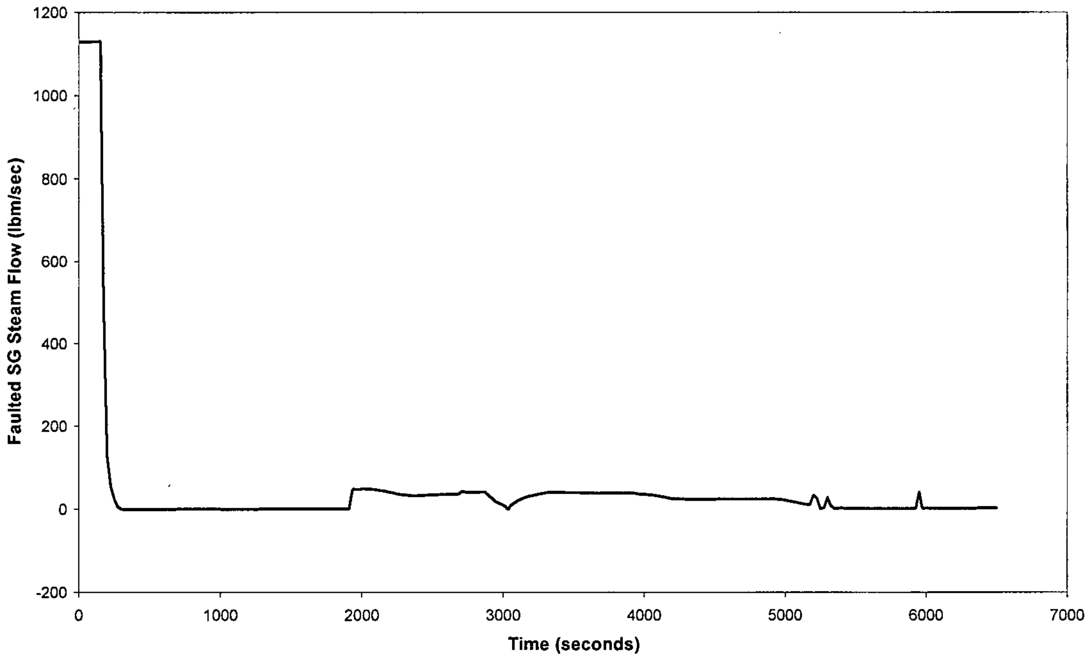


Figure 4.5-7 SGTR Overfill Analysis Faulted SG Steam Flow

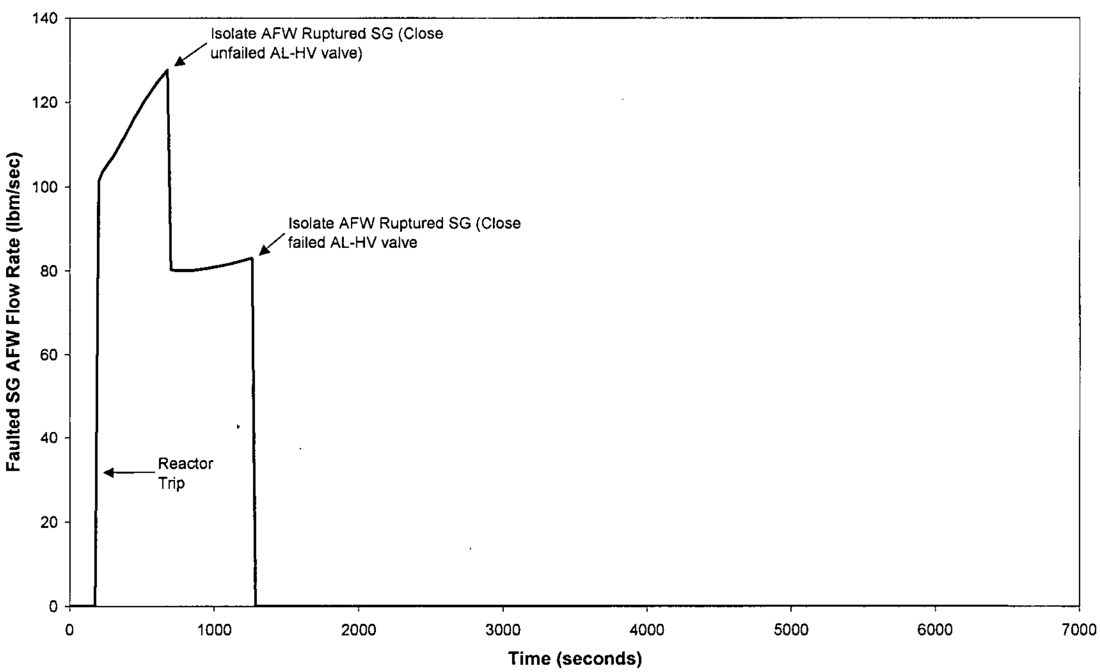


Figure 4.5-8 SGTR Overfill Analysis AFW Flowrate

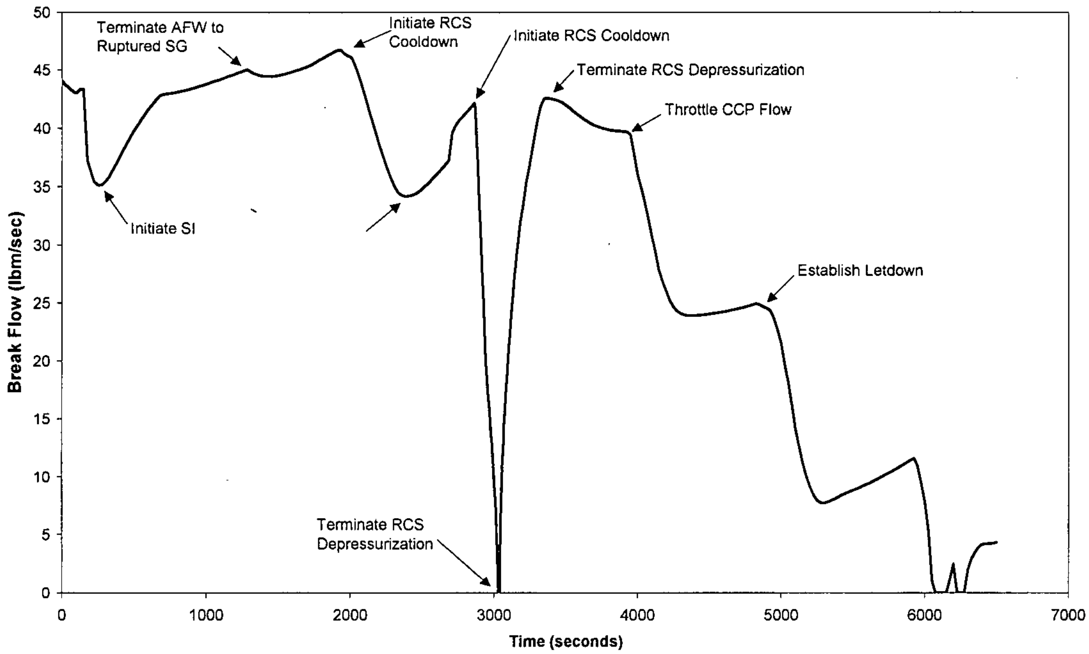


Figure 4.5-9 SGTR Overfill Analysis Break Flow

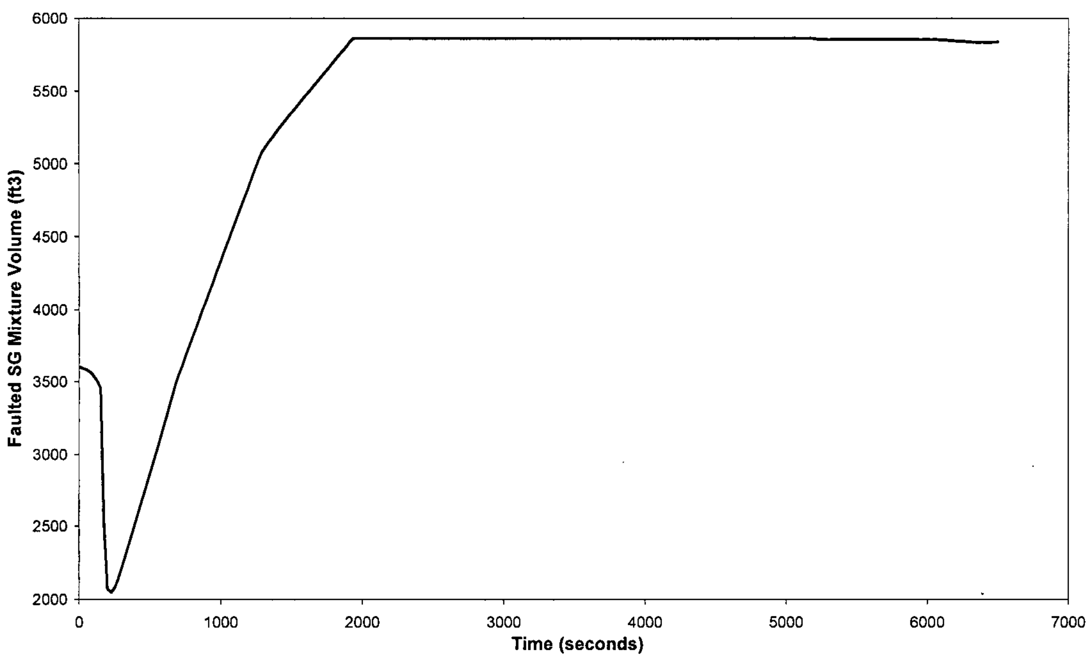


Figure 4.5-10 SGTR Overfill Analysis SG Mixture Volume

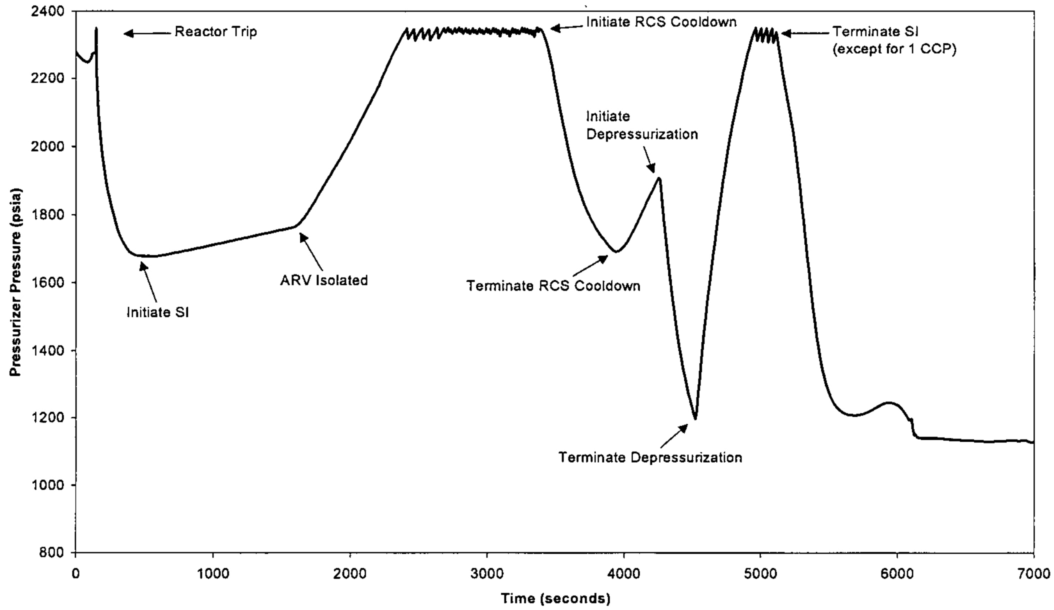


Figure 4.5-11 SGTR w/ Stuck-Open ARV Pressurizer Pressure

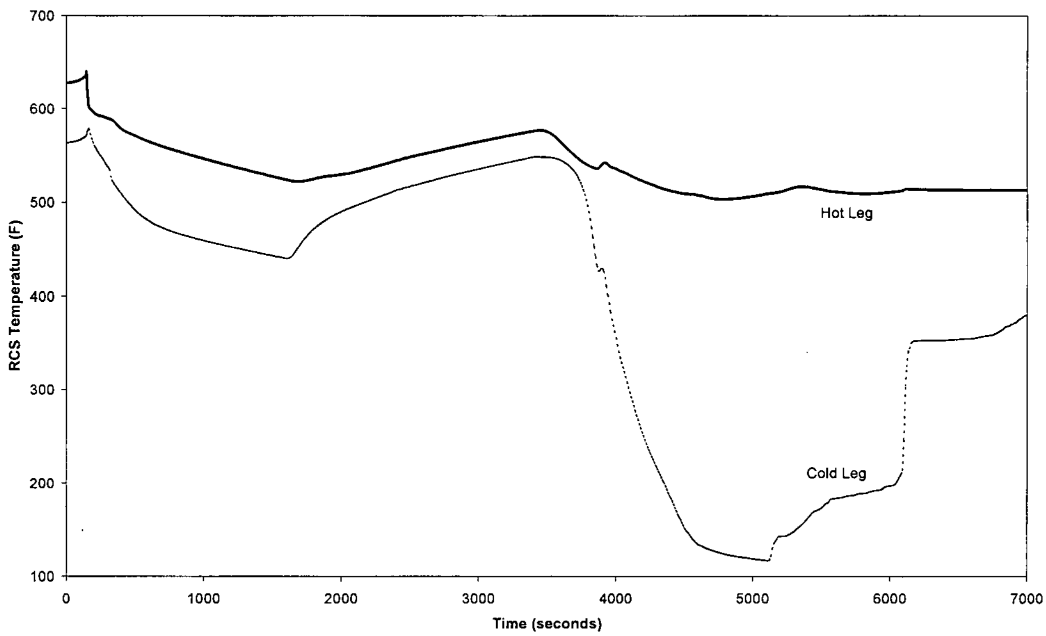


Figure 4.5-12 SGTR w/ Stuck-Open ARV Faulted Loop RCS Temperature

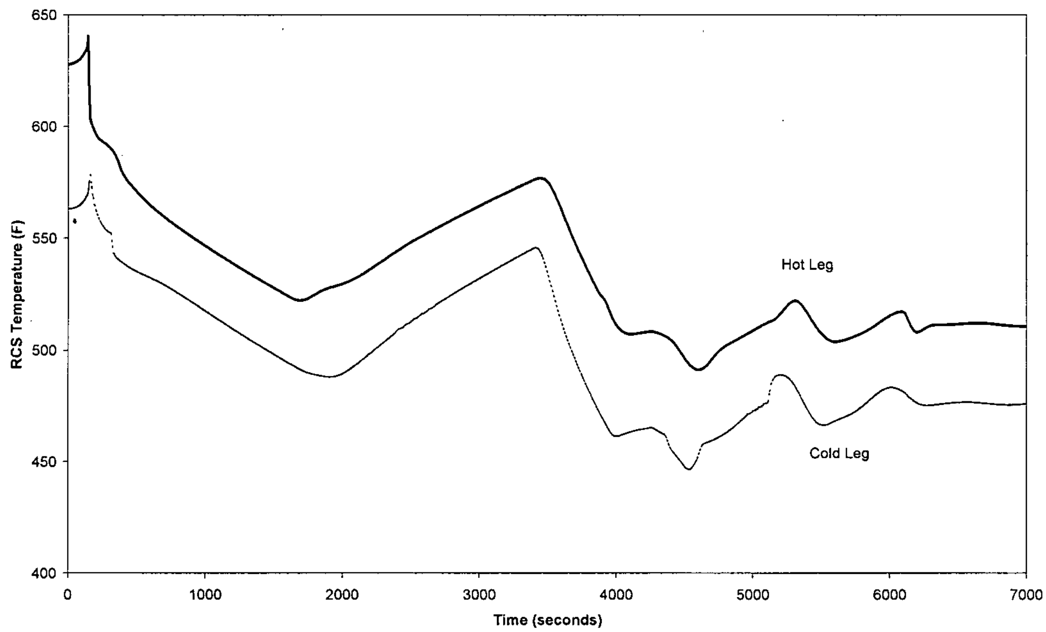


Figure 4.5-13 SGTR w/ Stuck-Open ARV Intact Loop RCS Temperature

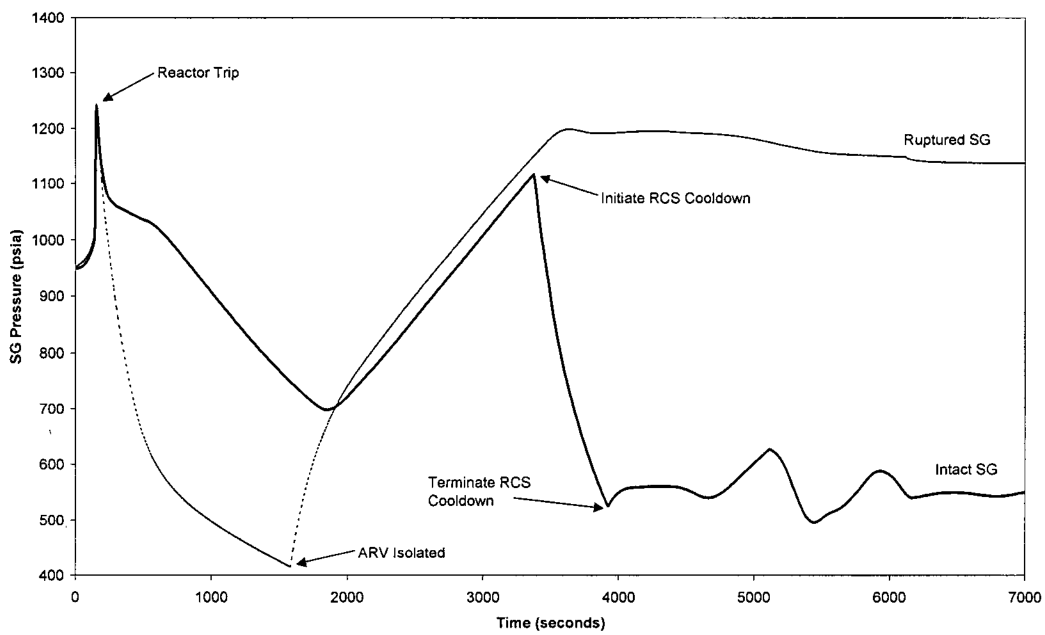


Figure 4.5-14 SGTR w/ Stuck-Open ARV SG Pressure

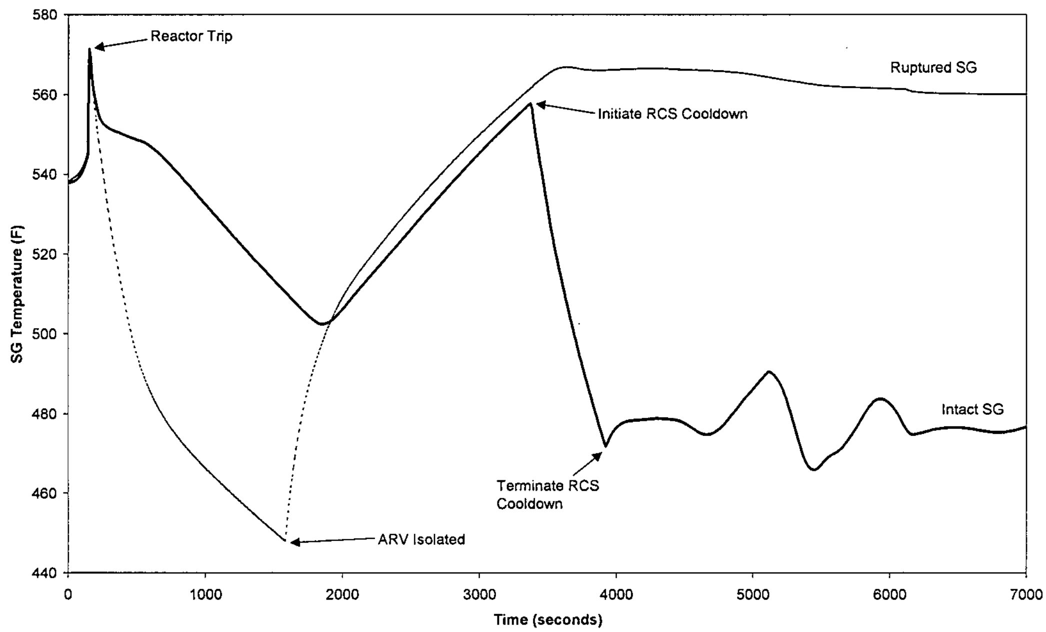


Figure 4.5-15 SGTR w/ Stuck-Open ARV SG Temperature

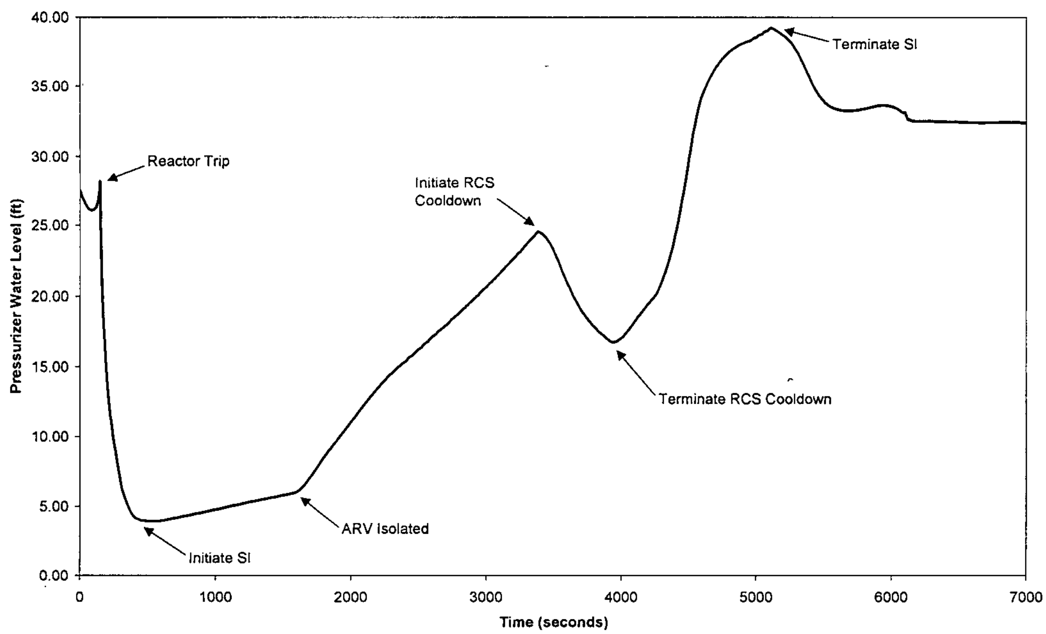


Figure 4.5-16 SGTR w/ Stuck-Open ARV Pressurizer Water Level

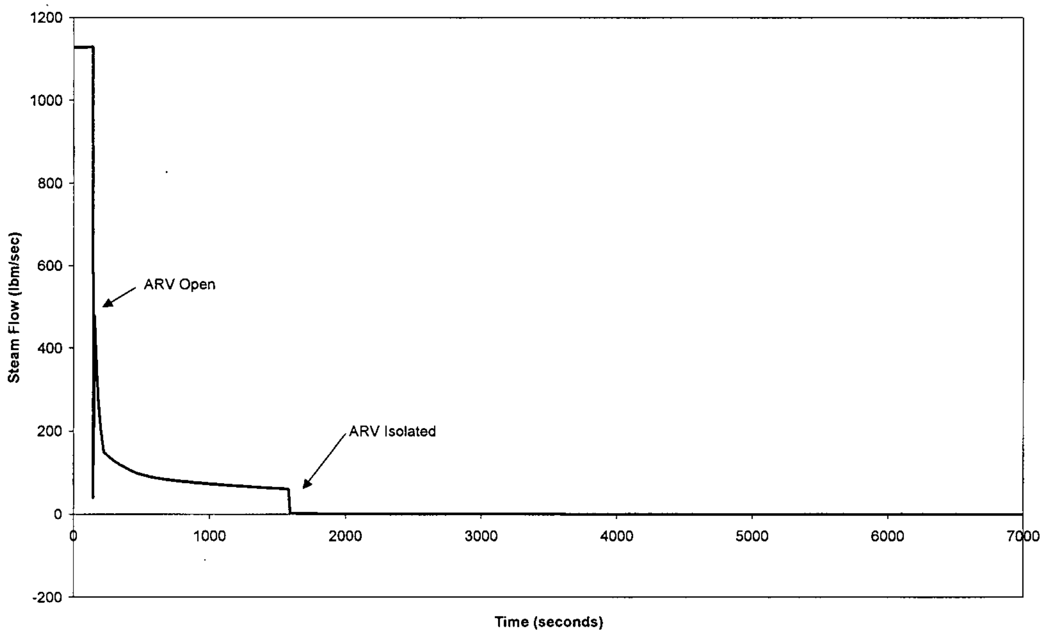


Figure 4.5-17 SGTR w/ Stuck-Open ARV Faulted SG Steam Flow

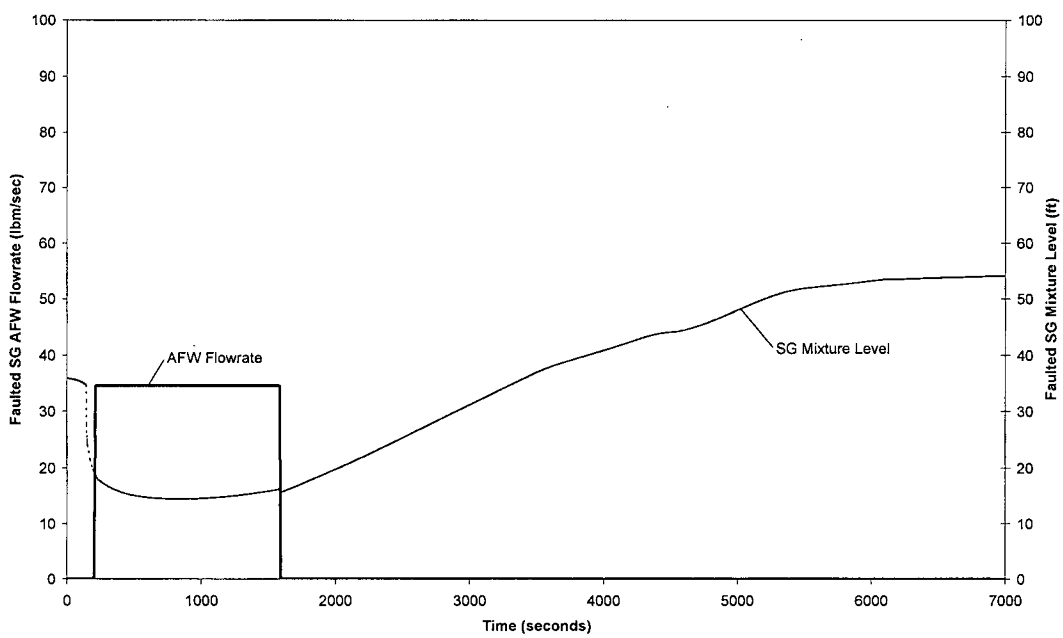


Figure 4.5-18 SGTR w/ Stuck-Open ARV Faulted SG AFW Flow and Mixture Level

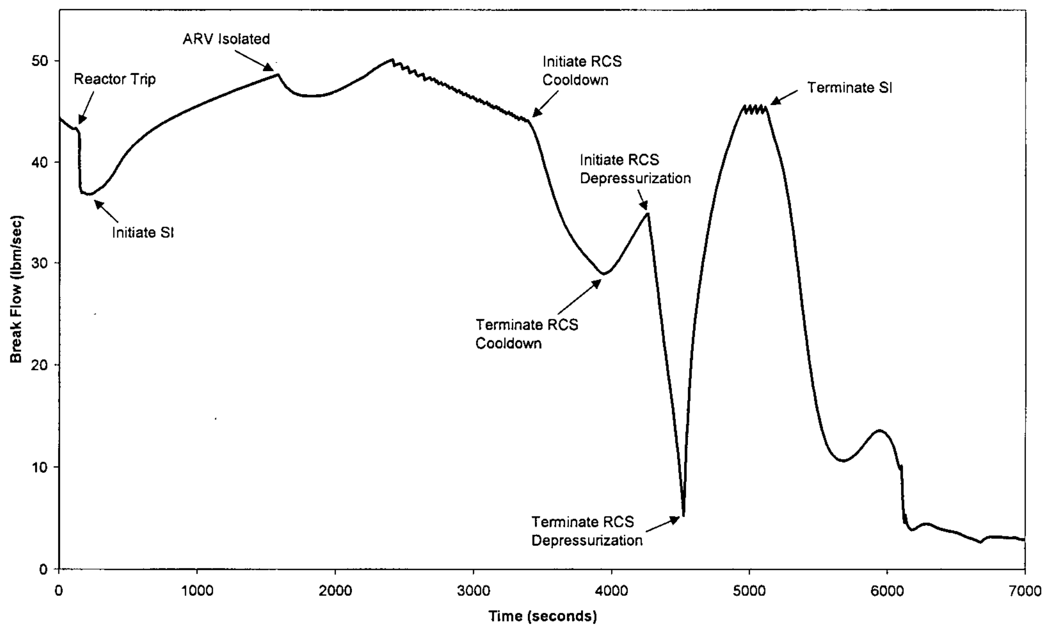


Figure 4.5-19 SGTR w/ Stuck-Open ARV Faulted SG Total Break Flow

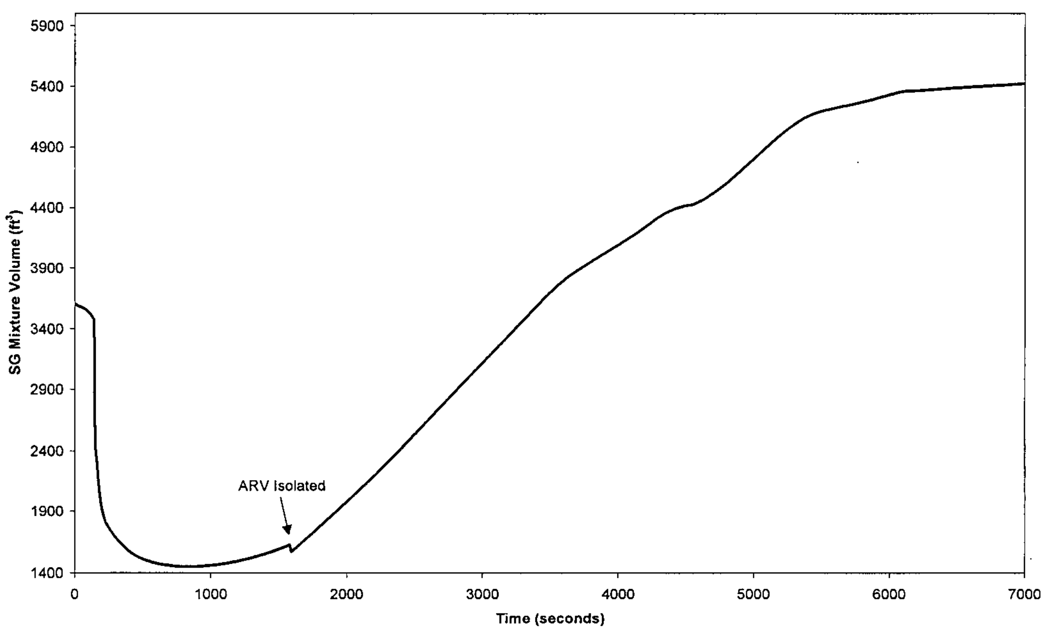


Figure 4.5-20 SGTR w/ Stuck-Open ARV Faulted SG Mixture Volume

4.6 Small Break Loss-of-Coolant-Accident (SBLOCA) (USAR Section 15.6.5)

Introduction

The small break Loss of Coolant Accident (SBLOCA) analysis has been performed using the 1985 Westinghouse SBLOCA Evaluation Model (EM) with NOTRUMP (References 6, 9, and 10). The analysis is performed in support of the MSIV and MFIV and associated actuator replacement to demonstrate conformance with the 10 CFR 50.46 requirements with consideration of the age (1992) of the SBLOCA analysis of record (AOR) and also the number of peak cladding temperature (PCT) assessments currently tracked on the SBLOCA PCT summary sheet.

NOTE: Several updates have been made to the NOTRUMP-EM since the previous analysis of record was completed, including the use of the COSI condensation model and SI in the broken loop (Reference 10), which have been incorporated in this analysis.

Assumptions, Input Parameters and Analysis Methodology

The SBLOCA methodology using, NOTRUMP-EM, was developed in accordance with the requirements of 10 CFR 50, Appendix K. Components of the NOTRUMP-EM methodology include, but are not limited to, 102% of full power, all peaking factors simultaneously at their most limiting values, the Baker-Just zirconium water reaction model, the decay heat model (1971 ANS Infinite +20%), and the Moody break flow during periods when two phase flow is calculated to occur at the break.

For SBLOCA analysis, the most limiting single active failure is that which results in the minimum ECCS flow delivered to the RCS. This single failure has been determined to be a loss of offsite power and failure of a diesel generator, resulting in the loss of an emergency power train which causes the loss of one complete train of ECCS components. This means that credit can be taken for only one high head charging pump, one safety injection pump, and one residual heat removal (low head) pump. Therefore, during the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (one for each loop) with one branch injection line either spilling to the RCS pressure or to the containment backpressure of 0 psig, depending on the break size. To minimize delivery to the core, the branch line with the least resistance is modeled as the spilling line.

All SI flows were assumed to inject into the RCS for the 2, 3, 4, and 6 inch breaks (i.e. the broken loop spills into the RCS backpressure). For the 8.75 inch break in the accumulator/SI line, the broken loop SI flow was assumed to spill to containment pressure. This assumption is conservative in that high head (charging) SI is assumed to spill to the containment, even though it would actually inject directly into the cold leg and would not be affected by an accumulator line break.

The NOTRUMP Evaluation Model includes the following computer codes:

NOTRUMP: Thermal-hydraulic response of RCS during transient.
SBLOCTA: Fuel rod / cladding heat-up during transient.

The key input parameters for the SBLOCA analysis are summarized in Tables 4.6-1 through 4.6-4.

Description of SBLOCA Analysis

The methodology employed consists of first determining system thermal hydraulic response to the SBLOCA event using the NOTRUMP code (References 6 and 9). These results are then analyzed as to their effect on the hot rod heat up using the SBLOCTA code (Reference 7), to demonstrate that the PCT, maximum cladding oxidation, and maximum hydrogen generation are below their limiting values as defined by 10 CFR 50.46 (Reference 8).

Prior to the break occurring, the RCS is assumed to be at normal full power, steady state operating conditions. Following the break, loss of offsite power is assumed to occur coincident with the reactor trip signal, which initiates control rod insertion, turbine trip and main steam isolation. SI is assumed to initiate when the low pressurizer pressure safety injection setpoint is reached as the RCS continues to depressurize. The SI signal initiates pumped ECCS actuation and main feedwater isolation. For small break sizes which have sufficient break flow to reduce the system pressure below the accumulator gas cover pressure, accumulator injection is modeled.

During a SBLOCA transient, the loss of primary inventory through the break causes a gradual depressurization of the RCS to a pressure slightly above the minimum lift pressure of the main steam safety valves (MSSVs). During the early part of the transient, the pump suction pipe U-bend will remain filled with liquid, sealing off steam flow to the break; therefore, at this time, the break flow is entirely liquid, which in conjunction with the low SI flow rates associated with the high system pressure, results in a net reduction in primary system mass. Due to hydrostatic balance with the loop seal, there is a core level depression which may lead to core uncovery, resulting in temporary clad heat up.

Note: It is not until the loop seal clears that steam generated by the core decay heat can vent through the break and increase the RCS depressurization rate.

The SGs and the break together provide the principal heat removal mechanism, until the steam generation in the core is sufficient to establish a flow path through the low elevation loop seal (loop seal clearing) and out of the break. This results in two-phase and ultimately all steam flow through the break, which then becomes the principal heat removal mechanism. The loss of system mass and the core level reduction produced by the presence of the steam bubble in the upper elevations of the reactor vessel, hot leg and SG tubes may lead to core uncovery.

For break sizes causing core uncovery, the vapor temperature at the top of the core increases corresponding to the mixture level decrease below the top of the core. The PCT occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam only. This time is characterized by the highest vapor superheating above the mixture level. The decrease in system pressure results in an increase in the SI flow. If the system pressure is below the accumulator gas cover pressure, the accumulators begin to inject, the mixture level in the core begins to increase, and the top of core vapor temperature and, correspondingly, the PCT begin to decrease.

The small break transient is considered to be terminated when the SI flow rate exceeds the break flow rate and the cladding temperature transient has turned around.

<u>Acceptance Criteria</u>

The criteria for acceptability for the SBLOCA analysis are specified in 10 CFR 50.46(b) (Reference 8) as follows:

1. PCT. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

2. Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

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

Criteria 1 – 3 are addressed explicitly by virtue of the models within NOTRUMP and SBLOCTA. Criteria 4 and 5 are addressed implicitly by virtue of the blockage model imposed when SBLOCTA predicts rod burst and NOTRUMP predicting switchover from injection to recirculation.

<u>Results</u>

The analysis performed is comprised of a break spectrum of 2, 3, 4, 6 and 8.75 inches, to determine the most limiting break size with respect to the PCT as well as the maximum transient oxidation. The NOTRUMP calculations showed no core uncovery for the 2 inch break and minimal core uncovery for the 8.75 inch break. Based on these NOTRUMP results it was determined that SBLOCTA calculations were not necessary for the 2 and 8.75 inch break cases. SBLOCTA calculations were performed for the 3, 4 and 6 inch break cases, with an additional case considering axial annular blanket pellets, being performed for the limiting 4 inch break.

The results for the break spectrum analyzed are presented in Tables 4.6-5 and 4.6-6. The results (Table 4.6-6) indicate that the limiting PCT, calculated for the 4 inch break case, is 936°F, with a maximum transient oxidation of 0.01%.

For the limiting 4inch break, the transient response results are presented in Figures 4.6-1 through 4.6-14.

The primary side pressure begins a rapid drop at the time of break initiation (Figure 4.6-1). A reactor trip signal is generated at 22.75 seconds followed by a SI signal at 30.40 seconds. This primary side depressurization reaches a quasi-equilibrium condition at approximately 1240 psia (see Figures 4.6-9 and 4.9-10), slightly above the MSSV lift pressure. The core mixture level begins to decrease (Figure 4.6-2) until it reaches the top of the hot leg elevation. The SGs and the break provide the principal heat removal mechanism until the steam generation in the core is sufficient to establish a flow path through the low elevation loop seal at approximately 248 seconds (loop seal clearing) and out of the break. This results in two-phase and ultimately all steam flow through the break which then becomes the principal heat removal mechanism (Figure 4.6-8). The rate of core level draining is then slowed as vapor is now allowed to enter the hot legs due to the loop seal clearing. When the core mixture level drops below the bottom of the hot legs, the rate of uncovery once again establishes itself. The RCS continues to depressurize and the core level continues to decrease until the top of the core uncovers at 713 seconds leading to an increase in the core exit vapor temperature (Figure 4.6-5) and the start of clad heat up (Figure 4.6-4). The PCT occurs near the time when the core is most deeply uncovered and the top of the core is being cooled by steam only. As illustrated in Figure 4.6-13, the SI flow rate continues to increase as the RCS pressure decreases. The accumulators begin to inject at 954 seconds (Table 4.6-5) when the RCS pressure reaches the accumulator setpoint (including uncertainties) of 583 psia (Figure 4.6-7). The SI replenishes the core level, which results in a reversal in the clad heat up transient. The PCT of 936°F occurs at 1033.5 seconds (Figure 4.6-4) followed by a steady increase in the core mixture level (Figure 4.6-2). The small break transient is considered to be terminated when the SI flow rate exceeds the break flow rate and the cladding temperature transient has reversed, as can be seen from the Figures 4.6-1 through 4.6-14:

- The core has recovered.
- The core exit vapor temperature has reached a maximum and is below 500°F at the end of the transient.
- SI flow and break flow have come to equilibrium.
- The RCS mass is increasing.
- The RCS pressure has leveled off.

As such, the transient can be considered terminated.

<u>Conclusions</u>

The results of the analysis show that the acceptance criteria discussed in the preceding section for the SBLOCA have been met. Annular Pellets have a negligible impact on results and as such the limiting PCT will be reported as 936 °F (Table 4.6-6), which occurs for the 4 inch equivalent diameter cold leg break. Local oxidation of the cladding is less than 17%, and the core-wide oxidation is less than 1.0%. All five acceptance criteria have been met as addressed in the SBLOCA analysis: (1) PCT, maximum cladding oxidation, and maximum hydrogen generation are addressed explicitly by virtue of the models within NOTRUMP and SBLOCTA, (2) maintaining a coolable geometry and long-term cooling are addressed implicitly by virtue of the blockage model imposed when SBLOCTA predicts rod burst and NOTRUMP predicting switchover from injection to recirculation.

Table 4.6-1 Input Parameters

A. Core Parameters	
100% Licensed Core Power	3565 MWt
Fuel Type	17x17 Standard RFA-2 Zirlo+2
Total Core Peaking Factor, F_Q	2.50
Hot Rod Enthalpy Rise Peaking Factor, $F_{\Delta H}$	1.65
Hot Assembly Peaking Factor, $\overline{P}_{HA}$	1.469
Axial Offset	+ 13%
K(z) limit	2 line segment
Core Power Calorimetric Uncertainty	2%
B. Reactor Coolant System	
Thermal design flow	90,324 gpm/loop
Total Core Bypass Flow	8.4%
Nominal vessel average temperature range	570.7 - 588.4°F
Pressurizer pressure	2250 psia
Pressurizer pressure uncertainty	50 psi
Reactor coolant pump type	Model 93A-1 7000HP
Reactor coolant pump weir height	0.4167 ft
C. Reactor Protection System	
Reactor Trip Setpoint	1805 psig
Reactor Trip Signal Processing Time (includes Rod Drop Time)	4.7 seconds
D. AFW System	
Maximum AFW temperature	124 °F
Minimum AFW flow rate	210 gpm/SG
Initiation Signal	Reactor Trip
AFW delivery delay time	60 seconds
Purge Volume	118.4 ft^3/loop
E. SGs	
SG tube plugging	10 %
MFW isolation signal	SI Signal
MFW isolation delay time	2.0 seconds
MFW flow coastdown time	15.0 seconds
Feedwater temperature	448 °F
F. SI	
Limiting single failure	1 EDG
SI water temperature	100 °F
Low-low pressurizer pressure signal	1700 psig
SI delay time	39 seconds
G. Accumulators	
Maximum initial temperature	120 °F
Initial water volume	850 gal
Minimum cover gas pressure (including uncertainties)	568 psig
H. RWST Draindown Input	
Maximum Containment Spray Flow	3395 gpm/train
Minimum Usable RWST Volume	239,324 gal
Maximum SI Water Temperature After Switchover to Cold Leg Recirculation Signal is Generated	212 °F

Table 4.6-2 Steam Generator Safety Valve Flows Per Steam Generator

MSSV	Set Pres (psig)	Uncert. %	Accum. %	Rated Flow at Full Open Pressure (lbm/hr)
1	1185	3.0	3	893,160
2	1197	3.0	3	902,096
3	1210	3.0	3	911,779
4	1222	3.0	3	920,715
5	1234	3.0	3	929,652

Table 4.6-3 Safety Injection Flows (Spilling to RCS Pressure)

RCS Pressure (psig)	CCP*		RCS Pressure (psig)	SI Pump*		RCS Pressure (psig)	RHR Pump*	
	Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop
0	317.48	112.59	0	444.82	154.97	0	2834.19	1075.76
100	309.44	109.74	100	429.33	149.57	20	2660.73	1009.97
200	301.32	106.87	200	413.26	143.97	40	2477.16	940.33
300	293.09	103.95	300	396.55	138.15	60	2280.95	865.91
400	284.76	101.00	400	379.11	132.07	80	2068.49	785.32
500	276.27	97.99	500	360.82	125.69	100	1834.31	696.50
600	267.61	94.92	600	341.54	118.98	120	1569.20	595.91
700	258.77	91.79	700	321.09	111.85	140	1254.38	476.48
800	249.72	88.58	800	299.20	104.22	150	1065.81	404.92
900	240.41	85.29	900	275.52	95.97	155	959.33	364.53
1000	230.86	81.90	1000	249.49	86.90	160	839.61	319.10
1100	221.00	78.41	1100	220.21	76.69	170	534.36	203.24
1200	210.79	74.79	1200	185.98	64.77	175	295.67	112.61
1300	200.18	71.03	1300	142.84	49.74	180	0	0
1400	189.13	67.12	1400	72.76	25.32			
1500	177.55	63.01	1420	40.00	13.91			
1600	165.35	58.69	1440	0	0			
1800	138.49	49.17	1500	0	0			
2000	106.72	37.91						
2235	55.68	19.82						
2300	34.70	12.38						
2350	11.76	4.22						
2400	0	0						

* One train operating

Table 4.6-4 Safety Injection Flows (Spilling to 0 psig/Containment Pressure)

RCS Pressure (psig)	CCP*		RCS Pressure (psig)	SI Pump*		RCS Pressure (psig)	RHR Pump*	
	Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop		Injecting (gpm) Intact Loops	Spilling (gpm) Broken Loop
0	317.48	112.59	0	444.82	154.97	0	2834.19	1075.76
100	301.07	122.40	100	424.60	159.61	20	2579.31	1163.27
200	284.72	131.75	200	403.66	164.22	40	2310.43	1248.78
300	268.39	140.72	300	381.93	168.80	60	2023.36	1333.04
400	252.01	149.36	400	359.28	173.35	80	1712.00	1416.88
500	235.50	157.73	500	335.54	177.90	100	1366.47	1501.38
600	218.82	165.85	600	310.53	182.44	120	967.94	1588.36
700	201.92	173.76	700	283.99	187.00	140	464.71	1683.57
800	184.73	181.49	800	255.56	191.59	150	186.62	1726.84
900	167.18	189.06	900	224.71	196.24	155	7.15	1751.62
1000	149.19	196.49	1000	190.63	200.98	160	0	1752.56
1100	130.67	203.80	1100	151.88	205.88	170	0	1752.56
1200	111.45	211.03	1200	105.54	211.07	175	0	1752.56
1300	91.40	218.18	1300	43.25	216.89	180	0	1752.56
1400	70.27	225.28	1380	0	220.15			
1500	47.68	232.37	1400	0	220.15			
1600	23.08	239.49						
1650	9.73	243.09						
1700	0	246.13						
1800	0	250.38						
2000	0	251.94						

* One train operating

Table 4.6-5 NOTRUMP Transient Results

Event (sec)	2-inch(3)	3-inch	4-inch	6-inch	8.75-inch(3)
Transient Initiated	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	112.02	40.29	22.75	9.71	7.16
SI Signal	121.62	48.85	30.40	15.08	8.77
SI Begins(1)	160.62	87.85	69.40	54.08	47.77
Loop Seal Clearing Occurs(2)	1166	418	248	55	17
Top of Core Uncovered	N/A	875	713	422	N/A
Accumulator Injection Begins	N/A	N/A	954	385	175
Top of Core Recovered	N/A	2233	1401	470	N/A
RWST Low Level(4)	1980	1943	1908	1793	1547

(1) SI begins 39.0 seconds (SI delay time) after the SI signal is reached.
(2) Loop seal clearing is considered to occur when the broken loop loop seal vapor flow rate is sustained above 1 lbm/s.
(3) There is no core uncovery for the 2 inch case, and only minimal core uncovery for the 8.75 inch case.
(4) The analysis assumes minimum usable RWST volume (239,324 gal) before the low-1 RWST water level signal for switchover to cold leg recirculation is reached.

Table 4.6-6 Beginning of Life (BOL) Rod Heatup Results

Result	2-inch(1)	3-inch	4-inch	4-inch w/ Annular Pellets	6-inch	8.75-inch(3)
PCT, °F	N/A	895.8	935.5	935.6	621.7	N/A
PCT Time, sec		1236.6	1033.5	1033.5	462.6	
PCT Elevation, ft		10.75	11.0	11.0	11.5	
Burst Time(2), sec		N/A	N/A	N/A	N/A	
Burst Elevation(2), ft		N/A	N/A	N/A	N/A	
Maximum ZrO_2, %		0.01	0.01	0.01	0.01	
Maximum ZrO_2 Elevation, ft		11.00	11.00	11.00	11.50	
Average ZrO_2, %		0.01	0.01	0.01	0.01	

(1) There is no core uncovery for the 2 inch case.

(2) None of the SBLOCTA calculations exhibited rod burst (hot rod or hot assembly average rod).

(3) The core only uncovers for a minimal time and does not warrant SBLOCTA analysis.

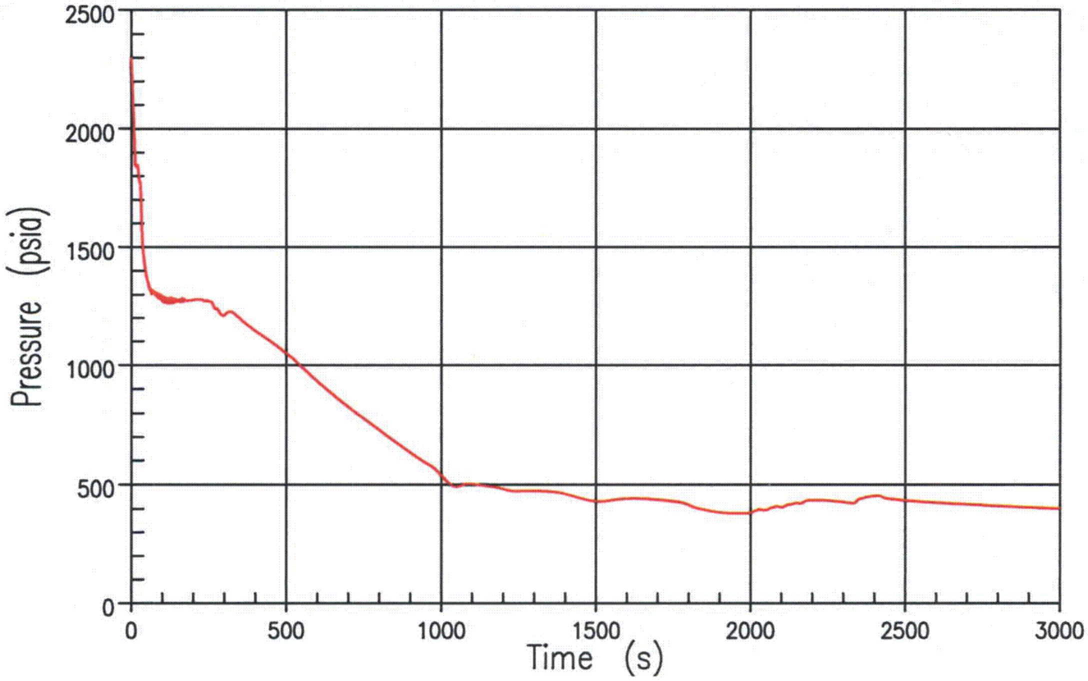


Figure 4.6-1 Pressurizer Pressure

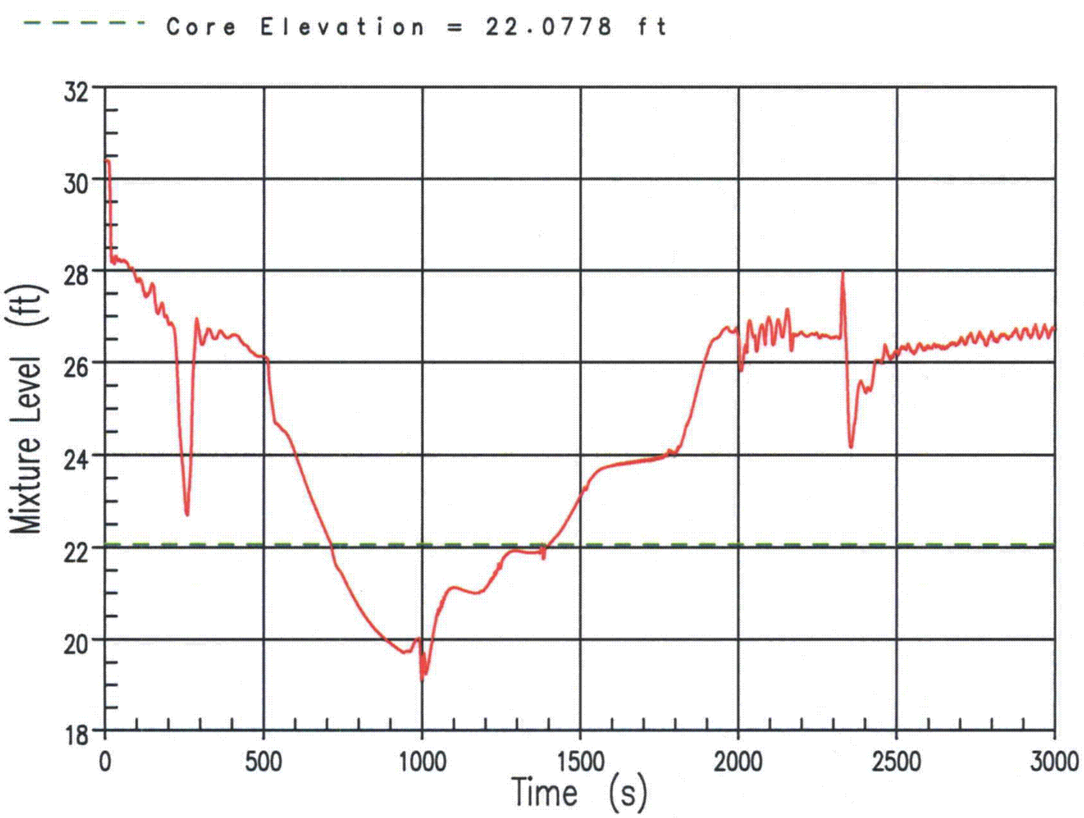


Figure 4.6-2 Core Mixture Level

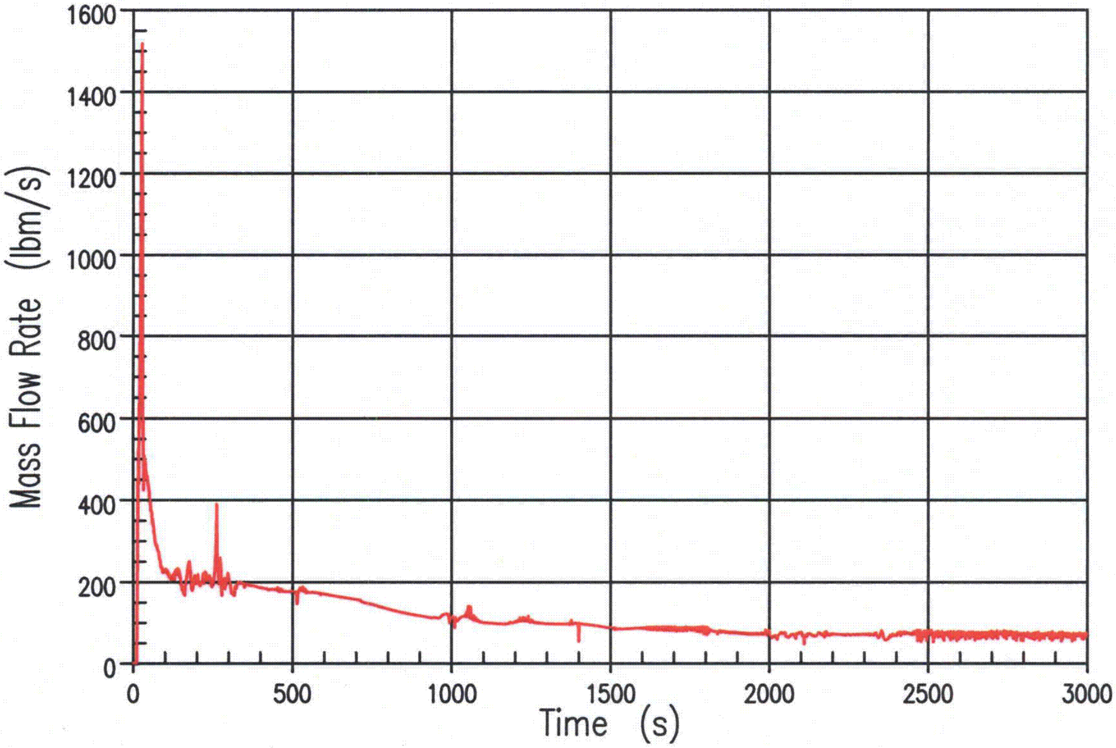


Figure 4.6-3 Core Exit Steam Flow

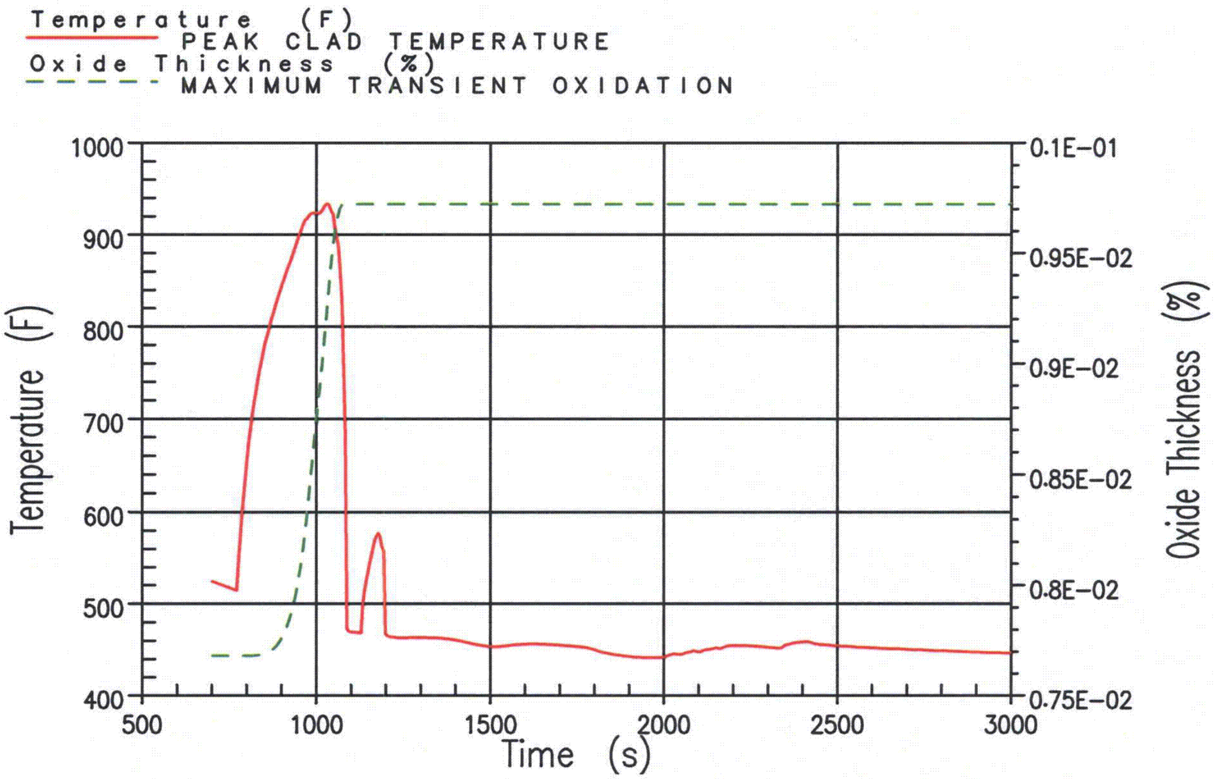


Figure 4.6-4 Clad Temperature & Maximum Transient Oxidation at PCT Elevation

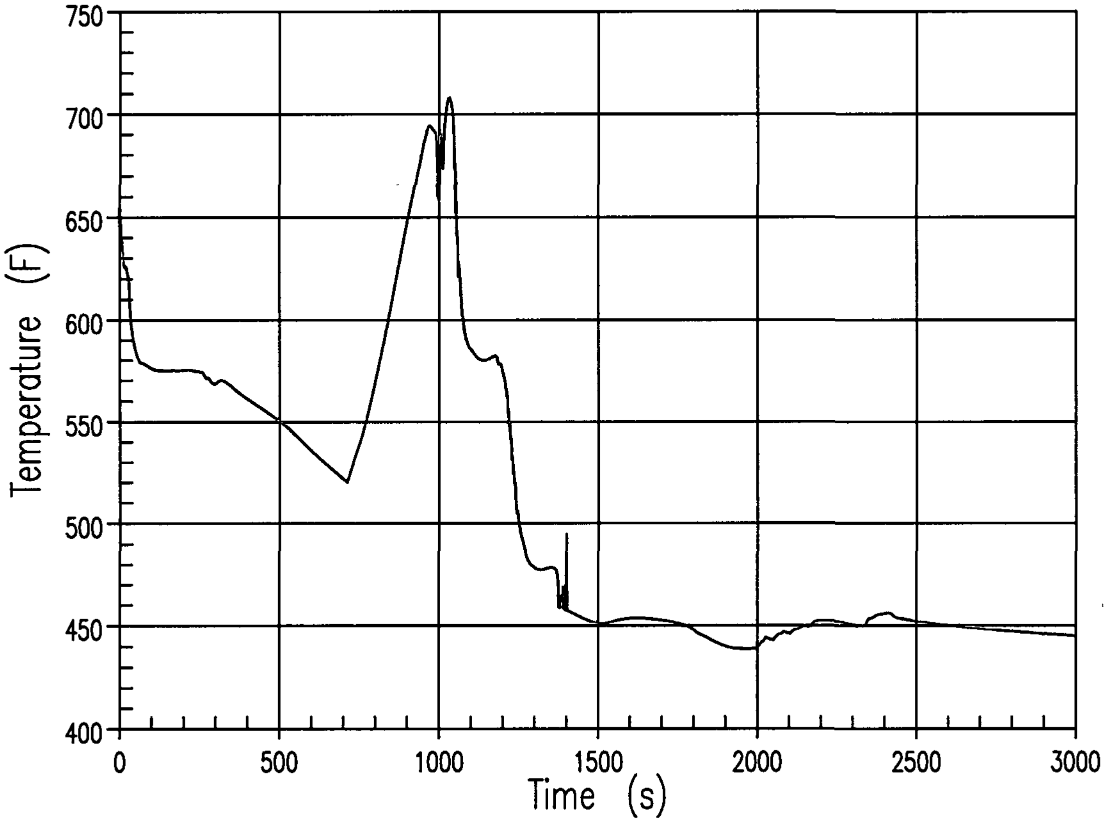


Figure 4.6-5 Top Core Exit Vapor Temperature

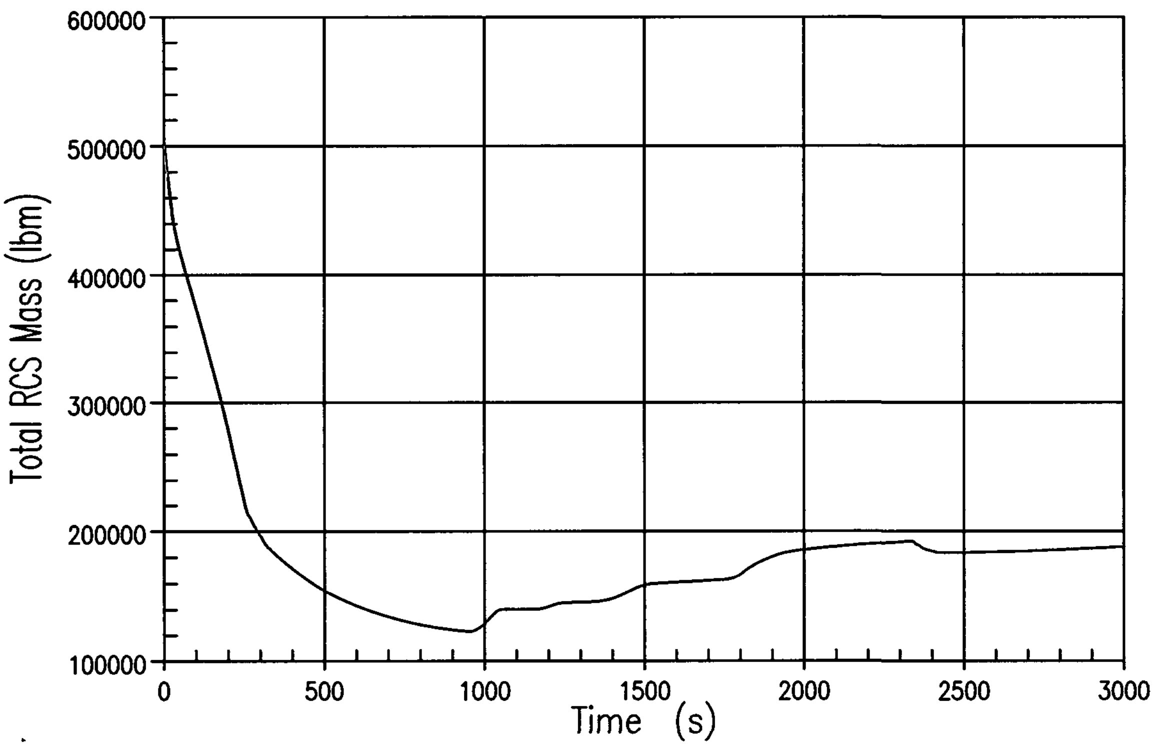


Figure 4.6-6 Total RCS Mass

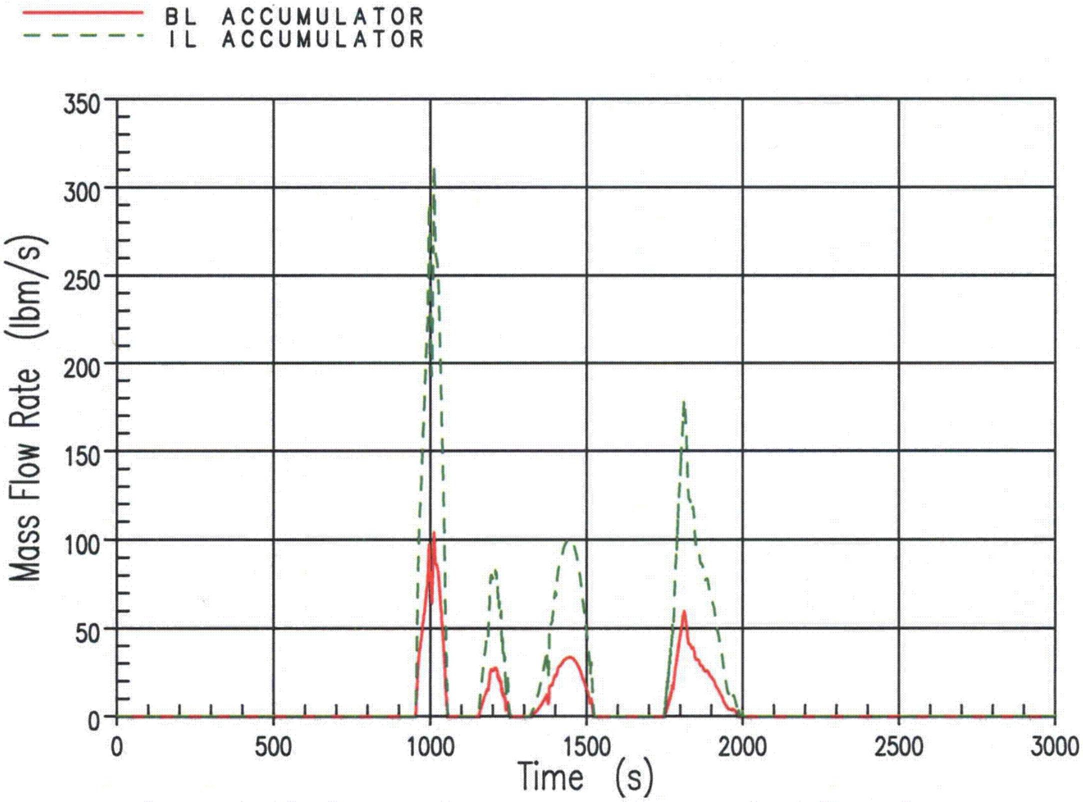


Figure 4.6-7 Broken & Intact Loop Accumulator Flow Rate

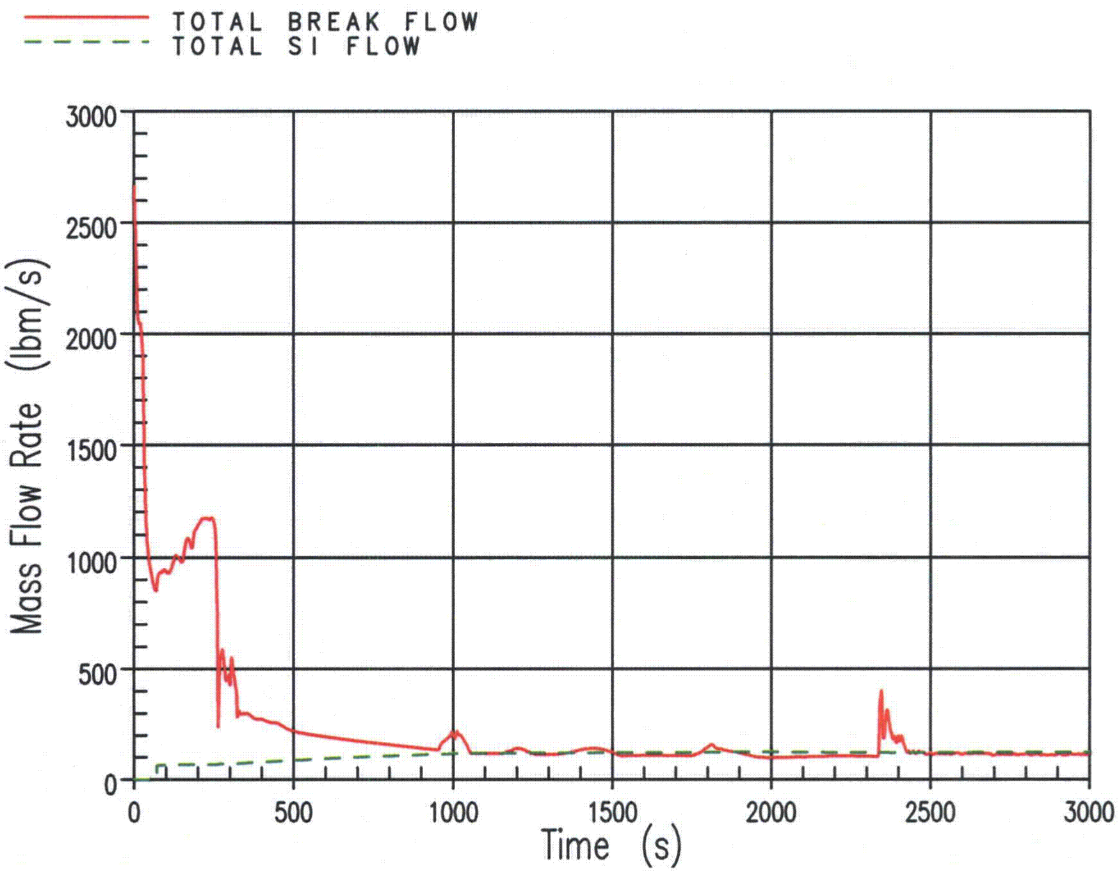


Figure 4.6-8 Total Break Flow Rate & SI Flow Rate

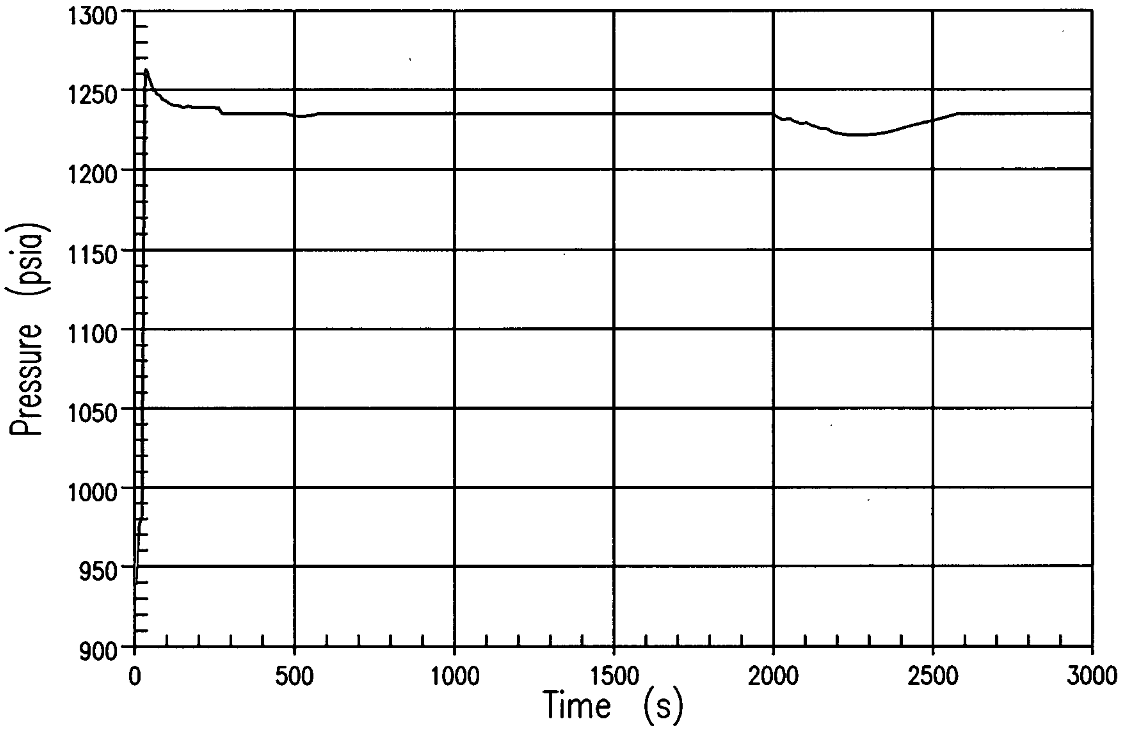


Figure 4.6-9 Broken Loop Steam Generator Secondary Pressure

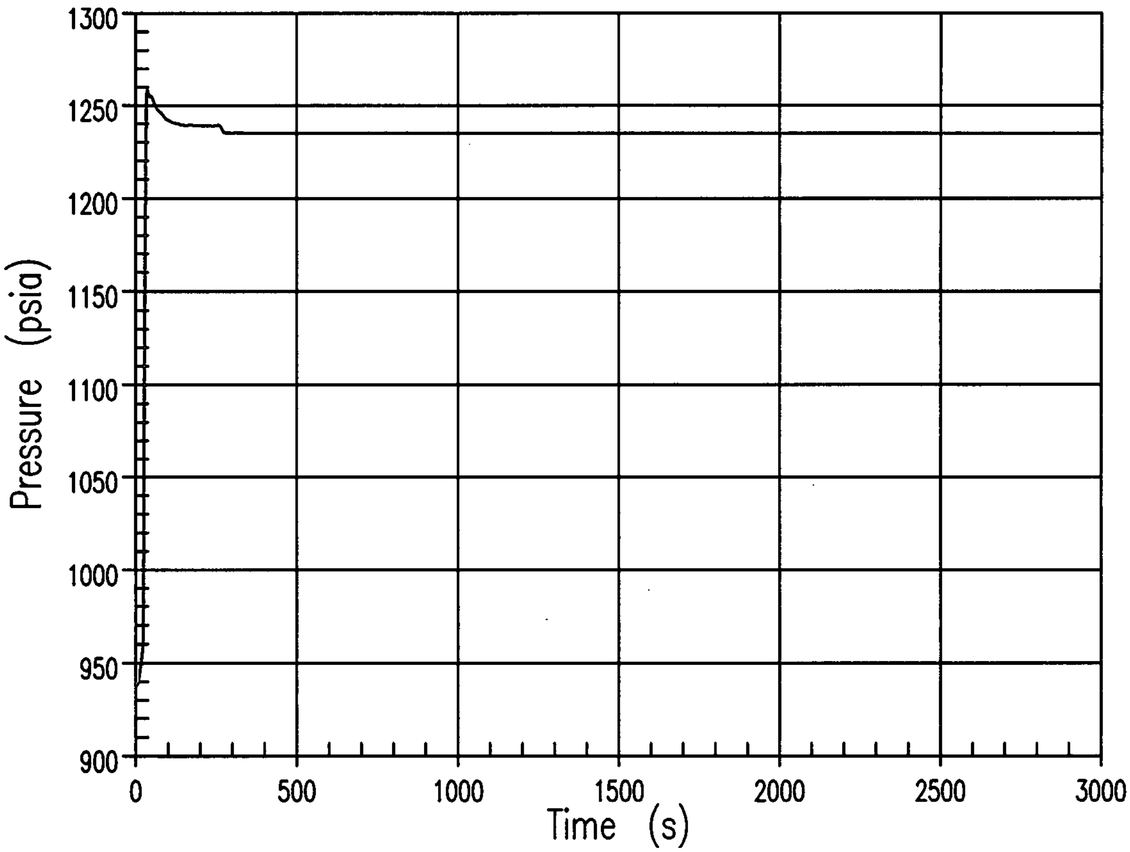


Figure 4.6-10 Intact Loop Steam Generator Secondary Pressure

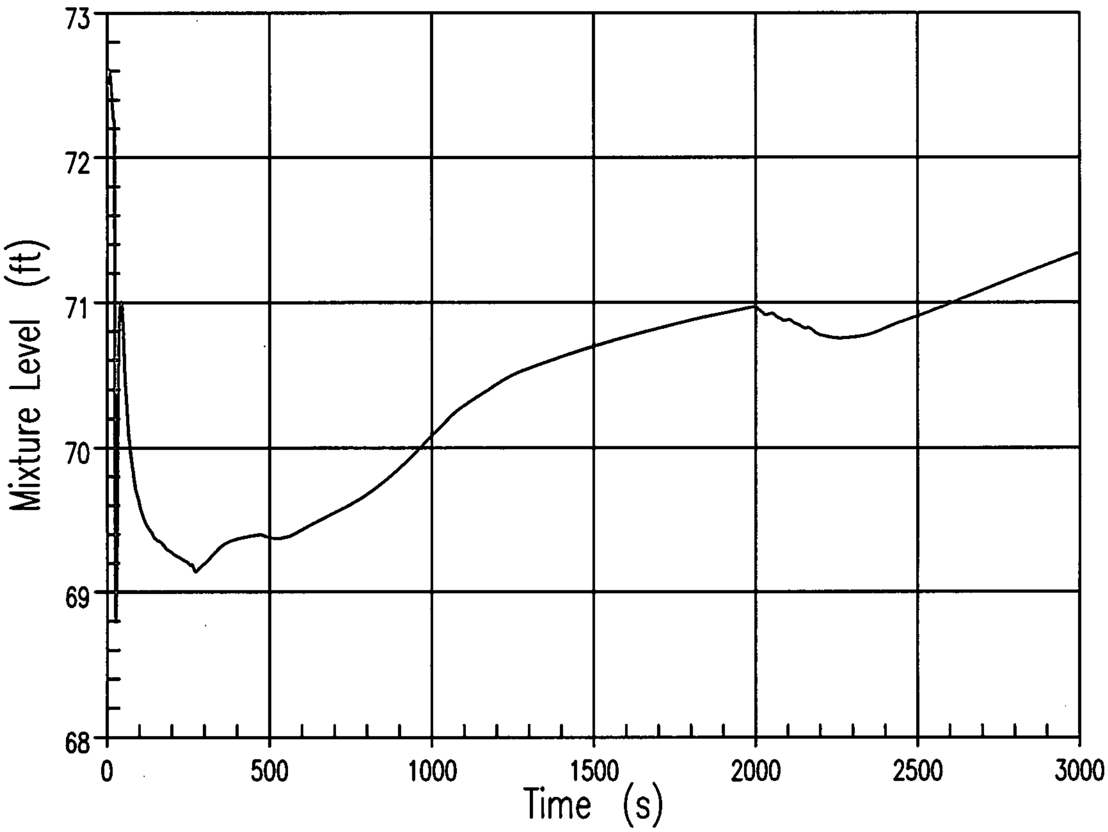


Figure 4.6-11 Broken Loop Steam Generator Secondary Mixture Level

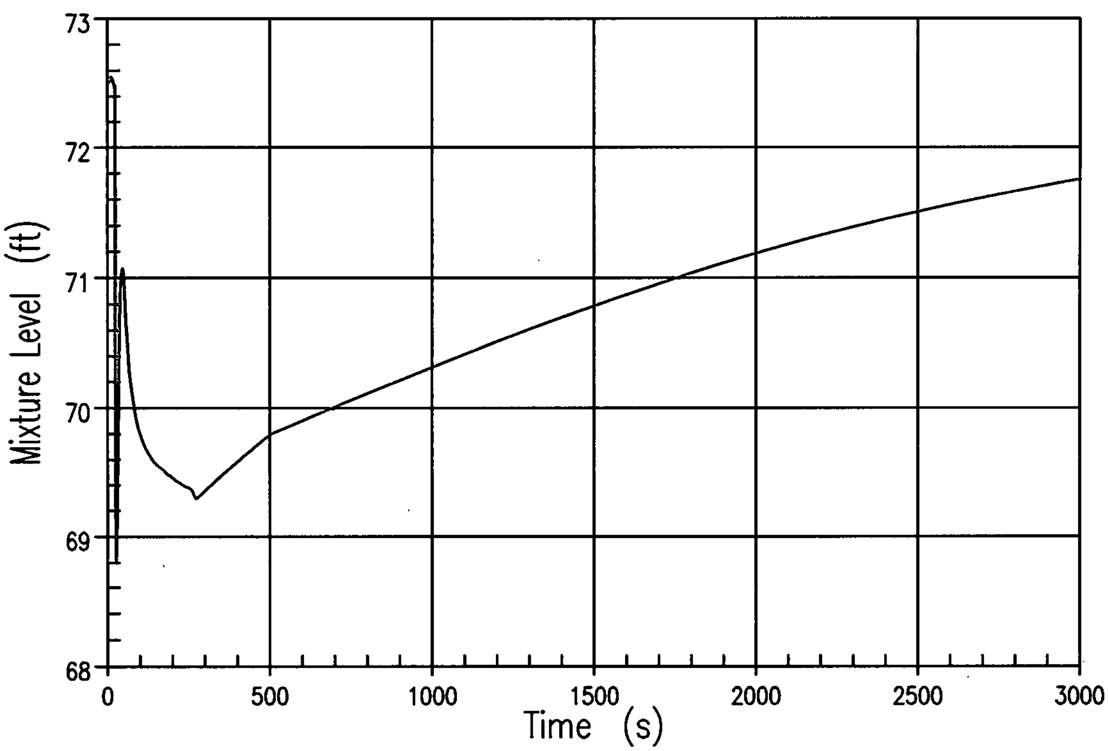


Figure 4.6-12 Intact Loop Steam Generator Secondary Mixture Level

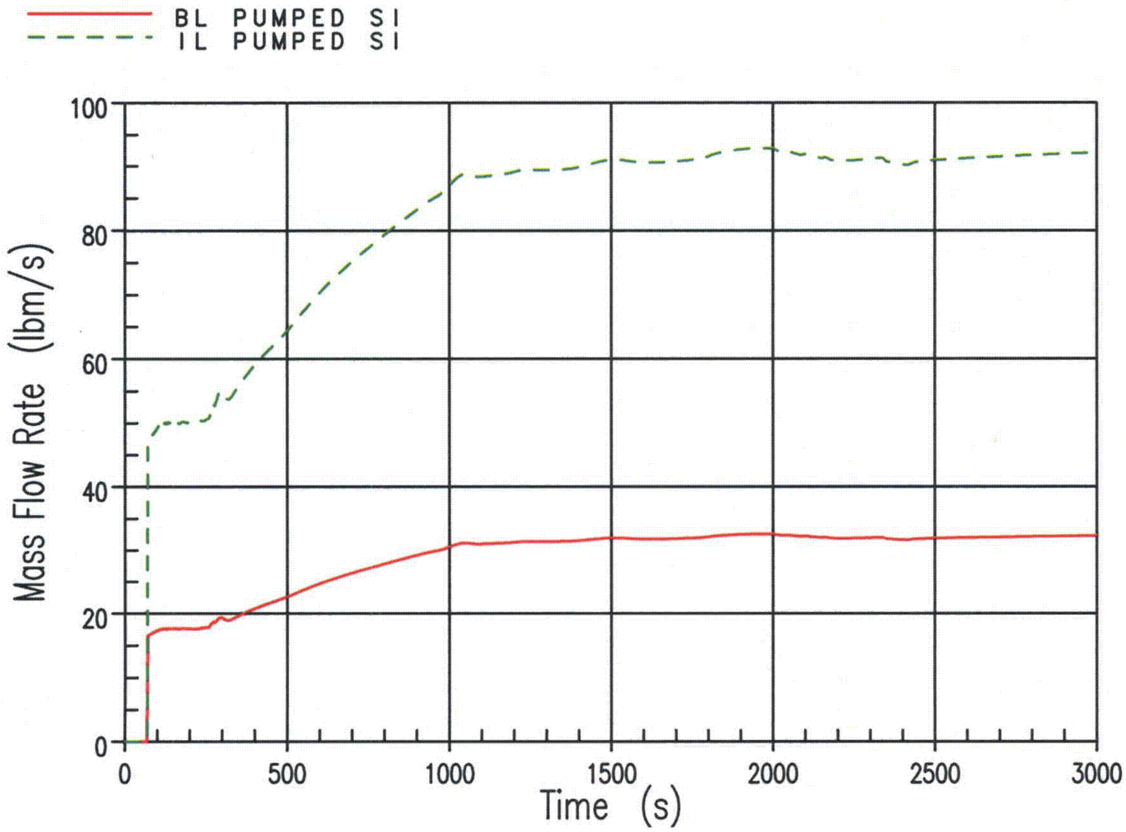


Figure 4.6-13 Broken Loop and Intact Loop Pumped SI Flow

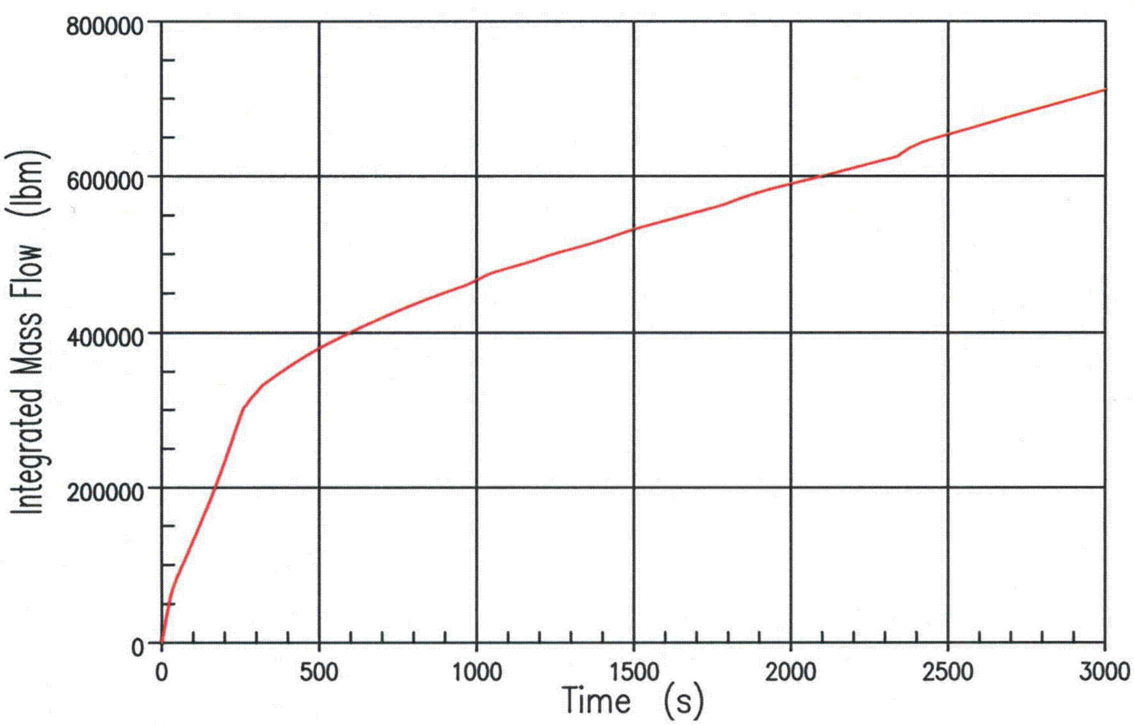


Figure 4.6-14 Integrated Total Break Flow

4.7 Main Steam Line Break Mass and Energy Releases Inside Containment

Introduction

Steam line ruptures occurring inside a reactor containment structure can cause a significant release of high-energy fluid to the containment environment which could result in high containment pressures and temperatures. The high pressures and temperatures can result in failure of any equipment which is not qualified to perform its function in an adverse environment. This could degrade the effectiveness of the protection system in mitigating the consequences of the steam line rupture. In addition, the containment structure is designed to withstand a limited internal pressure (i.e., 60 psig). Thus, an associated containment response analysis may be performed to demonstrate that the conditions inside the containment during a steam line rupture do not violate the existing environmental qualification (EQ) envelopes, and to demonstrate that the containment design pressure is not exceeded.

The proposed MSIV and MFIV and associated actuator replacement will result in an increase in the valve closure time. The increase in the MSIV and MFIV closure time results in additional feedwater being added to the SGs and causes additional mass and energy to be released to the containment. Consequently, there is a potential for the containment pressure and temperature to increase.

In order to confirm that these changes will not result in exceeding the containment design pressure and violating equipment's EQ envelopes, the effect of the proposed change on the MSLB mass and energy releases and the subsequent containment response to a postulated design basis MSLB inside containment must be evaluated.

Furthermore, the reanalysis also addresses issues such as the initial SG mass inventory due to SG water level uncertainty and a larger moderator density coefficient associated with optimized reload designs.

The re-calculated steam mass and energy releases will be used as input to an updated containment pressure and temperature response analysis, which would provide basis for confirming the EQ of equipment located inside containment, and demonstrating that the containment design margin is maintained.

Input Parameters and Assumptions

Consistent with the current licensing basis accident analyses, based on the NRC-approved methodology documented in Reference 11, the analyses are performed to maximize the amount of mass and energy released to the containment. The releases following a steam line rupture are dependent upon many possible configurations of the plant steam system and containment designs, as well as the plant operating conditions and the size of the rupture. There are competing effects as power and break size change, and thus multiple cases are typically analyzed. Therefore, the steam line break event is analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102% of full power (i.e., re-rated power of 3579 MWt). Break sizes are considered beginning with the full double-ended break and decreasing in area until no water entrainment is calculated to occur. The spectrum of powers and breaks analyzed is listed below in Table 4.7-1.

Table 4.7-1: Spectrum of Main Steam Line Ruptures Analyzed

Case #	Power Level(%)*	Break Size (ft^2)	Break Type	Remarks
1	102	full	Double-ended	***
2	102	0.6	Double-ended	
3	102	0.8	Split	
4	75	full	Double-ended	***
5	75	0.55	Double-ended	
6	75	0.84	Split	
7	50	full	Double-ended	***
8	50	0.45	Double-ended	
9	50	0.80	Split	
10	25	full	Double-ended	***
11	25	0.33	Double-ended	
12	25	0.66	Split	
13	0	full	Double-ended	***
14	0	0.20	Double-ended	
15	0	0.40	Split	
16**	0	0.40	Split	MSIV Failure

* The power % is scaled to reference the re-rated power of 3579 MWt

** Same as Case 15, except additional failure of the MSIV on faulted loop has been taken into consideration.

*** An MSIV failure is conservatively accounted for by combining the LOFTRAN results with the hand calculated initial steam line blowdown.

The analysis inputs, assumptions, and methods pertaining to the MSLB mass and energy releases inside containment are presented in this section.

Single-Failure Assumption

The maximum AFW flow delivered to the faulted steam generator represents the most limiting single failure from the perspective of mass and energy releases following a postulated steam line break. Failure of one protection train is assumed so that only one motor and the turbine driven AFW pump are operating during the transient. To maximize the AFW flowrate to the ruptured SG, the control valve on the discharge side of the operating motor driven AFW pump feeding the faulted SG is assumed to fail in the wide open position.

Main Feedwater System

To maximize the water inventory available to be released through the broken line, large values of feedwater flow were used in the blowdown analysis for the double-ended rupture cases. It is assumed that due to the reduced pressure in the broken loop, within 2 seconds the feedwater flowrate will increase up to approximately 350% of nominal (for the full d-e breaks) or approximately 300% of nominal (for the partial d-e breaks). The flowrate will remain at 100%

for the split breaks. The effects of the unisolatable feedwater mass between the faulted SG and isolation valves is also included in the analysis.

Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steam line ruptures [Moody Curve for an f(L/D) = 0 was used]. This is consistent the guideline presented in subsection 6.2.1.4 of the Standard Review Plan (SRP). No entrainment is assumed in the break effluent. The assumption of saturated steam being released for all break types is a conservative assumption that maximizes the energy release into containment.

Unisolable Feed Line Volume

The effects of any flashing of the feedwater trapped between the SG and the isolation valves are included in the analyses. The failure of the MFIV on the faulted loop results in additional fluid being added to the faulted SG. The quantity of the additional fluid to be released is based on the volume between the isolation valve and the MFRV on the faulted loop. Thus, the mass added to the faulted SG from both the pumped main feedwater flow and the feed line flashing will be larger with a failure of a MFIV. For added conservatism, a feedwater flashing volume of 650 ft^3 of unisolable feed line volume is assumed in the analysis. This value is significantly higher than any of the plant specific volumes between the SGs and the MFRVs.

Unisolable Steam Line Volume

The effect of the failure of the MSIV and the associated bypass valve on the faulted loop is considered. It should be noted that closure of the faulted loop MSIV does not terminate the break flow from the faulted SG, since the limiting break is postulated to be located between the SG and the MSIV. However, the faulted loop MSIV and the associated bypass valve do isolate the break from the remainder of the steam line and the other SGs. If the faulted loop MSIV and the associated bypass valve fail to close, blowdown from multiple SGs is prevented by the closure of the corresponding MSIV for each intact SG. But failure of the MSIV and the associated bypass valve does increase the unisolable steam line volume containing steam which will be released to the containment. The modeling methods for accounting for the increased steam line volume due to an MSIV failure is dependent on the break size/type. The details of the LOFTRAN modeling are provided in Appendix F of Westinghouse SAS 12.2 (Reference 12) and is briefly described as follows:

<u>Large Double-ended Rupture</u>

A large double-ended steam line rupture is postulated to occur just downstream of the integral flow restrictor, such that the forward flow comes only from the faulted SG, and all of the steam in the steam line is encompassed in the reverse flow. The initial steam in the steam line piping will blow down through the break immediately after the break and is conservatively determined based on a hand-calculation that assumes a constant flowrate until all of the steam is released out the break. An MSIV failure is conservatively accounted for by combining the LOFTRAN results with the hand-calculated initial blowdown.

<u>Small Double-ended Rupture</u>

A small double-ended rupture is postulated to have two equal break areas, one supplied by the faulted SG (forward flow) and the other supplied by the intact SGs (reverse flow). When steam line isolation terminates the steam flow from the intact SGs, the steam present in the unisolable portion of the steam line will blow down through the reserve flow area. This blowdown is accounted for in the analysis by using LOFTRAN input parameter VOLSH. The input value for VOLSH can be increased to model an MSIV failure.

Split Break

A split break is postulated as a break that is equally supplied by all SGs before steam line isolation. After steam line isolation, the break area is small enough that it can continue to be supplied by a single SG, thus there is little change in the steam mass within the steam line. The unisolable portion of the steam line will not affect the analysis results until the faulted SG inventory is almost gone. The LOFTRAN model with VOLSH is not recommended to be used, since this would cause the steam in the steam line to be released after steam line isolation. Instead, the timing of the release is more accurate if the initial mass of the unisolable portion of the steam line is added to the initial mass of the faulted SG. This method of addressing the steam line modeling is recommended by Westinghouse SAS No. 12.2 in order to accurately reflect the timing of the steam release, subsequent to steam line isolation, for a split break. The amount of the additional mass can be further increased if an MSIV failure is being modeled. For instance, the limiting case (i.e., Case 16) with respect to containment pressure for the split break type has accounted for the failure of the MSIV and the associated bypass valve by adding additional mass to the faulted SG, per the Westinghouse SAS 12.2 recommendation.

SG Fluid Mass

A maximum initial SG mass in all the SGs was used in all of the analyzed cases. The use of a high initial SG mass maximizes the SG inventory available for release to containment. The initial mass has been calculated as the value corresponding to the programmed level (i.e., 50% narrow-range span) plus 10% to account for the SG water level uncertainties, plus 10% to account for mass uncertainties. For split breaks, the mass in the unisolable steam line volume is also included in the initial faulted loop SG mass.

MSIV and MFIV Closure Time

A MSIV and MFIV closure time delay of 15 seconds was conservatively assumed in the analyses of these steam line break events. Note: The actual analysis assumption consists of a total delay of 17 seconds, which includes a 2-second allowance for signal processing delays.

Availability of Offsite Power

Loss of offsite power following a steam line rupture would result in tripping of the reactor coolant pumps, main feedwater pumps, and a possible delay of AFW initiation due to emergency diesel generator starting delays. Each of these occurrences aids in mitigating the effects of the steam line break releases by either reducing the fluid inventory available to feed the blowdown or reducing the energy transferred from the RCS to the steam generators. Thus, offsite power has been assumed to be available as it maximizes the mass and energy released from the break due to 1) the continued operation of the reactor coolant pumps and 2) the continued operation of the feedwater pumps and AFW System.

Operator Response Time

As long as AFW is being delivered to the faulted SG, the steam line break mass and energy release to containment will continue. Operator action is credited to re-align the AFW System to terminate the flow to the faulted SG, while continuing to feed the intact SGs. A 20 minute operator action time confirmed by simulator scenario measurements, is credited in this analysis.

RCS Metal Heat Capacity

The effect of the heat stored in the major components of the primary side of the RCS is explicitly accounted for in the steam line break mass and energy releases inside containment. The major components include reactor vessel inlet and outlet plenum, hot leg and cold leg (including reactor coolant pump), SG inlet and outlet plenum, and reactor vessel dead volume.

SI System

Minimum SI System flowrates corresponding to the failure of one SI System train are assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steam line break inside containment since the mass and energy releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

Protection Systems Actuations

The protection systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, SI, steam line isolation, and feedwater isolation. The analysis setpoints used are conservative values with respect to the plant-specific values delineated in the Technical Specification Bases. The protection system actuation signals modeled in the analysis are identified in Table 4.7-2.

For the full double-ended rupture MSLB at all power level and certain small double-ended ruptures at high power level, the first protection system signal is low steamline pressure (2-of-3 channels per loop, lead/lag compensated in each channel) in any loop that initiates SI and steamline isolation; the SI signal produces a reactor trip signal. Feedwater isolation and AFW actuation occur as a result of the SI signal.

For the split breaks at all power level and certain small double-ended ruptures at median to low power level, the steam line break protection function typically relies on the high containment pressure signals for reactor trip and feed line and steam line isolations. Specifically, a SI signal is generated on a Hi-1 (6 psig) containment pressure signal, and a steam line isolation signal is generated on a Hi-2 (20 psig) containment pressure signal. The timing of these signals must be determined iteratively with the containment response analysis and then modeled in LOFTRAN using "manual" actuation input parameters.

Rod Control

The Rod Control System is conservatively assumed to be in manual operation for all steam line break analyses. Assuming that the reactor is in manual rod control allows for a greater RCS cooldown prior to the reactor trip signal, which maximizes the reactivity feedback at end-of-cycle conditions and produces a greater post-trip power increase.

Core Decay Heat

Core decay heat generation assumed in calculating the steam line break mass and energy releases is based on the 1979 ANS standard (+2σ uncertainty) (Reference 13).

Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions were used to maximize the reactivity feedback effects resulting from the steam line break. Use of maximum reactivity feedback results in higher power generation if the reactor returns critical, thus maximizing heat transfer to the secondary side of the SGs.

<u>Description of Analysis Methods</u>

The MSLB mass and energy releases have been performed, based upon the NRC-approved methodology documented in Reference 11. The system transient that provides the break flows and enthalpies of the steam release through the steam line break has been analyzed with the LOFTRAN (Reference 14) code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and AFW additions, engineered safeguards systems, RCS thick metal heat storage, and reverse SG heat transfer. Note: The LOFTRAN code was used for the current licensing basis MSLB mass and energy releases inside containment analysis.

<u>Acceptance Criteria</u>

There are no specific acceptance limits against which the mass and energy releases calculated for a steam line rupture can be compared. The final results for a steam line break mass and energy release analysis are the containment pressures and temperatures that are determined by the associated containment response calculations, using the mass and energy releases as input.

<u>Results</u>

Using the MSLB analysis methodology documented in Reference 11, combined with the input parameters and assumption listed as above, the mass and energy release rates for the sixteen case spectrum of steam line breaks have been re-calculated. The results of a steam line break mass and energy release analysis are typically a set of tables of mass and energy release rates as a function of time. These recalculated mass and energy release values will replace the current steam line break mass and energy release analysis of record and will be used to update the containment environmental responses for each of the steam line break cases noted. Table 4.7-2 shows the time sequence of events for the steam line break mass and energy releases to containment for all sixteen cases. Tabulated data of the mass and energy release rates are provided in Tables 4.7-3 and 4.7-4 for the limiting peak containment pressure and temperature, respectively. Figures 4.7-1 to 4.7-2 show the plots of the mass and energy releases for these two limiting cases.

Conclusions

Based on an examination of the integrated mass and energy releases, the total MSLB blowdown mass and energy calculated from this calculation seems to be significantly higher than the current analysis of record, especially for the full double-ended break cases. For instance, USAR Table 6.2.1-62 shows the total mass and energy releases of 560.9 x 10^3 lbm and 666.9 x 10^6 Btu, respectively, for the current temperature limiting case, resulting from a full double-ended rupture at 50% power level (i.e., Case 7). Whereas, Case 7 shows total mass and energy releases as being 614.4 x 10^3 lbm and 731.3 x 10^6 Btu, respectively, for the Case 7 scenario. This represents an increase of approximately 10% for both the mass and energy releases. The significant increase in mass and energy releases can be attributed to the longer closure times for both MSIVs and MFIVs, as well as the higher SG fluid mass assumed in this analysis due to SG water level uncertainty. It is expected that the larger mass and energy releases will result in higher calculated containment pressures and temperatures.

It is important to note that the increase in the MSIV and MFIV closure time from 5 seconds to 15 seconds results in additional feedwater being added to the faulted SG and causes additional mass and energy to be released to the containment prior to isolation. Consequently, the calculated peak containment pressure or temperature is expected to increase. Thus, the margin of the containment design pressure is reduced. In order to regain margin (> 10%), some very conservative assumptions of the current MSLB containment analyses, such as the degradation of the fan cooler performance and operator's response time to terminate the AFW flow to the faulted SG have been revised (see the Containment Pressure and Temperature Response to a Postulated MSLB, Section 4.8).

Table 4.7-2 Time Sequence of Events* for the Steam Line Break Mass and Energy Releases to Containment

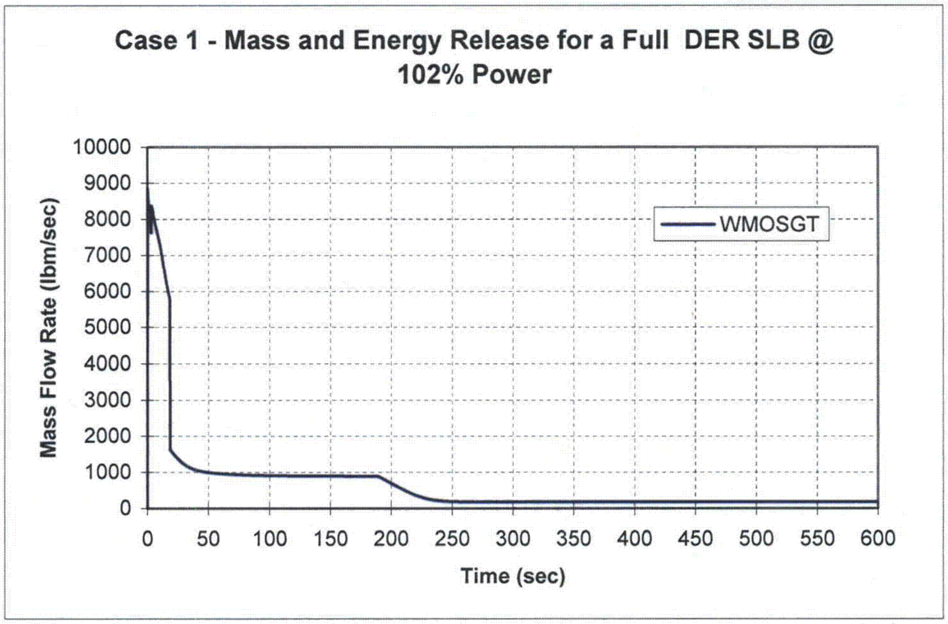
Case	Rx Trip Signal	SI Signal	Steam line Isolation Signal	SI Actuation (sec)	Feedwater Isolation (sec)	Steam line Isolation (sec)	SG Tube Uncovery (sec)	SG Dryout (sec)**
1	SI	LSP	LSP	1.389	18.389	18.389	188.0	276.6
2	SI	LSP	LSP	1.991	18.991	18.991	290.8	580.2
3	OPΔT	Hi-1 Cont P	Hi-2 Cont P	18.7	35.7	86.7	280.2	473.6
4	SI	LSP	LSP	1.209	18.209	18.209	158.4	300.7
5	SI	LSP	LSP	2.800	19.800	19.800	330.2	911.2
6	SI	Hi-1 Cont P	Hi-2 Cont P	16.8	33.8	84.7	285.0	600.8
7	SI	LSP	LSP	1.133	18.133	18.133	164.2	396.7
8	SI	Hi-1 Cont P	Hi-2 Cont P	14.7	31.7	79.5	481.2	545.7
9	SI	Hi-1 Cont P	Hi-2 Cont P	16.7	33.7	89.3	315.6	337.4
10	SI	LSP	LSP	1.115	18.115	18.115	175.2	192.0
11	SI	Hi-1 Cont P	Hi-2 Cont P	18.8	35.8	125.5	698.2	788.7
12	SI	Hi-1 Cont P	Hi-2 Cont P	19.1	36.1	108.2	402.8	443.2
13	SI	LSP	LSP	1.168	18.168	18.168	200.0	217.0
14	SI	Hi-1 Cont P	Hi-2 Cont P	29.7	46.7	219.9	1480.2	1744.0
15	SI	Hi-1 Cont P	Hi-2 Cont P	30.7	47.7	192.7	761.0	832.5
16	SI	Hi-1 Cont P	Hi-2 Cont P	30.7	47.7	192.7	824.0	895.0

* Times noted on this table are the actuation times of that function.

SI - Safety Injection/
LSP - Low Steamline Pressure
OPΔT - Overpower Delta T
Hi-1 Cont P - Hi-1 (6 psig) Containment Pressure
Hi-2 Cont P - Hi-2 (20 psig) Containment Pressure

Table 4.7-3 Steam Line Break Mass and Energy Release Rates, Case 1, 102% Power, Full Double-Ended Break

Time (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (E+6 Btu/sec)	Enthalpy (Btu/lbm)
0	0	0	0
0.2	8857	10.56	1192
0.4	8710	10.39	1192
0.6	8632	10.3	1193
0.8	8515	10.16	1193
1	8443	10.08	1194
1.4	8258	9.862	1194
1.8	8122	9.704	1195
2	8051	9.622	1195
3.2	7633	9.133	1197
3.4	8378	10.03	1197
3.6	8356	10	1197
3.8	8337	9.979	1197
4	8317	9.957	1197
5	8107	9.71	1198
6	7935	9.508	1198
8	7645	9.166	1199
10	7351	8.82	1200
14	6564	7.888	1202
15	6359	7.645	1202
18	5847	7.035	1203
18.4	5783	6.959	1203
18.6	1625	1.956	1204
18.8	1616	1.946	1204
19	1608	1.936	1204
20	1568	1.888	1204
25	1380	1.662	1204
30	1228	1.479	1204
35	1128	1.358	1204
40	1065	1.282	1204
50	998.5	1.202	1204
60	965.4	1.162	1204
70	944.7	1.137	1204
75	936.7	1.127	1203
100	913.5	1.099	1203
125	904	1.088	1203
150	899.4	1.082	1203
175	897.2	1.079	1203
180	897	1.079	1203
185	896.7	1.079	1203
188	896.6	1.079	1203
190	875.4	1.053	1203
195	787.5	0.9466	1202
200	696.9	0.8369	1201
205	611.8	0.7337	1199
210	525.7	0.6294	1197
215	442.8	0.5288	1194
220	369.3	0.4399	1191
225	310.6	0.369	1188
230	266	0.3153	1185
240	215.2	0.2542	1181
250	194.6	0.2295	1179
260	187.2	0.2207	1178
275	184.2	0.217	1178
300	183.5	0.2162	1178
450.2	183.5	0.2162	1178
600.2	183.5	0.2162	1178
900.2	183.5	0.2162	1178
1200	183.5	0.2162	1178



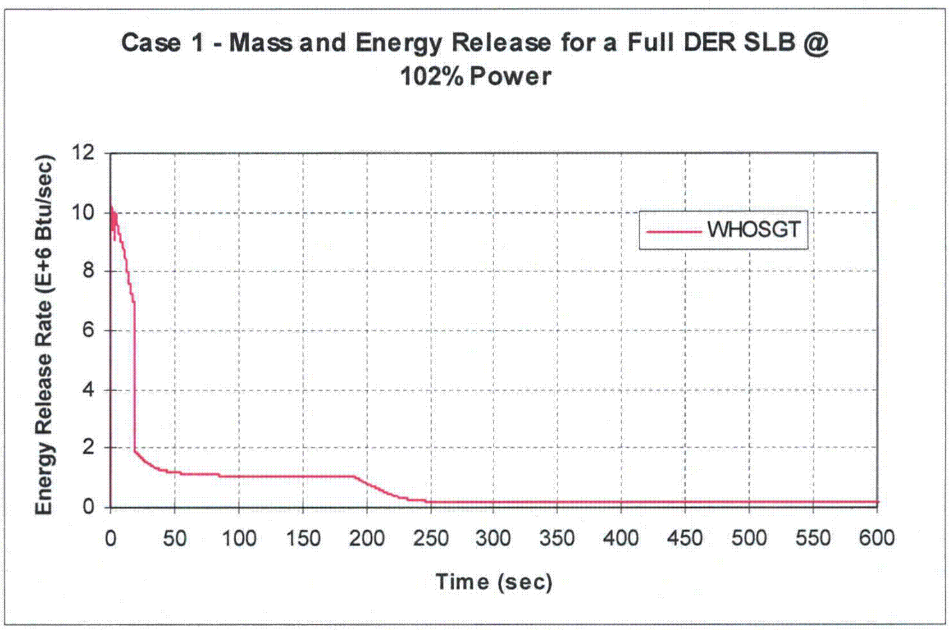
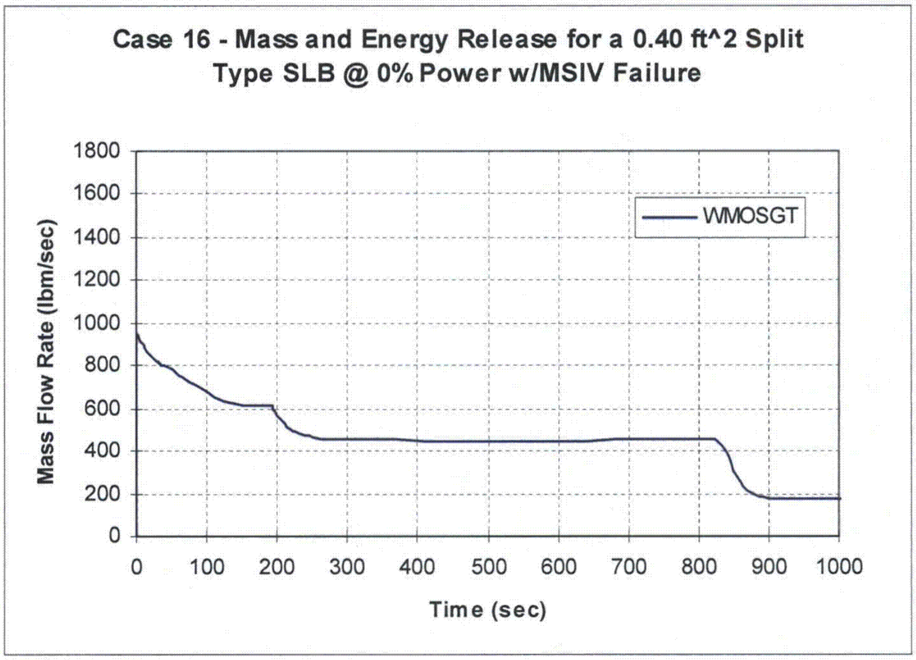


Figure 4.7-1 Steamline Break Mass and Energy Release Rates vs Time; Case 1, 102% Power, Full Double-Ended Break

Table 4.7-4 Steam Line Break Mass and Energy Release Rates, Case 16, Hot Zero Power, 0.40 ft^2 Split Break with MSIV Failure

Time (sec)	Mass Release Rate (lbm/sec)	Energy Release Rate (E+6 Btu/sec)	Enthalpy (Btu/lbm)
0	0	0	0
0.2	957.8	1.136	1186
0.6	954.3	1.132	1186
1	951.1	1.128	1186
2	943.4	1.119	1186
4	929	1.103	1187
6	916.2	1.088	1188
8	904.6	1.075	1188
10	894	1.063	1189
15	870.1	1.035	1190
20	848.8	1.011	1191
25	830.4	0.9896	1192
30	816.4	0.9733	1192
32	811	0.9671	1193
32.6	809.3	0.9651	1193
32.8	808.8	0.9646	1193
35	804.9	0.9601	1193
40	802.8	0.9576	1193
50	782.3	0.9339	1194
60	759.6	0.9076	1195
75	726.6	0.869	1196
100	676.3	0.8102	1198
125	635.2	0.7618	1199
150	616.6	0.7399	1200
175	612.4	0.7349	1200
190	612.2	0.7346	1200
192.6	612.2	0.7347	1200
192.8	612.2	0.7347	1200
195	598.3	0.7183	1201
200	569.9	0.6847	1201
205	547.8	0.6585	1202
210	530.1	0.6374	1202
220	503.8	0.6061	1203
225	494.1	0.5945	1203
230	486.2	0.5851	1203
240	474.5	0.571	1204
250	466.6	0.5616	1204
275	456.4	0.5494	1204
300	452.6	0.5449	1204
450	450	0.5418	1204
600	451.4	0.5435	1204
750	453	0.5453	1204
800	453.4	0.5458	1204
824	453.5	0.546	1204
830	440.2	0.53	1204
835	418.4	0.5038	1204
840	387.1	0.4662	1204
845	349.3	0.4207	1204
850	310.4	0.3738	1204
855	276	0.3322	1204
860	248.5	0.2989	1203
865	227.8	0.2738	1202
870	212.5	0.2553	1202
875	201.4	0.2418	1201
880	193.2	0.232	1200
885	187.4	0.2249	1200
890	183.2	0.2198	1200
900	178.1	0.2136	1199
1200	173.1	0.2076	1199



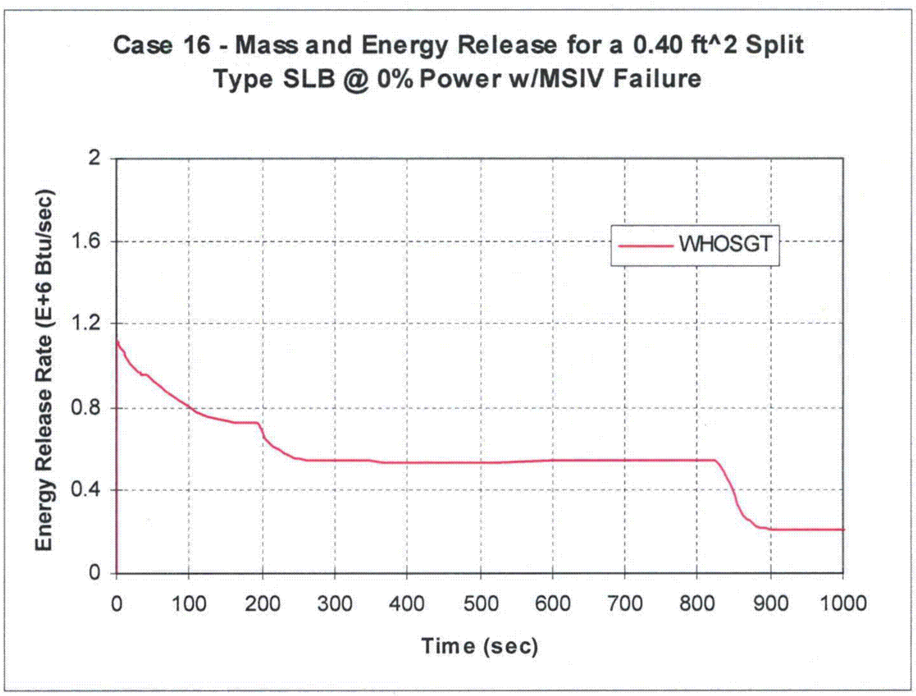


Figure 4.7-2 Steamline Break Mass and Energy Release Rates vs Time, Case 16, 0% Power, 0.40 ft^2 Split Break w/MSIV Failure

4.8 Containment Pressure and Temperature Response to a Postulated Main Steam Line Break (USAR Section 6.2.1.4)

Introduction

The containment structure is designed to withstand a limited internal pressure. In addition, primary or secondary pipe ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment, and high containment temperature and pressure conditions.

The high temperature and pressure conditions cause a failure of any equipment not qualified to perform its function in an adverse environment. This could potentially degrade the effectiveness of the protection system to mitigate the consequences of the event. Thus, it is necessary to demonstrate that the conditions that can exist inside containment during a pipe rupture do not violate the existing EQ envelopes.

To ensure containment integrity, the design and licensing of nuclear power plants require that the reactor containment be analyzed for pressure and temperature effects. These analyses include pressure and temperature transients to which containment may be exposed as a result of postulated line breaks, including LOCAs and secondary system steam and feedwater line breaks. Note: The main feedwater addition is generally below the SG water level; therefore, feedwater line break scenarios always commence with two-phase blowdowns. The enthalpy of the blowdown is less than the enthalpy of saturated steam at the secondary-side operating pressures. As a result, the long-term integrated energy released following a feedwater line break is bounded by the long-term integrated energy released following an MSLB. It is expected that feedwater line break cases would not produce peak containment pressure or temperature conditions as severe as MSLB cases; therefore, feedwater line break cases are not considered for long-term containment pressure and temperature analyses.

The proposed MSIV and MFIV and associated actuator replacement will result in an increase in the MSIV and MFIV closure time, especially at lower system pressures, as the valve is primarily operated with system pressure. The existing MSIVs and MFIVs are operated with electro-hydraulic actuators with a maximum closure time of 5 seconds, and are independent of the system pressure. The increase in the MSIV and MFIV closure time results in additional feedwater being added to the SGs, causing additional mass and energy to be released to the containment. Consequently, there is a potential for the containment pressure and temperature conditions to increase.

Based on an increased closure time of 15 seconds for both MSIVs and MFIVs, the mass and energy releases for a spectrum of MSLB have been re-generated, as summarized in Section 4.7. These revised mass and energy releases to the containment are used as input to update the containment pressure and temperature transients. This section summarizes the updated containment pressure and temperature analyses for a postulated MSLB, which were performed to confirm that the effect of the longer MSIV and MFIV closure time will not result in exceeding the containment design pressure and violating the EQ envelope of the equipment.

The current containment evaluation model for the MSLB scenarios, was based on the CONTEMPT-LT code. WCNOC utilized the GOTHIC program for the updated containment pressure and temperature analyses for a postulated MSLB, as the GOTHIC program is rapidly becoming the industry standard for performing both inside and outside containment pressure and temperature design basis analyses.

The major assumptions and input parameters, consistent with the current licensing basis accident analysis, are described as follows, unless otherwise noted:

1. Loss of offsite power is assumed as it delays the actuation of the containment heat removal systems (i.e., containment sprays and containment air coolers) due to the time required to start the emergency diesel generators. Offsite power has been assumed to be available in the associated mass and energy release analysis as it maximizes the mass and energy released from the break due to 1) the continued operation of the reactor coolant pumps which maximizes the energy transferred from the RCS to the SGs and 2) continued operation of the feedwater pumps and actuation of the AFW System which maximize SG inventory available for release.

2. Loss of one emergency diesel generator, associated with the loss of offsite power, is assumed. As a result, only one train of the containment heat removal systems (i.e., Containment Spray System and Containment Cooling System) is OPERABLE.

3. The heat removal capacity of the containment fan coolers is degraded uniformly by 20% based on their actual performance capability determined by the fan cooler vendor, shown in Table 4.8-1. This is different from the current licensing basis accident analysis, which assumed a degradation ranging from 32% to 95%.

4. The heat removal capability of containment fan coolers is not credited until a total response time of 70 seconds has elapsed. This response time considered the time interval between the time of steam line break initiation/loss of offsite power and the time full Containment Cooling System air and safety grade cooling water flow is established. Purging and filling of the voids that are expected to reside in the fan coolers and cooling water pipe lines as a result of the drain down scenario associated with a loss of offsite power is also accounted for.

5. The containment spray pump performance is assumed to be degraded by 5%. This results in a reduction of the spray injection flowrate from the calculated flowrate of 3086 gpm to 2931.7 gpm.

6. If the containment pressure reaches the containment Hi-3 pressure setpoint (30 psig, including uncertainty) before 27 seconds, full flow spray is conservatively assumed to occur at 60 seconds, accounting for time to attain operating speed and design flow of the containment spray pump and fill up the spray lines. The load sequencer applies power to containment spray pumps at 27 seconds. Otherwise, the containment spray injection starts 30 seconds after the containment pressure reaches the actuation setpoint (i.e., containment Hi-3 pressure). The 30 seconds time delay accounts for the spray pump startup and spray line filling.

7. The surface area for the liquid pool is assumed to be 0 ft^2 in order to neglect the heat transfer from the vapor region to liquid region.

8. Operator action is credited to re-align the AFW System to terminate the flow to the faulted SG, while continuing to feed the intact SGs. Actual termination of AFW flow to the affected SG due to operator action is expected to occur prior to 600 seconds (10 minutes), as discussed in USAR Section 10.4.9. A 20-minute operator action time is credited in this analysis.

Table 4.8-1 Containment Fan Cooler Performance Data*

Inlet Gas Temperature (^{0}F)	Actual Heat Removal Rate (Btu/hr)	Actual Heat Removal Rate (Btu/sec)	20% Degradation (Btu/sec)	CONTEMPT Analysis (Btu/sec)
0	0	0	0	0
95	0	0	0	0
125	8657947	2404.985	1923.988	123.61
131	10659337	2960.927	2368.742	456.945
153	18812329	5225.647	4180.518	2179.165
190	34175810	9493.281	7594.624	5318.055
220	46879993	13022.22	10417.78	8095.835
253	60836032	16898.9	13519.12	11081.95
275	69889629	19413.79	15531.03	13151.39
277	70655965	19626.66	15701.33	13387.5
440	70655965	19626.66	15701.33	13387.5

*** Based on 69400 cfm air flow with essential service water (ESW) flowrate = 1000 gpm per cooler and 95°F ESW water temperature**

<u>Description of Analysis Methods</u>

The updated containment pressure and temperature response evaluation model was developed with GOTHIC version 7.2(a) (Reference 15). The WCGS GOTHIC containment evaluation model for a MSLB event is shown in Figure 4.8-1. The model is comprised of three volumes representing the containment volume, the outside air and a separate volume representing the fan cooler ducts. The containment (Volume 1) is modeled with a single lumped parameter node. Two boundary conditions (1F and 2F) are used to represent the sources of mass and energy from the break and the Containment Spray System, respectively. Flow paths connect the boundary conditions to the containment volume. Fourteen heat sinks, a fan cooler component and a volumetric fan are also shown.

A single GOTHIC input deck file was constructed representing the basic WCGS containment design and contains the data inputs needed to construct the individual cases comprising the prescribed break size spectrum. Individual GOTHIC decks representing the various cases for the different break sizes are constructed by simply assigning the appropriate mass and energy data "functions" defining the MSLB flow for each case and by making a small number of changes within the deck needed to produce each case.

The current CONTEMPT containment evaluation model provides the primary source of the input parameters for the GOTHIC model. These input parameters include:

- Containment minimum free volume: 2.5 x 10^5 ft^3
- Initial conditions: Pressure – 14.7 psia
  Relative Humidity – 50%
  Temperature – 120°F
- Heat sinks: The surface areas and boundary conditions are the same as that shown in USAR Table 6.2.1-4

The MSLB transient mass and energy releases are calculated separately and input to the GOTHIC containment evaluation model via boundary conditions. The break mass and enthalpy are input to the containment evaluation model through forcing functions on flow boundary conditions 1F and 2F. The liquid portion of the break flow is released as drops with an assumed diameter of 100 microns (0.00394 inches).

The containment fan coolers are modeled with a cooler component. There are two fan coolers per train and two trains are normally available. The fan cooler heat removal rate, based on air flow of 69400 cfm and ESW flowrate of 1000 gpm with ESW temperature of 95°F, is given as a function of incoming gas temperature in Table 4.8-1. A 20% performance degradation of the fan coolers, is conservatively assumed to account for the potential situations such as coil tube leaking and blocking of portions of coil modules. The heat removal rate is read into a GOTHIC function and a multiplier, based on the number of fan coolers running, is used to calculate the heat removal rate from containment.

A flowrate of 138800 cfm is used for the volumetric fan to reflect two fan coolers running in the operating train.

Containment spray is modeled with a flow boundary condition, along with a spray nozzle component. There is one spray pump per train, with two trains normally available. GOTHIC requires an input value for the spray drop diameter. From USAR Figure 6.5-2, most of the drops created by the containment spray nozzles have diameters that are less than 1000 microns (0.0394 inches). A mean drop diameter value of 526 microns is presented in the USAR Table 6.5-2 (Sheet 3). Therefore, a drop diameter of 0.0207 inches (i.e., 526 microns) is assumed for the spray in the GOTHIC containment evaluation model.

The major difference between the CONTEMPT and GOTHIC containment evaluation model is the use of the condensing heat transfer coefficients for heat transfer to the exposed passive heat sinks during the steam blowdown phase. The CONTEMPT MSLB containment response model uses the Uchida correlation for the condensing heat transfer coefficients. A 0% re-vaporization input value is used for all of the heat sinks inside containment.

GOTHIC has a number of heat transfer coefficient options that can be used for containment analyses. These include the film, direct, Tagami, and user specified heat transfer coefficient options. GOTHIC also has a number of condensation options that can be used for containment analyses. These include the Uchida, Gido-Koestel, and diffusion layer model options.

The direct heat transfer coefficient set is used for the GOTHIC calculation, along with the diffusion layer model mass transfer correlation, for all of the internal heat sinks in the WCGS containment evaluation model. The diffusion layer model is used to calculate condensation mass transfer between the heat sinks and the atmosphere. The diffusion layer model is described in Reference 16 and the qualification for use in containment design basis analyses are described in Reference 17. The diffusion layer model correlation does not require the user to specify a revaporization input value, as was done in previous analyses using the Uchida correlation.

Acceptance Criteria

The specific acceptance criteria for the containment pressure and temperature response analysis are listed as follows:

1. To satisfy the requirements of General Design Criterion (GDC) 16 and 50 regarding sufficient design margin, the containment should provide at least a 10% margin above the accepted peak calculated containment pressure following a steam line break accident. The specified containment design pressure is 60 psig.

2. The specified containment design temperature is 320°F. However, the 320°F containment design temperature is the design temperature for safety related equipment and instrumentation located within the containment and not the maximum temperature allowed for the containment atmospheric vapor. For certain cases, the calculated containment vapor temperature may exceed the specified containment design temperature for a short period of time. Subsequent evaluations/calculations are then needed to demonstrate that the equipment surface temperatures remain below their design temperatures.

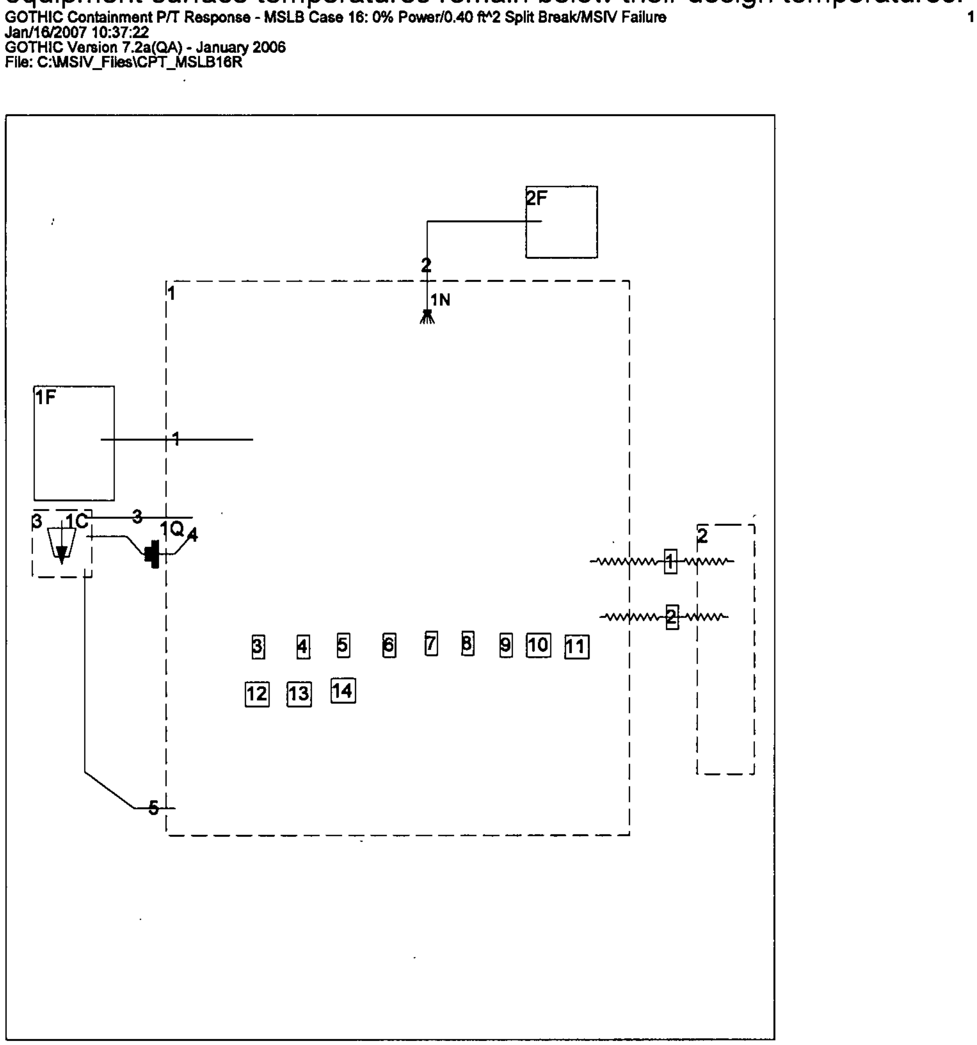


Figure 4.8-1 WCGS GOTHIC Containment Evaluation Model for MSLB Events

Results

The containment pressure and temperature response to a postulated MSLB has been analyzed, based on the developed GOTHIC model, for the 16 cases. The peak calculated containment pressure and temperature for each case is presented in Table 4.8-2. The 0.40 ft^2 split break at hot zero power with an additional MSIV failure (Case 16) and the full double-ended MSLB at the 102% power (Case 1), are found to result in the highest containment peak pressure and temperature, respectively. Figures 4.8-2 and 4.8-3 show the calculated containment pressure, vapor temperature, and sump water temperature for these two limiting cases.

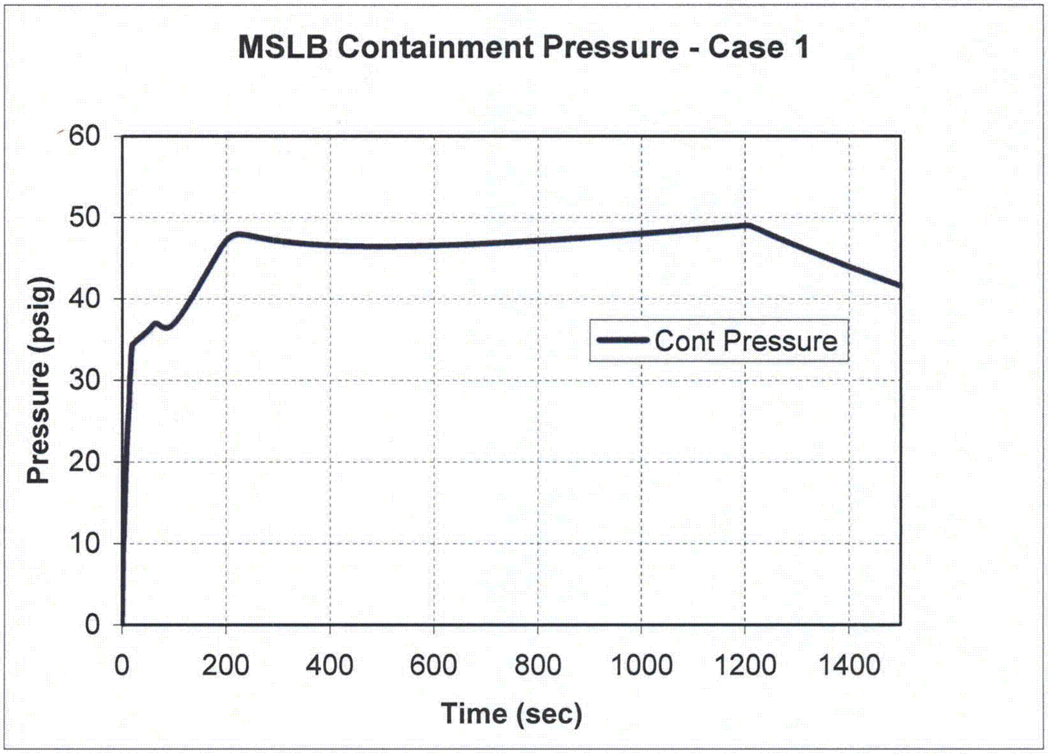
Conclusions

Based on the calculation results presented in Table 4.8-2, the longer MSIV/MFIV stroke time associated with the proposed replacement of the valves and their actuators, has a significant effect on the peak calculated containment pressure and temperature. It is noted that the effect of a longer stroke time on the scenarios with full double-ended break are more pronounced than with the other break scenarios. This is due to the faulted steam generator pressure being at an elevated level for these double-ended break scenarios with the MSIVs being stroked closed and consequently higher mass and energy release rates result. However, the revised assumptions with respect to the containment fan cooler performance and operator response time, allow the containment design pressure to accommodate the calculated pressure conditions with sufficient margin. In addition, use of the diffusion layer model correlation for the condensing heat transfer calculation effectively reduces the containment temperature increase during the initial blowdown of the steam line break.

In summary, the calculated peak containment pressure of 51.50 psig resulting from Case 16 scenario maintains sufficient margin and does not challenge the containment design pressure limit of 60 psig. The highest peak containment temperature of 360.0°F results from a full double-ended break occurring at full power conditions (Case 1). Note that the current containment analysis showed the calculated peak containment temperature was 386.5°F. Since the calculated peak containment temperature is less than that utilized in the current analysis of record for the equipment surface temperatures, no revised equipment surface temperatures analysis is necessary. The existing temperature profiles presented in USAR Figures 3.11(B)-7 and 3.11(B)-7A for the equipment environmental qualification remain bounding.

Table 4.8-2 MSLB Peak Pressure and Temperature Results

Case	Power Level /Break Type	Peak Pressure (psig)	Time of Peak Pressure (sec)	Peak Temperature (^oF)	Time of Peak Temperature (sec)
1	102%/Full DER	49.05	1202	**360.0**	18
2	102%/0.60 ft^2 DER	38.86	1202	296.6	298
3	102%/0.80 ft^2 Split	46.59	1202	303.8	186
4	75%/Full DER	48.66	1202	357.0	18
5	75%/0.55 ft^2 DER	39.50	1202	295.3	322
6	75%/0.84 ft^2 Split	45.57	1202	304.1	192
7	50%/Full DER	49.41	1202	356.8	18
8	50%/0.45 ft^2 DER	42.22	1202	293.4	62
9	50%/0.80 ft^2 Split	46.83	1202	303.4	202
10	25%/Full DER	51.25	1202	357.5	18
11	25%/0.33 ft^2 DER	43.61	1202	272.2	62
12	25%/0.66 ft^2 Split	44.34	1202	299.3	242
13	0%/Full DER	49.42	1202	359.0	18
14	0%/0.20 ft^2 DER	39.82	1204	263.3	1202
15	0%/0.40 ft^2 Split	48.69	1202	289.5	384
16	0%/0.40 ft^2 Split w/MSIV Failure	**51.50**	1202	289.5	382



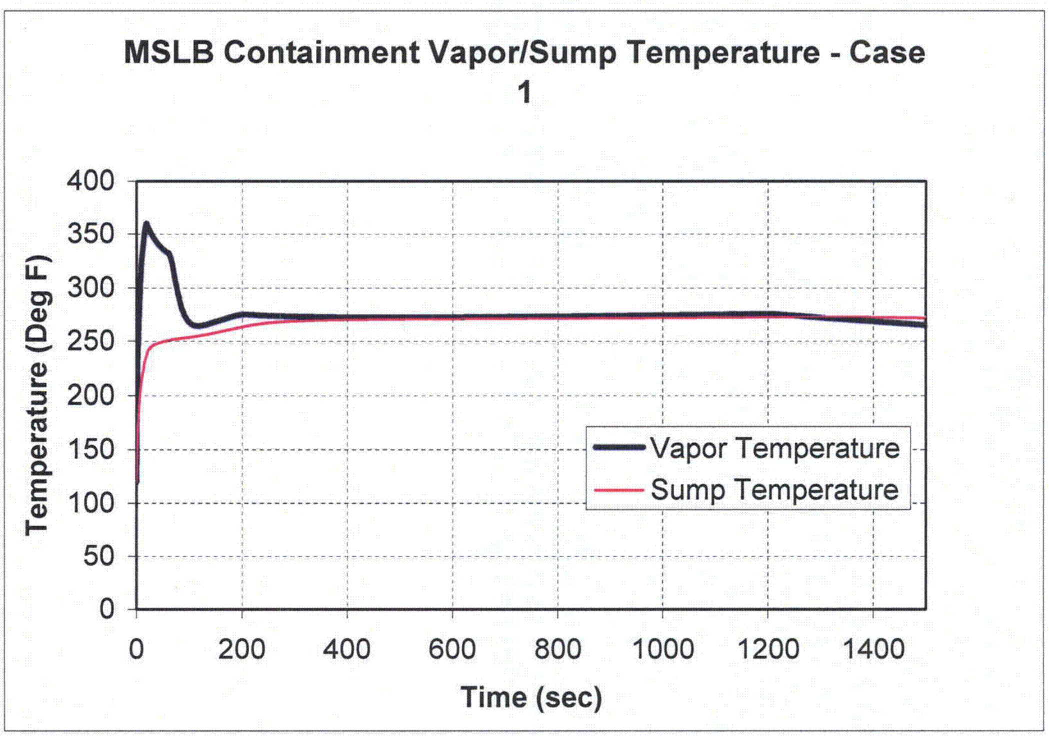
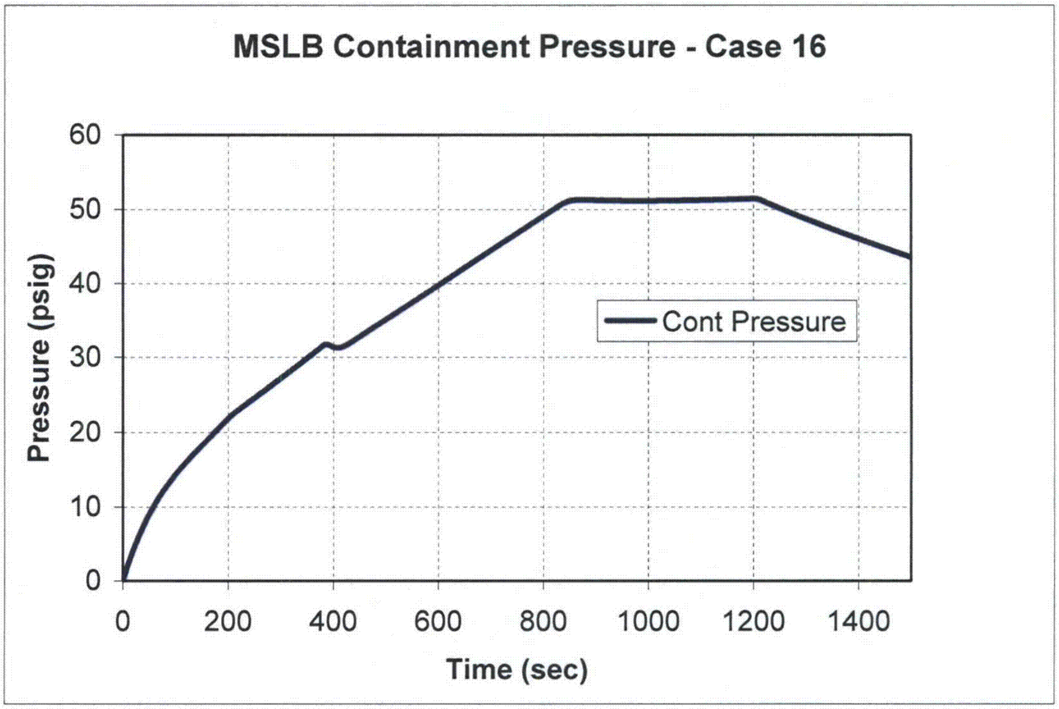


Figure 4.8-2 **Containment Pressure, Vapor Temperature, and Sump Water Temperature Response to a Postulated MSLB – Case 1 Scenario**



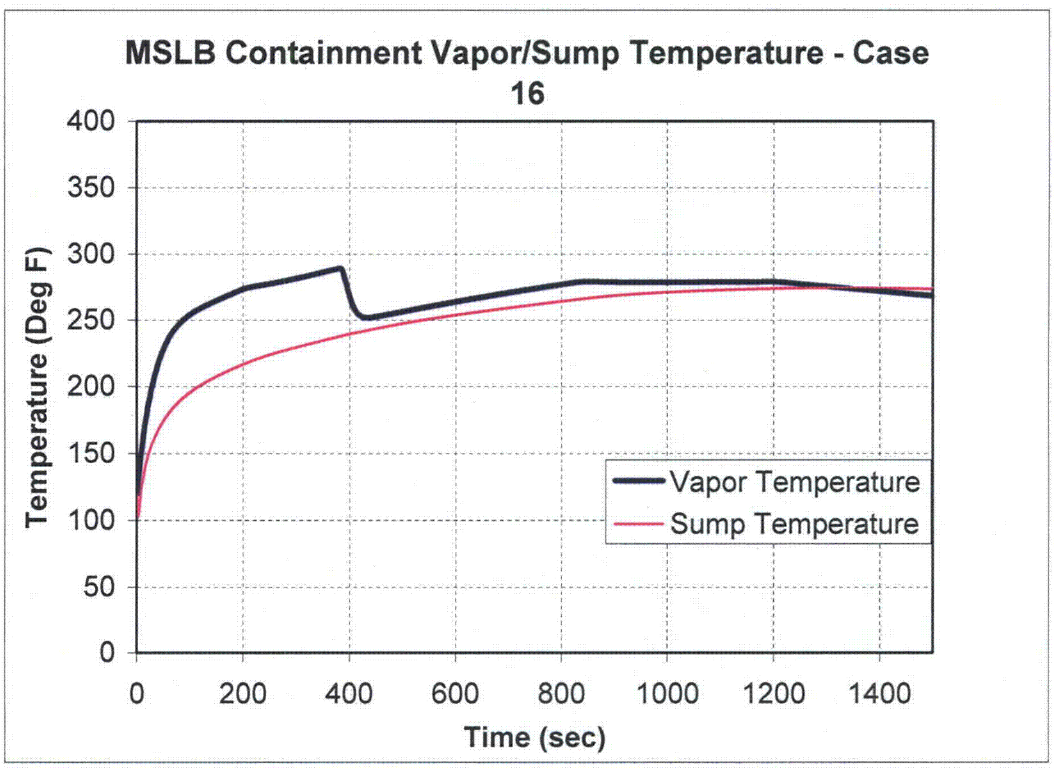


Figure 4.8-3 **Containment Pressure, Vapor Temperature, and Sump Water Temperature Response to a Postulated MSLB – Case 16 Scenario**

4.9 Main Steam Line Break Mass and Energy Releases Outside Containment (USAR Chapter 3, Appendix 3B)

Introduction

Steam line ruptures occurring outside the reactor containment structure may result in significant releases of high-energy fluid to the structures surrounding the steam systems. Superheated steam blowdowns following the steam line break have the potential to raise compartment temperatures outside containment and, therefore, equipment surface temperatures above those originally used for EQ of such equipment. Some equipment, located in compartments outside containment, is required to mitigate a steam line break, provide post-accident monitoring, or provide post-accident plant control. Thus, this equipment must be able to withstand the increased pressures and temperatures associated with superheated steam releases. Examples of such equipment include MSIV solenoid valves, MSIV wiring, ARV components (solenoid, accumulator, potentiometer, wiring), limit switches, junction boxes, seals, etc.

The current licensing basis accident analysis for the steam line break mass and energy releases outside containment is reflected in a generic analysis presented in WCAP-10961 (Reference 18) for the Westinghouse Owners' Group - High-Energy Line Break/Superheated Blowdowns Outside Containment subgroup. This generic analysis was performed in response to the NRC Information Notice No. 84-90 (Reference 19), which raised concerns over the effect of superheated steam releases on the EQ of equipment located outside containment. The analysis supporting the topical report used conservative inputs, so as to provide a limiting analysis for all plants falling within the defined category for the generic analysis. WCGS falls into Category 1, representing the typical 4-loop designs with a power level of 3425 MWt or greater configuration. Detailed conditions for Category 1 are discussed starting on pages 2-4 and 2-17 as well as Tables II-2 and II-3 of WCAP-10961.

Several non-conservative input assumptions have been identified with respect to the Category 1 analysis for the WCGS steam line break mass and energy releases outside containment (Reference 20). These include the assumed value for the minimum shutdown margin and the AFW flowrates. A greater shutdown margin value (1.6% $\Delta k/k$), instead of the WCGS specific shutdown margin value (1.3% $\Delta k/k$), was assumed in the generic bounding analysis. In addition, the AFW flow values used in that analysis were greater than the WCGS specific values. Use of a lower shutdown margin results in higher reactivity insertion and higher power generation, thus increasing heat transfer to the SG secondary. Decreases in AFW flow have the effect of accelerating the time of tube bundle uncovery and increasing the magnitude of tube bundle uncovery and the amount of superheat which can be attained during a steam line break. On the other hand, decreases in AFW flow cause the reduction in the rate of mass and energy release during a steam line break as well as the total mass and energy release because it has the effect of speeding up the depressurization of the SG.

To evaluate the impact of these non-conservative input assumptions on the compartment temperature response, the mass and energy releases have been re-calculated based on WCGS specific information, using the same methodology as the generic analysis. In addition, the reanalysis will also serve the purpose of resolving issues such as reduced initial SG mass inventory due to SG water level uncertainty, longer closure time for MSIVs due to a proposed valve and actuator replacement, and a larger moderator density coefficient to support optimized reload designs in the future. The re-calculated mass and energy releases will be used as input to an associated compartment temperature response analysis, which would provide the basis for confirming the EQ of equipment located outside containment (i.e., in the main steam tunnel).

Input Parameters and Assumptions

Consistent with the current licensing basis accident analysis, assumptions are made that minimize the time to achieve SG tube uncovery, which maximize the superheated release duration. The two most important factors that impact the results of the superheated steam releases following a steam line break outside containment are the RCS temperatures and the time of superheat initiation. RCS temperatures, and thus the superheated steam temperatures, are impacted by the initial power level, reactor coolant pump heat, feedwater enthalpy, AFW flows and enthalpy, reactivity feedback assumptions, and shutdown margin. The timing of the onset of superheated steam generation is primarily a function of break size, initial power level, initial SG mass, and the feedwater flow transient.

The cases examined are based on the results of the analyses presented in Reference 18 for Catergory-1 plants, representing the WCGS configuration. A subset of the cases presented in Table III.B-4 of Reference 18, specifically, Cases 59 through 62, have been identified as representing the WCGS licensing basis superheated steam mass and energy releases outside containment. These four cases have been included in the analysis supporting a reduction in the minimum plant shutdown margin value, revising the analysis assumptions pertaining to the AFW flow, and addressing the recently identified issues mentioned above.

The analysis inputs, assumptions, and important plant conditions and features that were assumed in the MSLB mass and energy releases outside containment are based on the guidelines provided in Reference 21 and are presented in this section.

Initial Power Level

The initial power level defines the conditions for the RCS average temperature, SG inventory, feedwater flow, and SG pressure, as well as the timing of reactor trip and the secondary-side mitigation functions. The initial power which is assumed for steam line break analyses outside containment affects the mass and energy releases and SG tube bundle uncovery in two ways. First, the SG mass inventory increases with decreasing power levels; this will tend to delay uncovery of the SG tube bundle, although the increased steam pressure associated with lower powers will cause a faster blowdown at the beginning of the transient. Secondly, the amount of stored energy and decay heat, as well as feedwater temperature, are less for lower power levels; resulting in lower primary temperatures and less primary-to-secondary heat transfer during the steam line break event.

Overall, steam line breaks initiated from lower power levels, result in lower levels of steam superheating than beaks initiated from full-power conditions. For this reason, steam line break outside containment mass and energy release calculations are limited to breaks initiated from full-power or near full-power conditions (Reference 18). Since the breaks analyzed at full-power conditions comprise the WCGS licensing basis; only full power-maximum allowable NSSS power plus uncertainty, i.e., 102% of rated power is considered in this analysis.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Table 4.9-1 identifies those values assumed for RCS pressure, RCS vessel average temperature, pressurizer water volume, SG fluid mass, and feedwater enthalpy at full-power conditions.

Single-Failure Assumption

Based on the sensitivity analyses performed in support of WCAP-10961 (Reference 18), the most limiting single failure minimizes the AFW flowrates. By minimizing the AFW flow to the faulted SG as well as the unfaulted SGs, tube bundle uncovery occurs earlier in the faulted SG and there is reduced RCS cooling through the unfaulted SGs. Thus, failure of a motor driven AFW train is assumed to be the limiting single failure for this analysis. The failure of one motor driven AFW pump to start results in a minimum AFW flow to the SGs; minimum flow is determined by the remaining motor driven and the turbine driven AFW pumps.

Availability of Offsite Power

The continued availability of offsite power is assumed for the steam line break mass and energy releases. The major effect of the offsite power assumption is the operation of the reactor coolant pumps and it is conservative if the reactor coolant pumps continue to operate throughout the event. A loss of offsite power is not assumed since the resultant calculated mass and energy releases are non-conservatively reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

Break Sizes

The size of the rupture has an impact on the results since larger break sizes result in early SG tube bundle uncovery, i.e., early initiation of superheated steam, and higher resulting compartment temperatures. However, the larger break sizes also result in the earliest actuation of required safety equipment. Smaller break sizes result in later SG tube bundle uncovery, but with delayed safety system actuation. Therefore, a spectrum of break sizes ranging from 0.5 ft^2 to the largest possible steamline rupture area (4.6 ft^2) are included in the analysis. Based on the previous analysis, break sizes smaller than 0.5 ft^2 do not result in tube uncovery and consequential steam superheating and therefore are not considered in this analysis.

SG Tube Plugging

Mass and energy releases following a steam line break are most severe when the primary-to-secondary heat transfer is maximized. This allows for the heat content of the primary side (core and RCS thick metal) to be most effectively transferred into the effluent through the SG in the faulted loop. The assumption of no SG tube plugging represents the condition in which the primary-to-secondary heat transfer is maximized, the steam temperature is maximized, and the SG pressure is elevated. This represents a worse situation than if a SG tube plugging percentage is assumed when defining the RCS operating conditions.

Main Feedwater System

The Main Feedwater System was conservatively modeled for steam line break mass and energy releases outside containment by assuming the following:

- Nominal main feedwater flow corresponds to full power conditions until the time of reactor trip

- Nominal main feedwater temperature corresponds to full power conditions

The rapid depressurization which typically occurs following a steam line rupture results in large amounts of water being added to the SGs through the Main Feedwater System. However, main feedwater flow was conservatively modeled by assuming no increase in feedwater flow in response to the increases in steam flow following the steam line break event. This conservatively minimizes the total mass addition and associated cooling effects in the SGs. High main feedwater temperatures were assumed to minimize the cooling effect of the main feedwater.

Isolation of the main feedwater flow was conservatively assumed to be coincident with reactor trip, irrespective of the function which produced the reactor trip signal. This assumption reduces the total mass addition to the SGs. Closing of the MFRVs in the main feedwater lines is instantaneous with no assumption regarding signal processing or valve closure time.

AFW System

Steam line break sensitivity analyses (Reference 18) have shown that the AFW flow can have a significant impact on the calculated mass and energy releases following a steam line break outside containment. With respect to the production of superheated steam, increased AFW flow has the beneficial effect of reducing the enthalpy of the mass release. Variations in AFW flow can affect steam line break mass and energy releases in a number of ways including break mass flow rate, RCS temperature, protection system actuation times, tube bundle uncovery time and steam superheating.

Generally, within the first few minutes following a steam line break, the AFW System provides a mitigation function by being actuated on SI or Low-Low SG Water Level signal. Addition of AFW to the SGs will increase the secondary mass available to cover the tube bundle and is beneficial as regards to the amount of superheated steam produced. For this reason, the assumed AFW flow is delayed and minimized to accentuate the depletion of the initial secondary side inventory. The AFW flow to all SGs is a function of the backpressure in the SGs.

Operator Action

As long as AFW is being delivered to the faulted SG, the steam line break mass and energy release to the main steam tunnel will continue. Operator action is typically credited to re-align the AFW System to terminate the flow to the faulted SG, while continuing to feed the intact SGs. The operator action time that is credited in the analysis is 30 minutes.

SG Fluid Mass

A minimum initial SG mass in all the SGs was used in all of the analyzed cases. The use of a reduced initial SG mass minimizes the availability of the heat sink afforded by the SGs and leads to earlier tube bundle uncovery. The initial mass was calculated as the value corresponding to the programmed level (i.e., 50% narrow-range span) minus 5% to account for mass uncertainty, and minus 10% to account for the SG water level uncertainty.

Main Steam Line Isolation

Steam line isolation is assumed in all loops except the faulted steam line. Blowdown from the three intact SGs is terminated upon receipt of the signal to isolated and valve closure. The main steam line isolation function is accomplished via the main steam line isolation valves in each of the three unbroken steam lines. A conservative delay time of 17 seconds (15 sec stroke time + 2 sec signal transmission) for steam pressure > 600 psig or 22 seconds (20 sec stroke time + 2 sec signal transmission) for steam pressure < 600 psig is assumed with full steam flow through the valve during the valve stroke. Note: A longer closure time for MSIVs is anticipated when the electro-hydraulic actuators are replaced with system-medium actuators. An inherent design feature of this type of actuator is that the closure time is a function of the system conditions (steam pressure).

RCS Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps. As this occurs, the heat stored in the metal is available to be transferred to the SG with the broken line. Stored metal heat does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick metal masses and heat transfer coefficients.

SI System

Minimum SI System flowrates corresponding to the failure of one SI train are assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available.

Protection Systems Actuations

The protection systems available to mitigate the effects of a MSLB accident outside containment include reactor trip, SI, steam line isolation, and AFW. The setpoints used are conservative values with respect to the plant-specific values delineated in the Technical Specification Bases.

For the largest break size, 4.6 ft^2, the first protection system signal actuated was the Low Steam Line Pressure (lead/lag compensated in each channel, 2-of-3 channels in any loop) signal which actuated SI and steam line isolation; reactor trip being actuated as a result of the SI signal. Main feedwater flow was isolated at the time of reactor trip; AFW initiation occurred as a result of the SI signal.

For break sizes from 1.0 ft^2 down to 0.5 ft^2, the first protection system signal actuated was the Overpower ΔT (2-of-4 channels) or the Overtemperature ΔT (2-of-4 channels) signal; SI was started as a result of a Low Pressurizer Pressure (2-of-4 channels) signal; steam line isolation occurred late due to Low Steam Line Pressure (lead/lag compensated in each channel, 2-of-3 channels in any loop). Main feedwater flow was isolated at the time of reactor trip; AFW initiation occurred as a result of either the SI signal or a Low-Low Steam Generator Water Level (2-of-4 channels in any loop) signal.

Break Flow Model

Piping discharge resistance was not included in the calculation of the releases resulting from the steam line ruptures (Moody Curve for an $f(L/D) = 0$ was used). This maximizes the break flow rate and increases the energy release into the compartment, resulting in a maximum temperature response for the assumed break area.

Rod Control

The Rod Control System was assumed to be in manual operation for all steam line break analyses. Assuming that the reactor is in manual rod control allows for a greater RCS cooldown prior to the reactor trip signal, which maximizes the reactivity feedback at end-of-cycle conditions and produces a greater post-trip power increase.

Core Decay Heat

Application of the 1979 standard (+2σ uncertainty) for the decay heat (Reference 13) was used in calculating the steam line break mass and energy releases. The existing analysis assumed the use of the 1971 standard (+20% uncertainty) for the decay heat as noted on page 2-9 of Reference 18. Since the analysis assumptions documented in Reference 18 and seen by the NRC include the 1971 standard for decay heat, the assumption of using the 1979 version represents a deviation from the prior documented inputs. This version of the decay heat input has been applied previously to the WCGS licensing basis accident analyses.

Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions were used to maximize the reactivity feedback effects resulting from the steam line break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to critical, thus maximizing heat transfer to the secondary side of the SGs. A minimum plant shutdown margin value of 1.3% $\Delta k/k$ has been assumed in the steam line break analysis. The analysis also assumes the full-power EOL MDC value (0.50 Δk/gm/cc) prior to trip and stuck-rod values following reactor trip.

Applicable Criteria

There are no specific limits on the mass and energy releases calculated for a steam line rupture outside containment. Limiting criteria are specified for the temperature analyses that use the mass and energy releases as input. The temperature response in the main steam tunnel (Area 5) is examined in a separate calculation to determine the adverse environment to which different equipment could be exposed.

<u>Description of Analysis Methods</u>

The system transient that provides the break flows and enthalpies of the steam release through the steam line break outside containment has been analyzed with the LOFTRAN (Reference 21) code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, RCS thick metal heat storage, and reverse SG heat transfer.

The following licensing basis cases of the MSLB outside containment have been analyzed at the noted conditions for the MSIV and MFIV and associated actuator replacement:

- At 102% power, break sizes of 4.6, 1.0, 0.7, and 0.5 ft^2

Results

Using Reference 18 as a basis, including parameter changes associated with limiting WCGS plant-specific values for the NSSS model, protection system setpoints, safeguards system setpoints, and AFW flows, the mass and energy release rates have been recalculated. These recalculated mass and energy release values will be used to determine the temperature responses of safety related equipment outside containment for each of the steam line break cases noted. The time sequence of events for the steam line breaks at 102% power is presented in Table 4.9-2. Tabulated data of mass and energy release rates are provided as Tables 4.9-4- to 4.9-7. Figures 4.9-1 to 4.9-4 show the plots of the mass and energy releases.

A review of these results concludes that the level of steam superheating is moderately higher; however, the mass and energy release rates are significantly lower after steam line isolation. This is a result of the lower minimum shutdown margin (1.3%) and lower AFW flow to the faulted SG used in the reanalysis. Table 4.9-3 provides a comparison of the actuation times pertaining to steam line isolation and SG tube uncovery (start of superheat) to the original generic analysis (see Table III.B-4 of Reference 18) and from the reanalysis performed with 1.3% shutdown margin and revised AFW flow.

Timing is the key issue. Prior to steam line isolation, all SGs support the flow through the break. Closure of the MSIVs terminates the break from the unaffected SGs and limits the blowdown to a single SG.

Based upon the above comparison, all cases are seen to have steam line isolation and SG tube uncovery times that occur faster than the original, except for the 4.6 ft^2 case. However, the revised analyses for the 1.0 ft^2, 0.7 ft^2, and 0.5 ft^2 cases show a significantly shorter period of time at superheat prior to steam line isolation than the original analysis results. For the 4.6 ft^2 case, the tubes have not yet been uncovered and thus superheat has not occurred prior to steam line isolation.

Conclusions

The mass and energy releases from a subset of the licensing basis steam line break cases identified in Reference 18 have been reanalyzed with limiting WCGS plant-specific parameters to support reductions in the analysis values for the AFW flowrates and in the minimum plant shutdown margin. The assumptions along with the plant specific parameter values delineated in the "Input Parameters and Assumptions" section have been incorporated into the analysis such that conservative mass and energy releases would be generated. Note that there are no specific limits on the mass and energy releases calculated for a steam line rupture outside containment. Limiting criteria are specified for the EQ analyses that use the mass and energy releases as input. The analysis has been performed with sufficient conservatism to assure that the EQ temperature envelope is maintained even in the presence of superheated steam releases from the SGs. The steam mass and energy releases discussed in this section provide the basis for evaluating the impact on the EQ of equipment located outside containment (i.e., main steam tunnel).

Table 4.9-1 Initial Conditions and Assumptions used in MSLB Outside Containment Mass and Energy Release Analysis

Initial Conditions	
NSSS Power, % Rated Thermal Power	102
Reactor Coolant Flow (total), gpm	374,400
Pressurizer Pressure, psia	2250
RCS Average Temperature, °F	594.9
Pressurizer Water Volume, ft^3	1188.9†
Feedwater Enthalpy, Btu/lbm	426.1
SG Fluid Mass, lbm	88230.0*

† Pressurizer water volume includes a level uncertainty of +5% of span.
* Corresponding to the value at rerated design conditions (103.8×10^3 lbm), minus 10% to account for the SG water level uncertainty and 5% for mass uncertainty.

Table 4.9-2 Transient Summary for the Spectrum of Steam Line Breaks at 102% Power

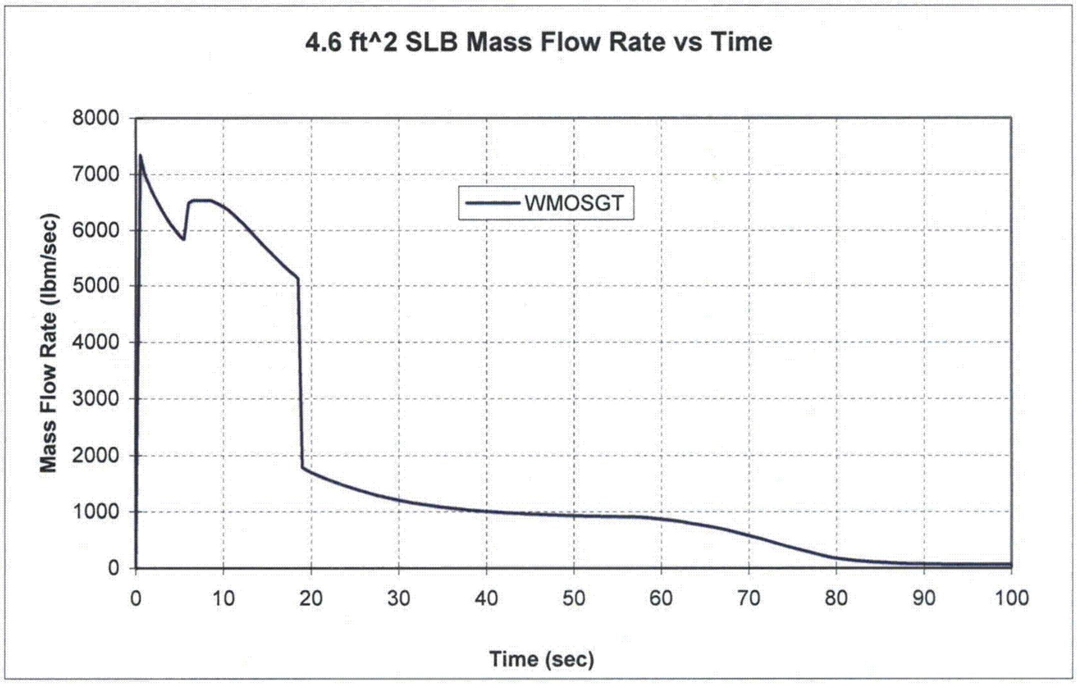
Break Size (ft^2)	Reactor Trip Signal	SI Signal	Rx Trip/FW Isolation (sec)	SI (sec)	Steam Line Isolation (sec)	AFW Actuation (sec)	SG Tube Uncovery (sec)
4.6	SI/LSP	LSP	3.230	28.230	18.230	61.230	57.500
1.0	OPΔT	LPP	13.323	76.577	262.203	109.577	172.500
0.7	OPΔT	LPP	16.828	102.487	383.784	135.487	235.500
0.5	OTΔT	LPP	21.715	141.513	583.342	169.395	289.500

SI/LSP - Safety Injection/Low Steam Line Pressure
OPΔT/OTΔT - Overpower/Overtemperature Delta T
LPP - Low Pressurizer Pressure

Table 4.9-3 Comparison of the Actuation Times Relating to Steam Line Isolation and SG Tube Uncovery

Case No/ Break Size	Steam Line Isolation Time (sec)		SG Tube Uncovery Time (sec)	
	Original	Updated	Original	Updated
59/4.6 ft^2	9.3	18.23	52.5	57.5
60/1.0 ft^2	377.4	262.2	195.5	172.5
61/0.7 ft^2	687.0	383.8	263.5	235.5
62/0.5 ft^2	1800.0*	583.3	463.5	289.5

* Manual Actuation



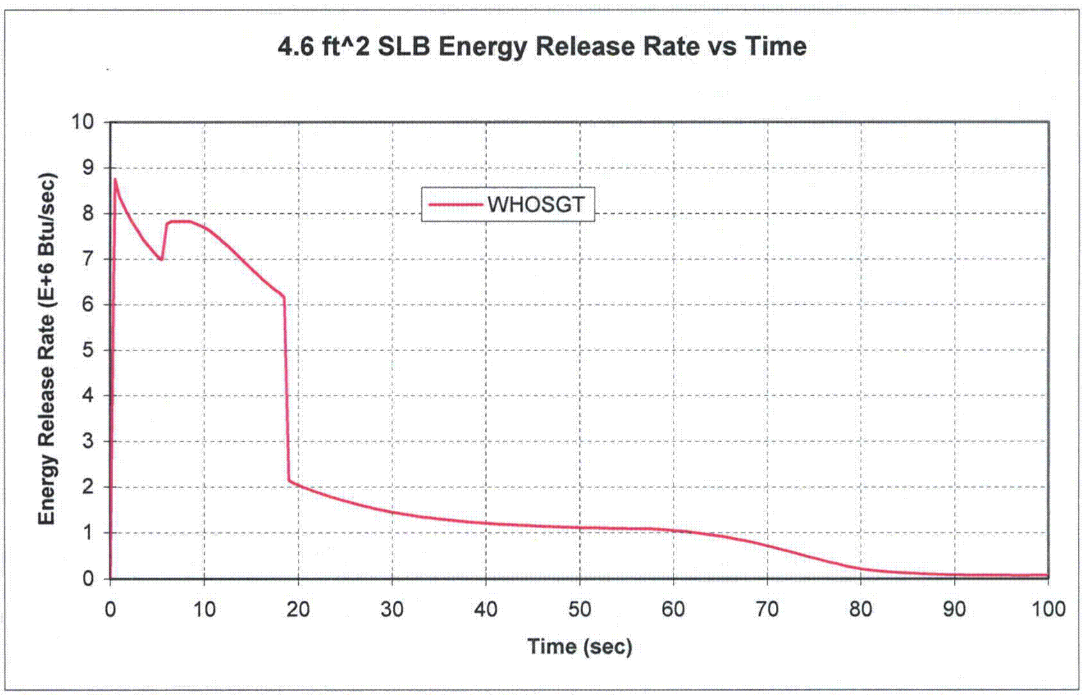
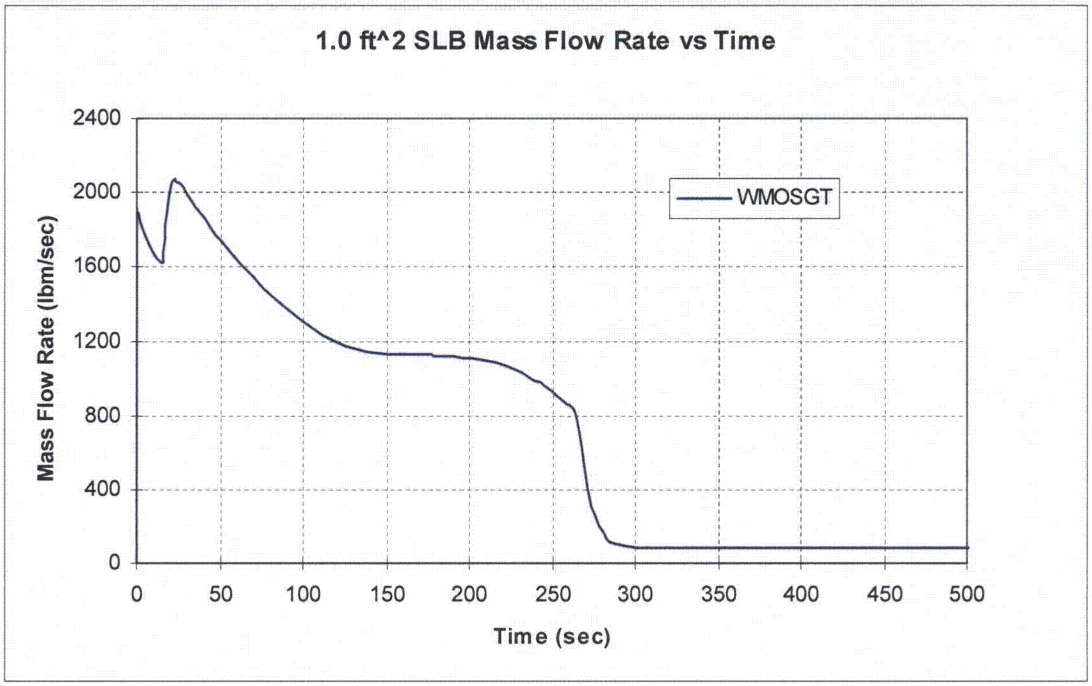


Figure 4.9-1 Steam Line Break (4.6 ft^2) Mass and Energy Flow Rates vs Time

Table 4.9-4 Steam Line Break Mass/Energy Releases for Break Size: 4.6 ft^2

Time (sec)	Break Flow (lbm/sec)	Energy Flow(10^6 Btu/sec)	Break Enthalpy (Btu/lbm)	Saturation Enthalpy (Btu/lbm)	Superheat (Btu/lbm)
0	0	0	0	0	0
0.5	7346	8.765	1193	1193	0
1.5	6819	8.149	1195	1195	0
5	5905	7.078	1199	1199	0
5.5	5834	6.994	1199	1199	0
6	6496	7.789	1199	1199	0
6.5	6537	7.838	1199	1199	0
8.5	6537	7.839	1199	1199	0
9	6503	7.799	1199	1199	0
9.5	6468	7.758	1199	1199	0
11.5	6235	7.483	1200	1200	0
13.5	5909	7.098	1201	1201	0
15.5	5572	6.697	1202	1202	0
18	5207	6.264	1203	1203	0
18.5*	5130	6.171	1203	1203	0
19	1795	2.159	1203	1203	0
20	1706	2.053	1203	1203	0
25	1410	1.698	1204	1204	0
27.5	1298	1.564	1204	1204	0
30	1210	1.457	1204	1204	0
35	1087	1.309	1204	1204	0
40.5	1006	1.211	1204	1204	0
45.5	960.7	1.156	1204	1204	0
50.5	932.5	1.122	1203	1203	0
55.5	915	1.101	1203	1203	0
57.5**	909.5	1.097	1206	1203	3
60.5	864	1.052	1218	1203	15
65.5	744.2	0.9196	1236	1201	35
70.5	563	0.7055	1253	1198	55
75.5	345.3	0.4384	1270	1189	81
80.5	171.5	0.2197	1281	1176	105
85.5	106.8	0.1375	1288	1166	122
90.5	80.11	0.1035	1292	1161	131
100.5	73.74	0.0955	1295	1160	135
120.5	77.46	0.1005	1297	1161	136
140.5	77.38	0.1004	1298	1161	137
160.5	85.98	0.1115	1297	1163	134
199.5	91.88	0.1191	1296	1164	132
299.5	91.89	0.1191	1296	1164	132
599.5	91.87	0.1187	1292	1164	128
1200	91.86	0.1176	1281	1164	117
1800	91.85	0.1165	1268	1164	104

Notes: * Approximate MSIV Closure Time
** Time of Faulted SG Tube Uncovery



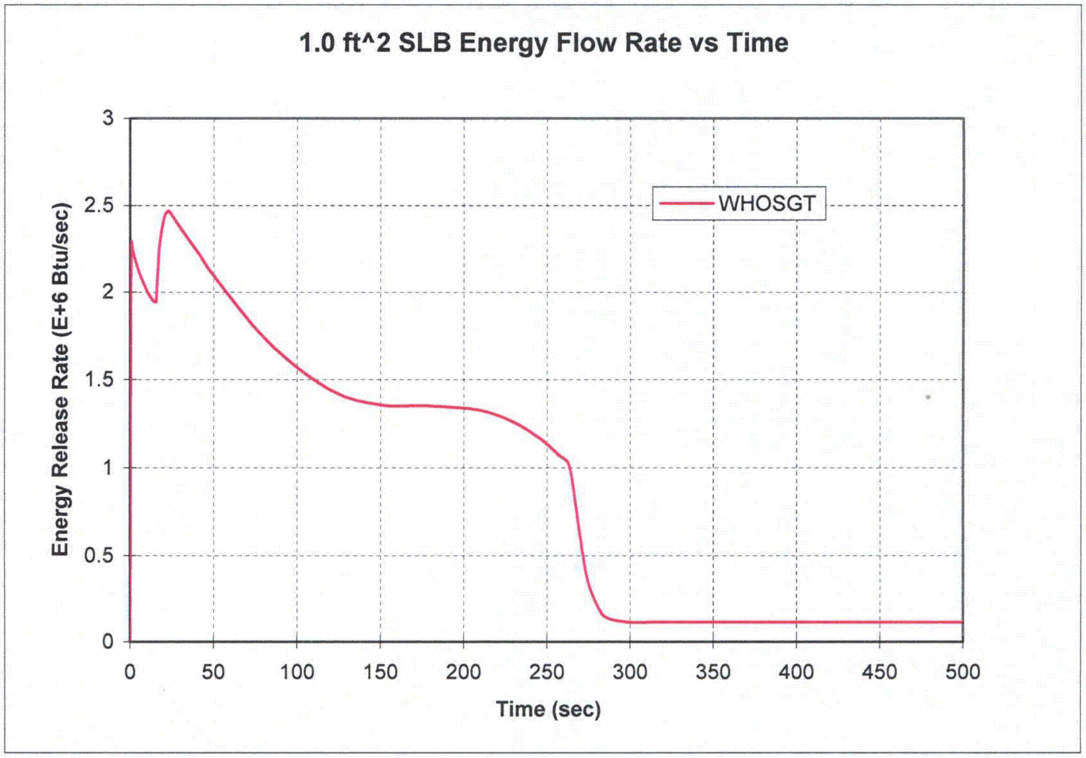
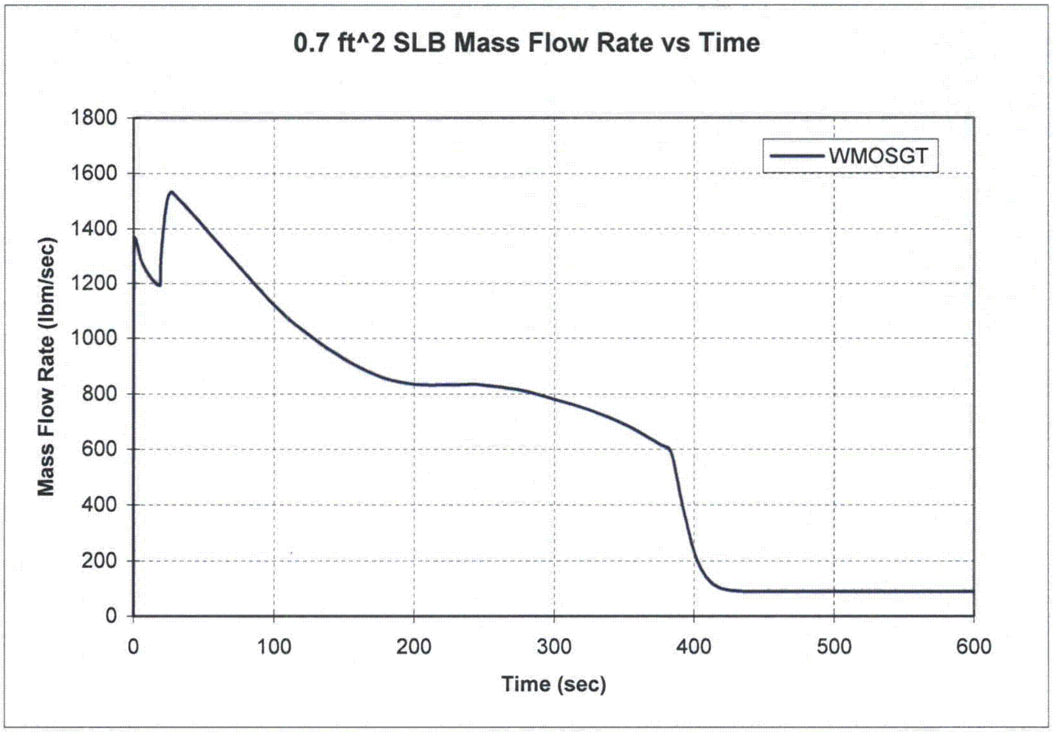


Figure 4.9-2 Steam Line Break (1.0 ft^2) Mass and Energy Flow Rates vs Time

Table 4.9-5 Steam Line Break Mass/Energy Releases for Break Size: 1.0 ft^2

Time (sec)	Break Flow (lbm/sec)	Energy Flow(10^6 Btu/sec)	Break Enthalpy (Btu/lbm)	Saturation Enthalpy (Btu/lbm)	Superheat (Btu/lbm)
0	0	0	0	0	0
0.5	1921	2.29	1192	1192	0
1.5	1877	2.239	1193	1193	0
5	1779	2.125	1194	1194	0
10	1680	2.009	1196	1196	0
14	1628	1.948	1197	1197	0
15	1628	1.948	1197	1197	0
15.5	1626	1.946	1197	1197	0
17.5	1899	2.268	1194	1194	0
20.5	2047	2.44	1192	1192	0
22.5	2073	2.47	1191	1191	0
23	2073	2.47	1191	1191	0
24.5	2060	2.456	1192	1192	0
25	2054	2.448	1192	1192	0
30	1989	2.373	1193	1193	0
40.5	1865	2.229	1195	1195	0
50.5	1745	2.089	1197	1197	0
76.5	1482	1.779	1201	1201	0
100.5	1304	1.568	1203	1203	0
125.5	1178	1.418	1204	1204	0
150.5	1130	1.36	1204	1204	0
172.5**	1125	1.355	1204	1204	0
180.5	1123	1.354	1205	1204	1
201.5	1111	1.34	1206	1204	2
215.5	1085	1.315	1212	1204	8
231.5	1031	1.257	1219	1204	15
245.5	957.4	1.173	1226	1204	22
251.5	916.9	1.127	1229	1204	25
257.5	870.7	1.074	1233	1204	29
263.5*	818.4	1.013	1237	1204	33
273.5	315.7	0.4006	1269	1193	76
281.5	147	0.1878	1278	1179	99
287.5	105.6	0.1353	1280	1173	107
301.5	89.59	0.1147	1281	1170	111
315.5	90.9	0.1164	1280	1171	109
351.5	91.45	0.117	1280	1171	109
451.5	91.43	0.1168	1278	1171	107
601.5	91.42	0.1165	1275	1171	104
901.5	91.4	0.1159	1268	1171	97
1200	91.38	0.1153	1261	1171	90
1500	91.36	0.1146	1255	1171	84
1800	91.34	0.114	1248	1171	77

Notes: * Approximate MSIV Closure Time
** Time of Faulted SG Tube Uncovery



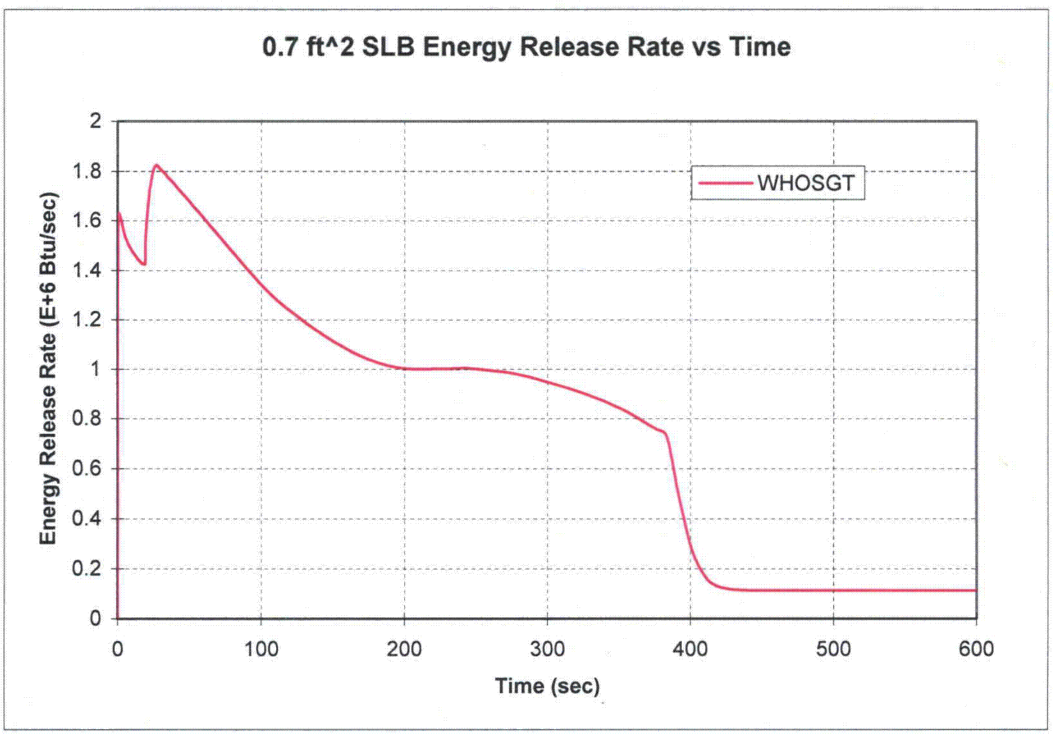
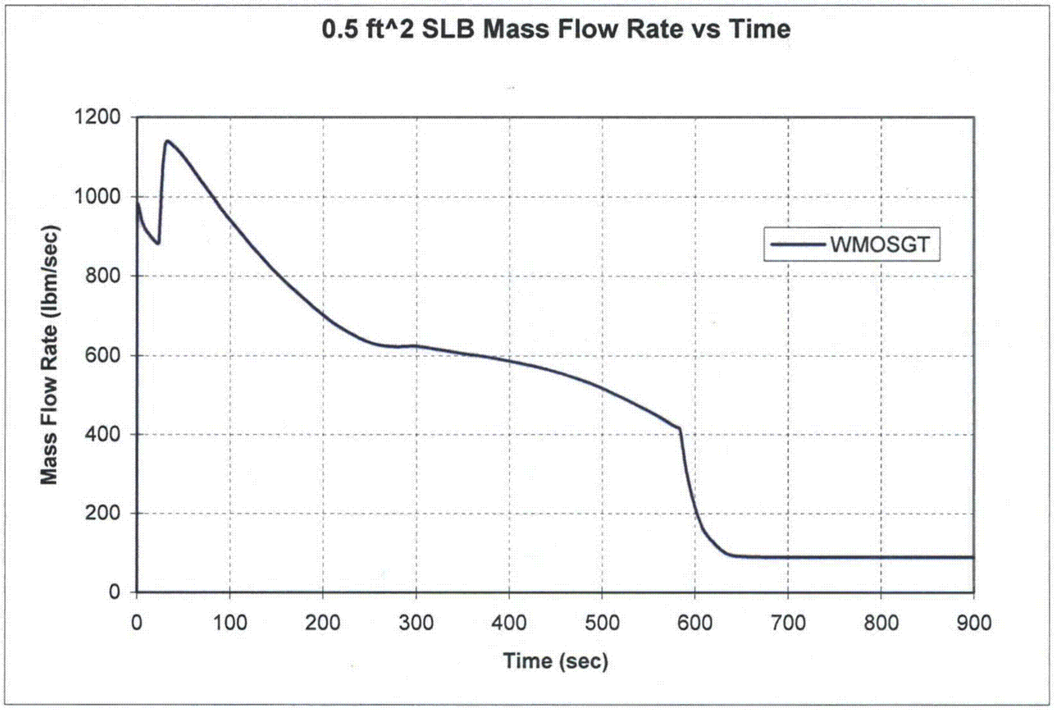


Figure 4.9-3 Steam Line Break (0.7 ft^2) Mass and Energy Flow Rates vs Time

Table 4.9-6 Steam Line Break Mass/Energy Releases for Break Size: 0.7 ft^2

Time (sec)	Break Flow (lbm/sec)	Energy Flow(10^6 Btu/sec)	Break Enthalpy (Btu/lbm)	Saturation Enthalpy (Btu/lbm)	Superheat (Btu/lbm)
0	0	0	0	0	0
0.5	1361	1.623	1192	1192	0
5	1290	1.54	1194	1194	0
10	1238	1.479	1195	1195	0
15	1205	1.44	1196	1196	0
17.5	1194	1.428	1196	1196	0
19	1194	1.428	1196	1196	0
19.5	1287	1.538	1195	1195	0
22	1434	1.709	1192	1192	0
24.5	1516	1.803	1190	1190	0
26.5	1534	1.825	1189	1189	0
27.5	1534	1.824	1189	1189	0
30	1520	1.808	1190	1190	0
40.5	1463	1.743	1191	1191	0
100.5	1119	1.342	1199	1199	0
125.5	1011	1.214	1201	1201	0
150.5	925.3	1.113	1202	1202	0
175.5	863.5	1.039	1203	1203	0
199.5	835.3	1.005	1203	1203	0
225.5	833.1	1.003	1204	1204	0
235.5**	834.3	1.004	1204	1203	1
245.5	834.6	1.005	1204	1203	1
275.5	814.6	0.9852	1209	1204	5
299.5	782.7	0.9502	1214	1204	10
325.5	742.1	0.9048	1219	1204	15
351.5	687.6	0.8428	1226	1204	22
375.5	617.9	0.7621	1233	1204	29
383.5*	590.5	0.7301	1236	1204	32
391.5	403.4	0.5057	1254	1202	52
399.5	241.8	0.3061	1266	1195	71
405.5	169.7	0.2158	1272	1189	83
411.5	128	0.1632	1275	1184	91
415.5	112	0.143	1276	1182	94
421.5	99.37	0.1269	1277	1179	98
431.5	92.47	0.1181	1277	1178	99
441.5	90.94	0.1161	1277	1178	99
451.5	90.73	0.1158	1277	1178	99
599.5	90.78	0.1156	1274	1178	96
901.5	90.75	0.115	1267	1178	89
1200	90.72	0.1143	1260	1178	82
1500	90.69	0.1137	1253	1178	75
1800	90.65	0.113	1246	1178	68

Notes: * Approximate MSIV Closure Time
** Time of Faulted SG Tube Uncovery



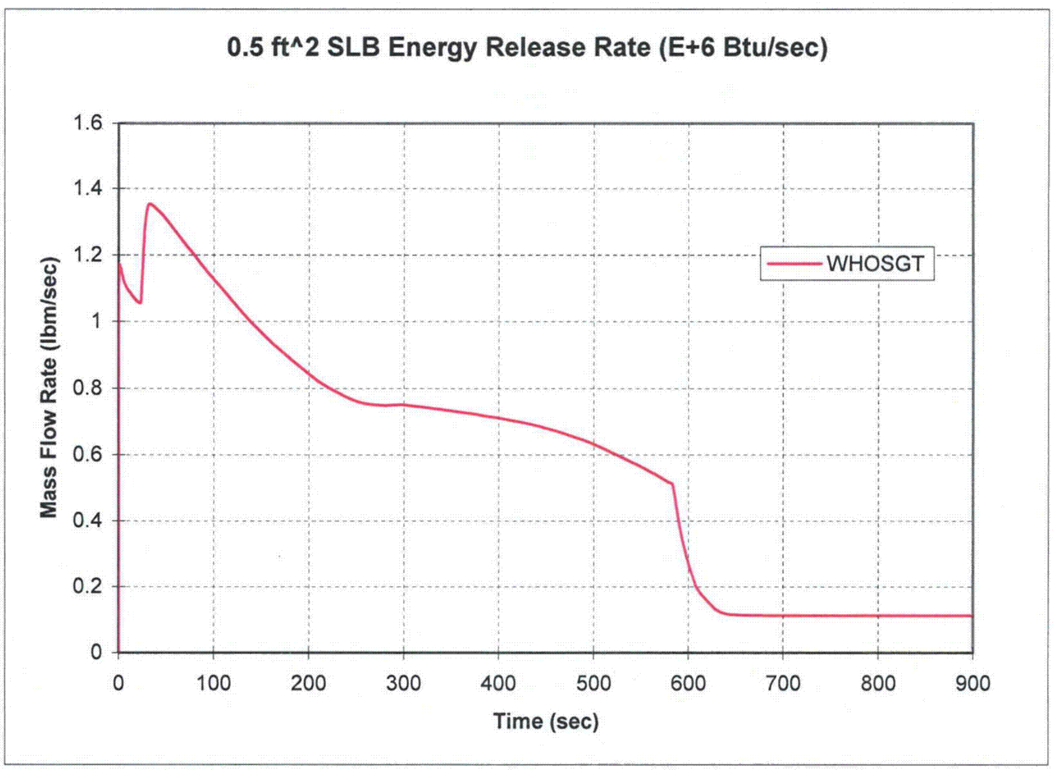


Figure 4.9-4 Steam Line Break (0.5 ft^2) Mass and Energy Flow Rates vs Time

Table 4.9-7 Steam Line Break Mass/Energy Releases for Break Size: 0.5 ft^2

Time (sec)	Break Flow (lbm/sec)	Energy Flow(10^6 Btu/sec)	Break Enthalpy (Btu/lbm)	Saturation Enthalpy (Btu/lbm)	Superheat (Btu/lbm)
0	0	0	0	0	0
0.5	980.3	1.169	1192	1192	0
5	942.9	1.125	1193	1193	0
10	916	1.094	1194	1194	0
20	886.5	1.059	1195	1195	0
22	883.2	1.056	1195	1195	0
22.5	883.2	1.055	1195	1195	0
23.5	885.8	1.058	1195	1195	0
27.5	1076	1.281	1190	1190	0
30	1129	1.341	1188	1188	0
32.5	1141	1.355	1188	1188	0
40.5	1126	1.338	1188	1188	0
50.5	1100	1.308	1189	1189	0
75.5	1019	1.215	1192	1192	0
100.5	941	1.125	1195	1195	0
150.5	806.2	0.9667	1199	1199	0
201.5	699.2	0.8403	1202	1202	0
225.5	661.1	0.7949	1202	1202	0
251.5	631.7	0.7599	1203	1203	0
275.5	623.2	0.7498	1203	1203	0
289.5**	624.2	0.7511	1203	1203	0
301.5	623.7	0.7511	1204	1203	1
351.5	605.6	0.7321	1209	1203	6
375.5	597	0.7228	1211	1203	8
425.5	574	0.6974	1215	1204	11
451.5	557.5	0.6787	1217	1204	13
475.5	539.1	0.6577	1220	1204	16
501.5	515.2	0.6303	1223	1204	19
551.5	457.2	0.5628	1231	1204	27
575.5	425.1	0.525	1235	1204	31
579.5	419.7	0.5186	1236	1204	32
583.5*	414.2	0.5121	1236	1204	32
589.5	323.9	0.4039	1247	1203	44
593.5	274.8	0.3441	1252	1202	50
599.5	219.6	0.2763	1258	1199	59
605.5	178.6	0.2254	1263	1196	67
611.5	148.6	0.1881	1266	1193	73
631.5	101.6	0.1291	1271	1186	85
651.5	91.57	0.1165	1272	1185	87
699.5	89.88	0.1142	1271	1184	87
1200	89.79	0.1131	1260	1184	76
1800	89.66	0.1117	1246	1184	62

Notes: * Approximate MSIV Closure Time
** Time of Faulted SG Tube Uncovery

4.10 Main Steam Tunnel (Area 5) Temperature Analysis (USAR Chapter 3, Appendix 3B)

Introduction

This analysis determines the main steam tunnel compartment temperature transients and peak pressures, based upon the mass and energy releases resulting from the postulated steam line breaks outside containment. The specific assumptions associated with the development of these mass and energy releases outside containment are discussed in Section 4.9. The mass and energy release transients are postulated to occur in the west main steam tunnel of the Auxiliary Building.

GOTHIC Computer Code Model

The compartment analysis is performed with the GOTHIC version 7.2(a) code (Reference 15). GOTHIC is a multi-node containment code developed by Numerical Applications, Inc. (NAI). GOTHIC is becoming the industry standard for performing containment and outside containment compartment transients for design basis events.

The compartment model for the main steam tunnel is comprised of two lumped parameters nodes representing the west and east compartments. The third and fourth nodes represent the environment and containment. Flow boundary condition 1F is used to represent the source of mass and energy from the break. The east and west steam tunnel compartments are connected by a flow path that models the clear areas through column AC between the compartments. Venting from each compartment to the environment is modeled with flow paths 3 and 4, respectively. Pressure boundary condition 2P is employed to maintain the environment node closely to atmospheric conditions through the transient. Each node includes a heat loading from the intact steam lines. The break is assumed to occur in the west compartment.

The GOTHIC model schematic is presented in Figure 4.10-1. Table 4.10-1 provides a summary of the compartment volume, flow path and heat sink data for the main steam tunnel GOTHIC model.

Results

Four break cases were generated for the GOTHIC analysis of the main steam tunnel. Each break is assumed to occur in the west compartment of the steam tunnel. These included 102% power level and break sizes ranging from 0.5 ft^2 to 4.6 ft^2. The details of the input assumptions that determine the mass and energy releases for these postulated transients are provided in Section 4.9.

For these break sizes, the compartments mix and heat up together during the initial blowdown from the steam line. The compartment gas temperatures increase even more rapidly as the SG tubes uncover and the break flow becomes superheated. Compartment gas temperatures peak at approximately 436°F (Figures 4.10-2 through 4.10-5). As SG inventory depletes, the break flow rate deceases, and the compartments become less pressurized. The large density difference between the hot compartment gases and the ambient air begins to overcome the momentum of the break and natural circulation of ambient air into the steam tunnel begins. The break flow exits the steam tunnel through the vent in the west compartment, and the natural circulation draws air into the steam tunnel through the vent in the east compartment. Cool air enters the break compartment from the east compartment through the clear areas in column AC.

The addition of ambient air into the steam tunnel rapidly cools down the system, and the east (non-break) compartment temperatures decrease and stabilize between the initial compartment temperature and the ambient temperature. At the same time, the west (break) compartment temperatures decrease and stabilize below 300°F.

The peak temperatures and pressures that were predicted in each compartment for the four break size cases analyzed are presented in Table 4.10-2. Figures 4.10-2 through 4.10-5 show plots of the temperature response of each break case in each region.

Conclusions

A GOTHIC model was created to simulate the steam tunnel in the Auxiliary Building and the compartment response to postulated main steam line break transients at full power operating conditions. The compartment steam temperatures within each region were calculated. The peak compartment temperature was calculated to be 436°F, occurring in the west compartment at approximately 84 seconds after event initiation. A comparison of these temperature profiles with that presented in USAR Figures 3B-6a through 3B-6d for the EQ of the equipment installed in the main steam tunnel reveals that they are bounded by the existing temperature envelope. Note: The current analysis of record showed the calculated peak compartment temperature was 469°F. Therefore, it is concluded that the existing justification of the EQ of the equipment remains valid.

Table 4.10-1 Main Steam Tunnel GOTHIC Model Parameters

NODE	Description	Volume (ft^3)	Bottom Elevation	Top Elevation	Initial Relative Humidity
1	MST West	59,098.92	2,026'	2,088'-2"	0.70
2	MST East	59,239.92	2,026'	2,088'-2"	0.70
3	Environment	1.0E+08	2,000'	3,000'	0.50
4	Containment	2.5E+06	2,000'	2,205'	0.50

Flow Path	Description	Upstream Node	Downstream Node	Flow Area (ft^2)	Loss Coefficient
1	Clear Areas through Column AC	1	2	633.81	0.82
2	Break Flow	1F	1	--	--
3	MST West Vent	1	3	203.14	0.82
4	MST East Vent	2	3	203.14	0.82
5	Atmospheric pressure	3	2P	1.0E+05	0.001

Heat Sink	Material	Node	Surface Area (ft2)	Thickness (ft)	Boundary Condition	
					Left	Right
1	Structural Steel	1	4,287.13	0.042	DLM	Adiabatic
2	Structural Steel	2	4,300.96	0.042	DLM	Adiabatic
3	Concrete Floor	1	945	2	DLM	Adiabatic
4	Concrete Floor	2	945	2	DLM	Adiabatic
5	Concrete Column	1	3,200	1	DLM	Adiabatic
6	Concrete Column	2	3,200	1	DLM	Adiabatic
7	Concrete Column	1	3,352.5	2	DLM	Adiabatic
8	Concrete Column	2	3,352.5	2	DLM	Convective
9	Interior Concrete	1	1,665	1	DLM	Adiabatic
10	Interior Concrete	2	1,665	1	DLM	Adiabatic
11	Concrete Column	1	1,639	2	DLM	Adiabatic
12	Concrete Column	2	1,639	2	DLM	Adiabatic
13	Reactor Bldg Wall	1	1,865	4	DLM	Isothermal
14	Reactor Bldg Wall	2	1,865	4	DLM	Isothermal
15	Concrete Roof	1	365	2	DLM	Convective
16	Concrete Roof	2	365	2	DLM	Convective

DLM – diffusion layer model

Table 4.10-2 Main Steam Tunnel Peak Temperature and Pressure Resulting from a Steam Line Rupture, Superheated Blowdown

Break Size (ft^2)	Peak Temperature (Deg F)		Peak Pressure (psia)	
	West Compartment	East Compartment	West Compartment	East Compartment
0.5	413.7	361.9	14.738	14.734
0.7	426.0	376.0	14.759	14.754
1.0	434.9	384.5	14.796	14.788
4.6	436.0	375.2	15.497	15.437

MST Temperature Response to a 4.6 ft^2 MSLB
Jan/26/2007 08:42:27
GOTHIC Version 7.2a(QA) - January 2006
File: C:\MSIV_Files\mst_mslb46r

1

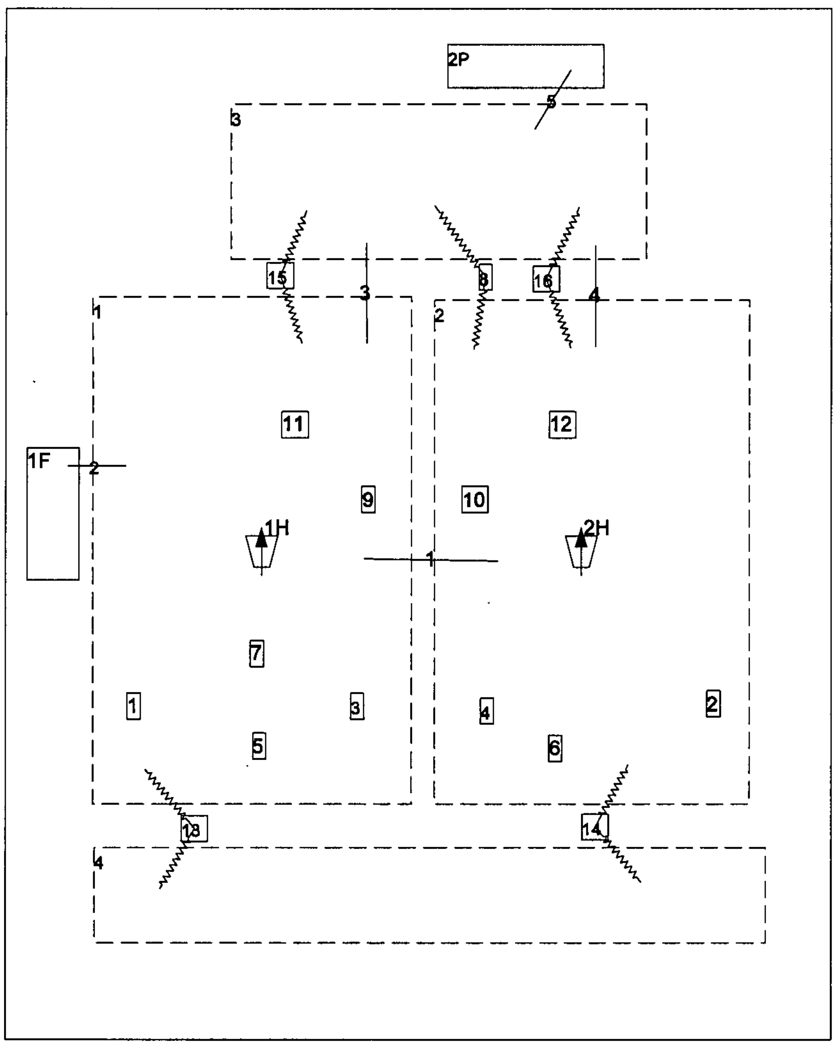


Figure 4.10-1 GOTHIC Compartment Model for the Main Steam Tunnel Temperature Analysis

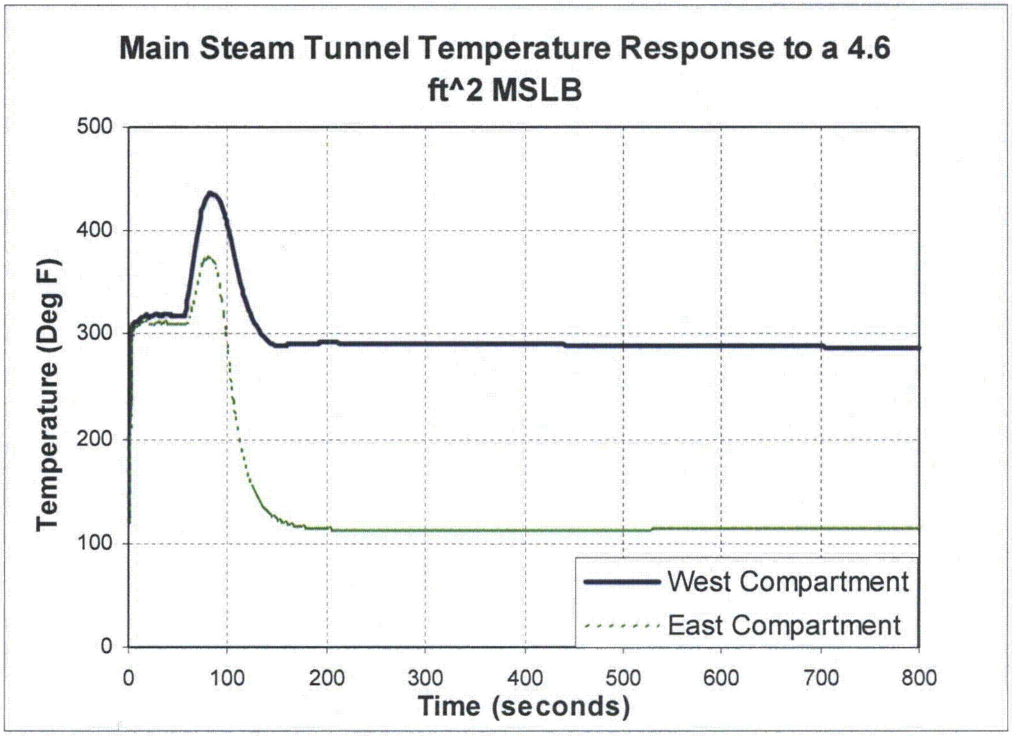


Figure 4.10-2 Main Steam Tunnel Temperature Transient for a 4.6 ft^2 Steam Line Rupture

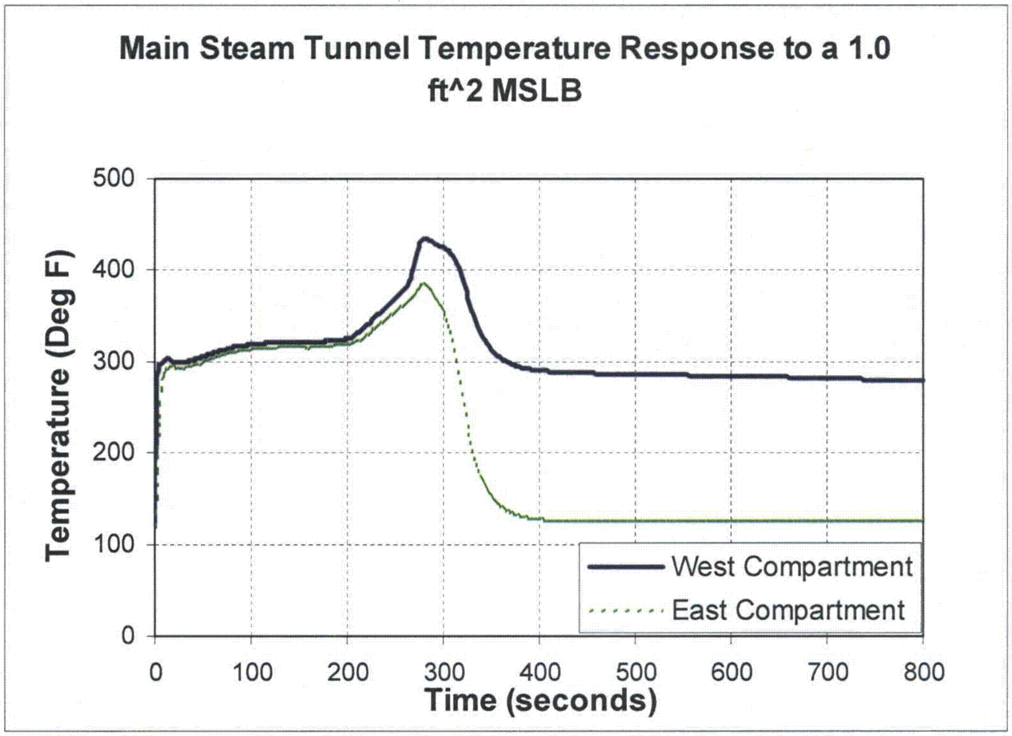


Figure 4.10-3 Main Steam Tunnel Temperature Transient for a 1.0 ft^2 Steam Line Rupture

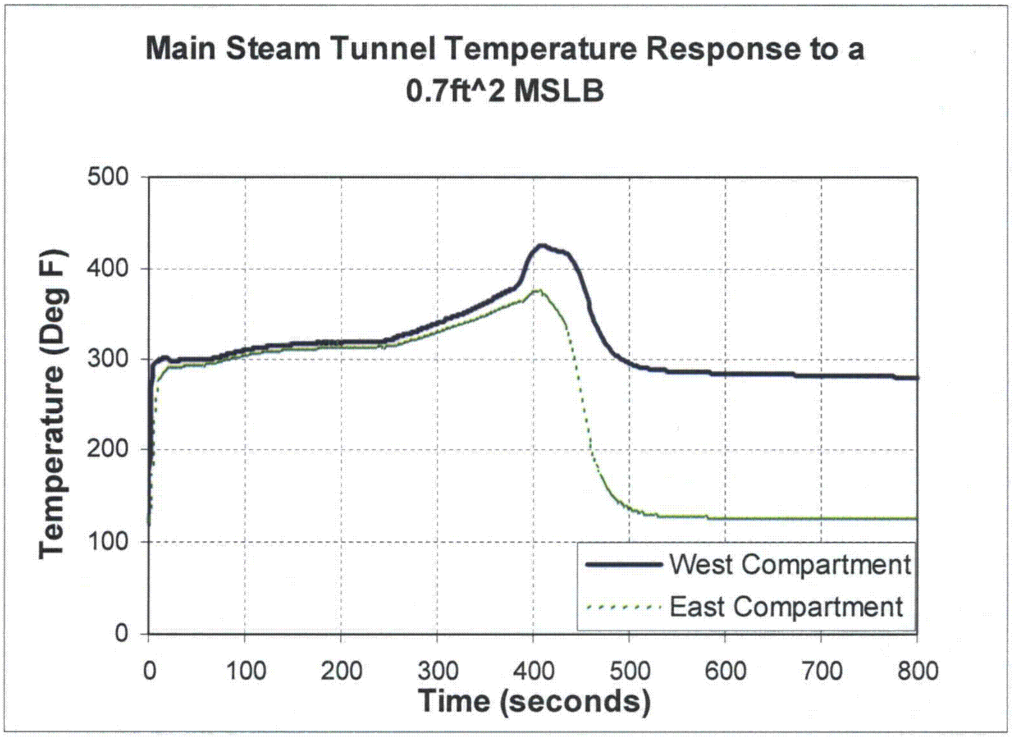


Figure 4.10-4 Main Steam Tunnel Temperature Transient for a 0.7 ft^2 Steam Line Rupture

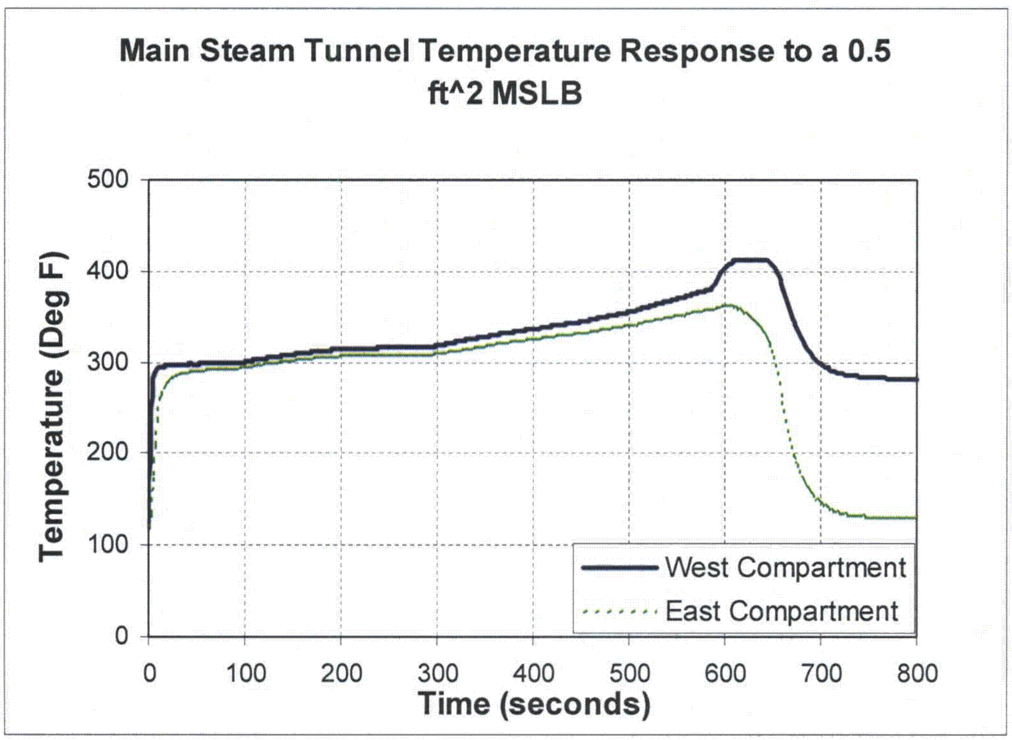


Figure 4.10-5 Main Steam Tunnel Temperature Transient for a 0.5 ft^2 Steam Line Rupture

4.11 Radiological Consequences of a Postulated SGTR (USAR Section 15.6.3)

Introduction

The evaluation of the radiological consequences due to a postulated SGTR assumes a complete severance of a single SG tube while the reactor is operating at full rated power and a coincident loss of offsite power. Occurrence of the accident leads to an increase in contamination of the secondary system due to reactor coolant leakage through the tube break. A reactor trip occurs automatically, as a result of Low Pressurizer Pressure. The reactor trip will automatically trip the turbine. The resulting sharp increase in radioactivity in the secondary system will be detected by radiation monitors which will automatically terminate SG blowdown. The assumed coincident loss of offsite power will cause closure of the steam dump valves to protect the condenser. The SG pressure will then increase rapidly, resulting in steam discharge as well as activity release through the SG safety and relief valves. Venting from the affected SG, i.e., the SG which experiences the tube rupture, will continue until the secondary system pressure is below the SG safety valve setpoint. At this time, the affected SG is effectively isolated and, thereafter, no steam or activity is assumed to be released from the affected SG. The remaining unaffected SGs remove core decay heat by venting steam through the ARVs until the controlled cooldown is terminated.

The increase in MFIV closure time associated with the MSIV and MFIV and associated actuator replacement affects the analysis for the SGTR accident by introducing additional feedwater into the faulted SG. As a result, the faulted SG will be overfilled at an earlier time and will increase the total radioactive releases to the atmosphere. Furthermore, the SGTR has been reanalyzed to incorporate the revision to the operator response times, in order to reflect the latest operator performance measured from simulator exercises, based upon the updated operating procedures that emphasize three-way communication. Therefore, the radiological consequences of a postulated SGTR are calculated and presented in this section.

Assumptions and Methods

The analysis of the radiological consequences of an SGTR considers the most severe release of secondary activity, as well as reactor activity leaked from the tube break. The inventory of iodine and noble gas fission product activity available for release to the environment depends on the primary-to-secondary coolant leakage rate, the percentage of defective fuel in the core, and the mass of steam discharged to the environment.

The reanalysis of the radiological consequences of a postulated SGTR uses the guidelines provided in Regulatory Guide 1.195 for assumptions and methods to calculate the offsite dose consequences. This involved changing three elements of the analysis of record methodology:

1. The reanalysis uses thyroid dose conversion factors for inhalation of radioactive material based on the data provided in Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose conversion Factors for Inhalation, Submersion, and Ingestion." This is a departure from the more conservative dose conversion factor values listed in Regulatory Guide 1.109, which were used in the current analysis of record,

2. The effective dose conversion factors, provided in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil," are used to calculate the whole body doses. This deviates from the current analysis of record,

which used the more conservative dose conversion factor values listed in Regulatory Guide 1.109, and

3. The reanalysis uses a factor of 335 for the accident initiated iodine spike release rate, which is a departure from the more conservative factor of 500 modeled in the current analysis of record.

These methodology changes reduce the magnitude of the accident source term and result in lower doses than would be obtained using the methodology currently presented in the SGTR with overfill analysis of record. However, the Regulatory Guide 1.195 source term methodology is recognized by the nuclear industry as having a better scientific basis and it has been approved by the NRC for use by other licensees.

The initial or equilibrium iodine activities in the primary coolant and SG water are assumed to correspond to their TS limits (TS 3.4.16, "RCS Specific Activity," and 3.7.18, "Secondary Specific Activity") of 1.0 and 0.1 μCi/gm DOSE EQUIVALENT I-131, respectively. Based on the thyroid dose conversion factors obtained from EPA Federal Guidance Report No. 11, the individual iodine isotopic initial concentrations are recalculated, assuming the spectral distribution for these isotopes to be the same as the maximum coolant activities corresponding to 1% failed fuel.

Noble gas concentrations in the primary and secondary coolant are assumed to correspond to 1% failed fuel. The SG secondary water has negligible noble gas activity. All the noble gas activity that leaks from the primary to the secondary is assumed to be transported instantaneously to the environment.

The equilibrium/initial activities in the primary and secondary are listed in Table 4.11-1.

Iodine spiking effects are considered, in accordance with Regulatory Guide 1.195. For a pre-accident iodine spike case, the initial reactor coolant iodine activity is assumed to have been raised to the maximum value (60.0 μCi/gm DOSE EQUIVALENT I-131), permitted by TS 3.4.16 at full power operations, since a reactor transient has occurred prior to the postulated SGTR. For the accident induced iodine spike case, the iodine release rate from the fuel rods to the primary coolant is assumed to increase to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (1.0 μCi/gm DOSE EQUIVALENT I-131) specified in the TSs. This increased escape rate is assumed to last 8 hours.

The following assumptions and parameters are used to calculate the activity released and the associated offsite dose following an SGTR:

1. The amount of break flow discharged from the primary-to-secondary, and releases to the environment from the affected SG and the intact SGs are obtained from the RETRAN analysis, presented in Section 4.5.

2. It is assumed that all of the iodine in the fraction of reactor coolant that flashes to steam upon reaching the secondary side is released to the steam phase. No credit is taken for scrubbing.

3. A 1 gpm primary-to-secondary leak is assumed to occur to the unaffected SGs, throughout the accident sequence.

4. All noble gas activity in the reactor coolant that is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is released to the atmosphere.

5. The iodine partition fraction between the liquid and steam in the SG is assumed to be 0.01.

6. The steam releases were assumed to continue from the intact SGs for a period of 8 hours until the RHR System is placed into service.

7. Radioactive decay prior to the release of activity is considered. No decay during transit or ground deposition is considered.

Results

Consistent with the current methodology for evaluating radiological consequences of a postulated SGTR, along with the changes mentioned above, the offsite (Exclusion Area Boundary and Low Population Zone) doses to the thyroid and whole body have been calculated, using the RADTRAD 3.03 computer code (Reference 25). The results of the dose calculations are presented in Table 4.11-2. As can be seen from this table, the calculated radiological consequences of a postulated SG tube failure accident do not exceed: (1) the exposure guidelines as set forth in 10 CFR Part 100, Section 11 for the accident with an assumed pre-accident iodine spike, and (2) 10% of these exposure guidelines, for the accident with an equilibrium iodine concentration in combination with an assumed accident initiated iodine spike.

Table 4.11-1. Initial Primary and Secondary Coolant Activity (μCi/gm)

Isotope	Primary	SG Water
I-131	0.7518	0.07518
I-132	0.8431	0.08431
I-133	1.3237	0.13237
I-134	0.2019	0.02019
I-135	0.7733	0.07733
Xe-131m	3.41	1.07E-04
Xe-133m	5.37	1.73E-04
Xe-133	290.0	9.12E-03
Xe-135m	0.604	8.62E-05
Xe-135	9.82	3.20E-04
Xe-137	0.224	6.90E-06
Xe-138	0.815	2.55E-05
Kr-83m	0.554	1.92E-05
Kr-85m	2.26	7.10E-05
Kr-85	9.41	2.96E-04
Kr-87	1.47	4.62E-05
Kr-88	4.26	1.34E-04
Kr-89	0.121	3.71E-06

Table 4.11-2 Radiological Consequences of a SGTR Accident

Pre-Accident Iodine Spike				
	Calculated Dose (rem)			Limit (rem)
Location	**Dose Type**	**Overfill Scenario**	**Stuck-Open ARV Scenario**	
EAB	Thyroid	51.769	48.583	300
	Whole-Body	0.226	0.153	25
LPZ	Thyroid	7.192	6.564	300
	Whole-Body	0.034	0.021	25
Concurrent Iodine Spike				
	Calculated Dose (rem)			Limit (rem)
Location	**Dose Type**	**Overfill Scenario**	**Stuck-Open ARV Scenario**	
EAB	Thyroid	22.797	15.966	30
	Whole-Body	0.132	0.083	2.5
LPZ	Thyroid	4.825	2.646	30
	Whole-Body	0.025	0.013	2.5

4.12 Issue-specific Analyses

The following accident analysis issues are not presented in the USAR, but were evaluated to determine the impact of the proposed TS change.

Steam Line Rupture – Full Power Core Response

The Westinghouse methodology for analyzing the reactor core response to excessive secondary steam releases is documented in topical report WCAP-9226 (Reference 23). This WCAP, known as the "Steam Line Break Topical," examined the effect of power level (including full power cases), break size, and plant variations for typical three-loop and four-loop Westinghouse designed pressurized water reactors. This WCAP concludes that the largest double-ended steam line rupture at end-of-life, hot zero power (MODE 2) conditions, with the most reactive RCCA in the fully withdrawn position, bounds all other power levels and other MODES for the post-trip phase of the transient.

The conclusions of WCAP-9226 are based on at-power steam line break analyses that credit specific protection system performance characteristics. However, no explicit limits of applicability for the conclusion that the hot zero power case is the limiting case were explicitly defined in WCAP-9226. Westinghouse recognizes that limits of applicability do exist and, during a recent review of the at-power steam line break analyses supporting WCAP-9226, has defined and internally documented these limitations. Since the WCAP was first issued, plant modifications have been made which might not be bounded by the generic assumptions of the WCAP, such as increases in the OPΔT response times, lead/lag time constant changes, reduced Low Steam Line Pressure setpoints, etc. As a result, the full power steam line break event, which credits the OPΔT reactor trip function, could be initiated from the plant conditions that are outside the limits of applicability.

Westinghouse has defined explicit limits of applicability (Reference 24) for portions of the analyses for the steam line break event that support the conclusions in WCAP-9226. This topical report concludes that the steam line break performed at hot zero power conditions is a limiting and sufficiently conservative licensing basis to demonstrate that the 10 CFR 100 criteria is met.

WCNOC uses the Westinghouse methodology (Steam Line Break Topical and Safety Analysis Standards) to perform the WCGS analyses, and therefore, may be impacted by this issue. Thus, a plant specific at-power steam line break analysis was performed to supplement the current licensing basis hot zero power case. The purpose of this analysis was to ensure that the guidance, methodology, and conclusions documented in WCAP-9226 are valid, even with assumed parameters being outside the generic limits established by Westinghouse.

This event was primarily analyzed to demonstrate the adequacy of the protection systems. This is demonstrated by showing that with the appropriate actions of these systems, the DNB design basis is satisfied and fuel centerline melting is precluded.

A range of break sizes were analyzed, ranging from small breaks (starting at 0.1 ft^2), which typically do not result in a reactor trip, through intermediate break sizes which typically trip on the OPΔT reactor trip, up to break sizes large enough (usually around 1.4 ft^2) to initiate a reactor trip on a SI signal. The most limiting break size is the largest break case that results in a reactor trip as a result of the OPΔT reactor trip function.

The steamline rupture - full power core response analysis is analyzed with the LOFTRAN code (Reference 14)

Results

Table 4.12-1 presents the time sequence of events for the limiting break size (0.75 ft^2). Transient response time histories for the following parameters: nuclear power, core heat flux, pressurizer pressure, pressurizer water volume, core inlet temperature (intact and faulted loops), SG pressure (intact and faulted loops), and SG mass flowrate are shown in Figure 4.12-1 through 4.12-7 for this limiting break. Figure 4.12-8 shows the result of the DNBR calculation.

Conclusions

The analysis shows the acceptance criteria applicable to this event are satisfied. The DNBR safety limit is met, and therefore, there is no melting at the fuel centerline.

Note that MSIVs remain open until the Low Steam Line Pressure or the Containment Pressure High-2 setpoint is reached, however, in a limiting steam line break, this will not occur during the time of interest. As such, MSIV closure time would have no effect on the results.

Table 4.12-1 Time Sequence of Events for Full Power Steam Line Rupture

Event	Time (sec)
Steam line rupture	0.0
OPΔT reactor trip setpoint reached	15.38
Rods begins to drop	17.38
Minimum DNBR occurs	18.60
Maximum core heat flux occurs	18.60

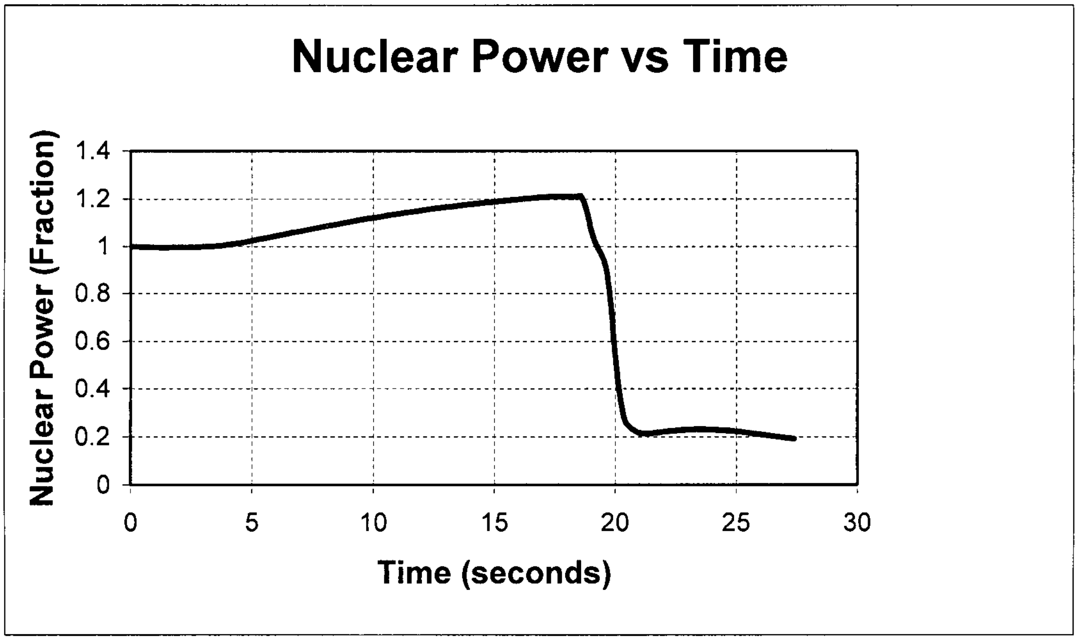


Figure 4.12-1 Nuclear Power versus Time

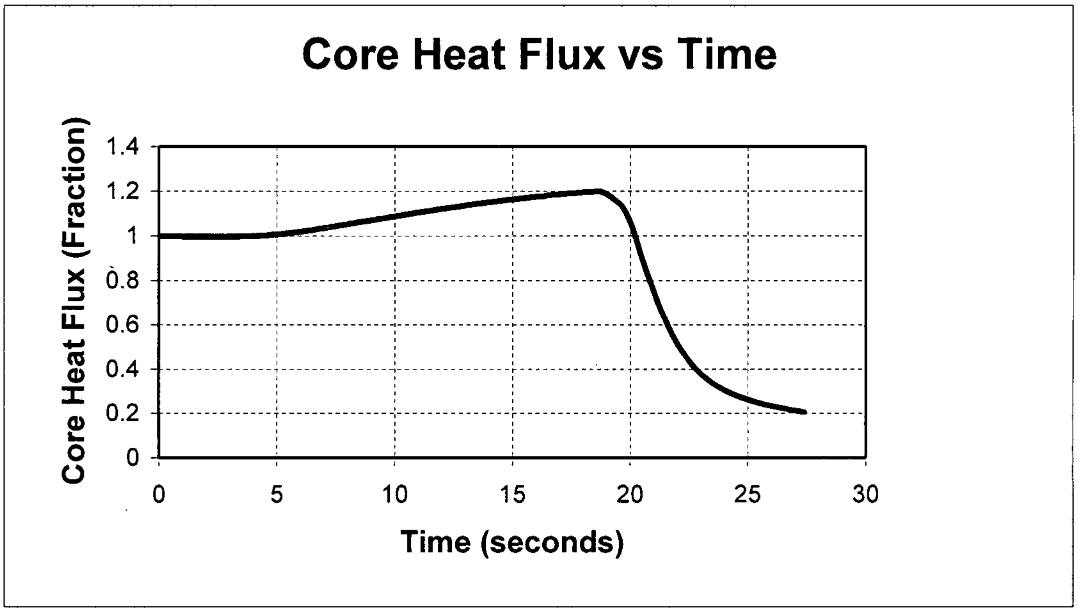


Figure 4.12-2 Core Heat Flux versus Time

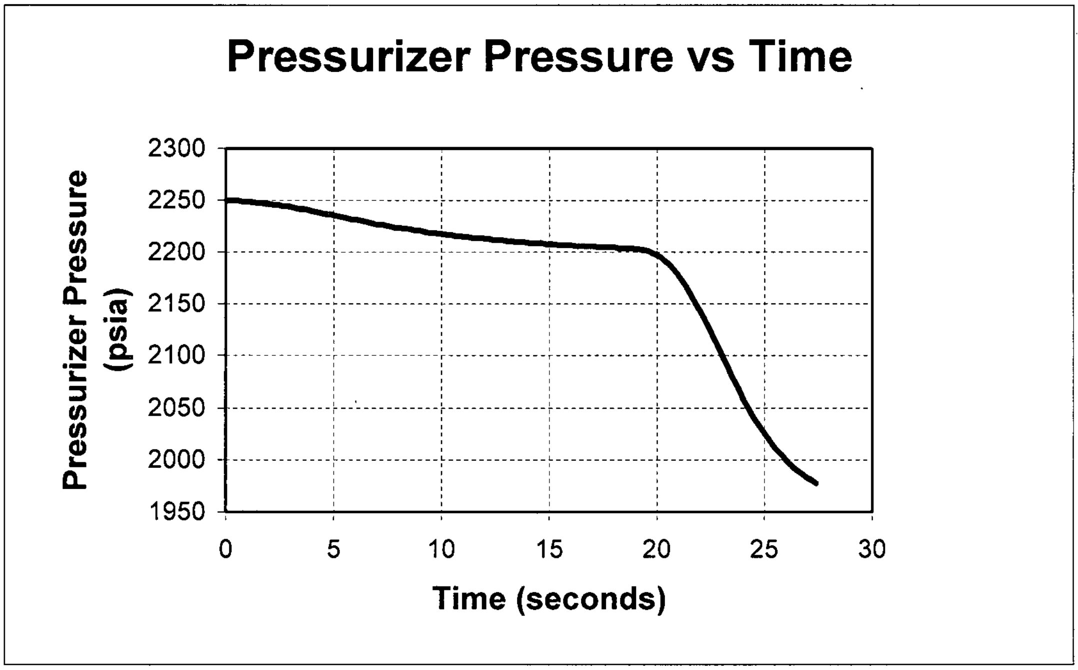


Figure 4.12-3 Pressurizer Pressure versus Time

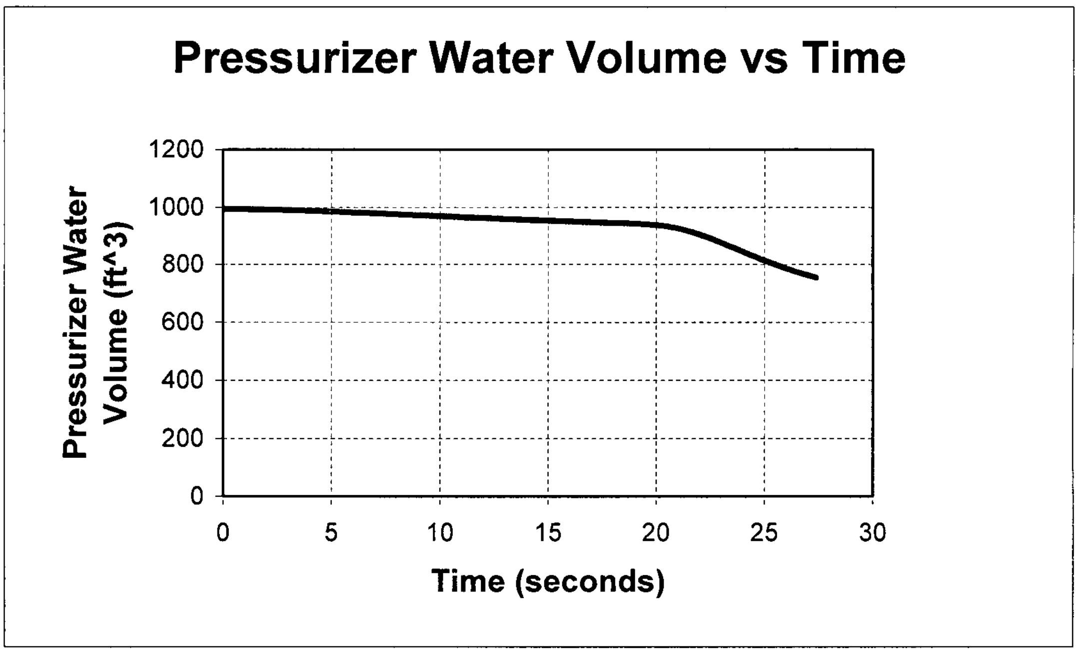


Figure 4.12-4 Pressurizer Water Volume versus Time

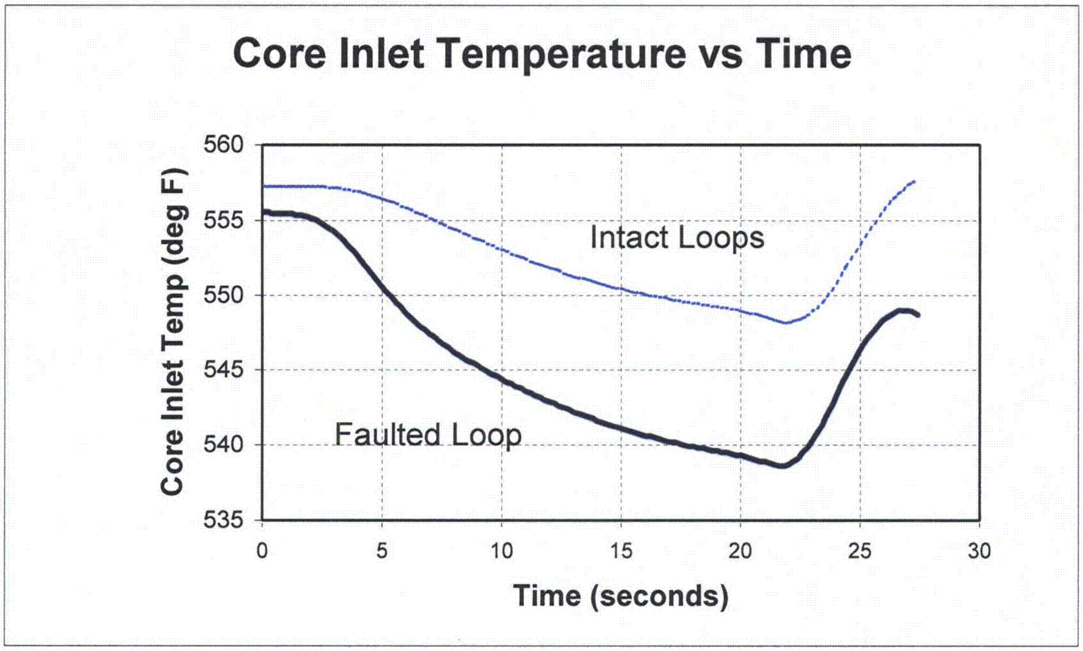


Figure 4.12-5 Core Inlet Temperature (Intact and Faulted Loops) versus Time

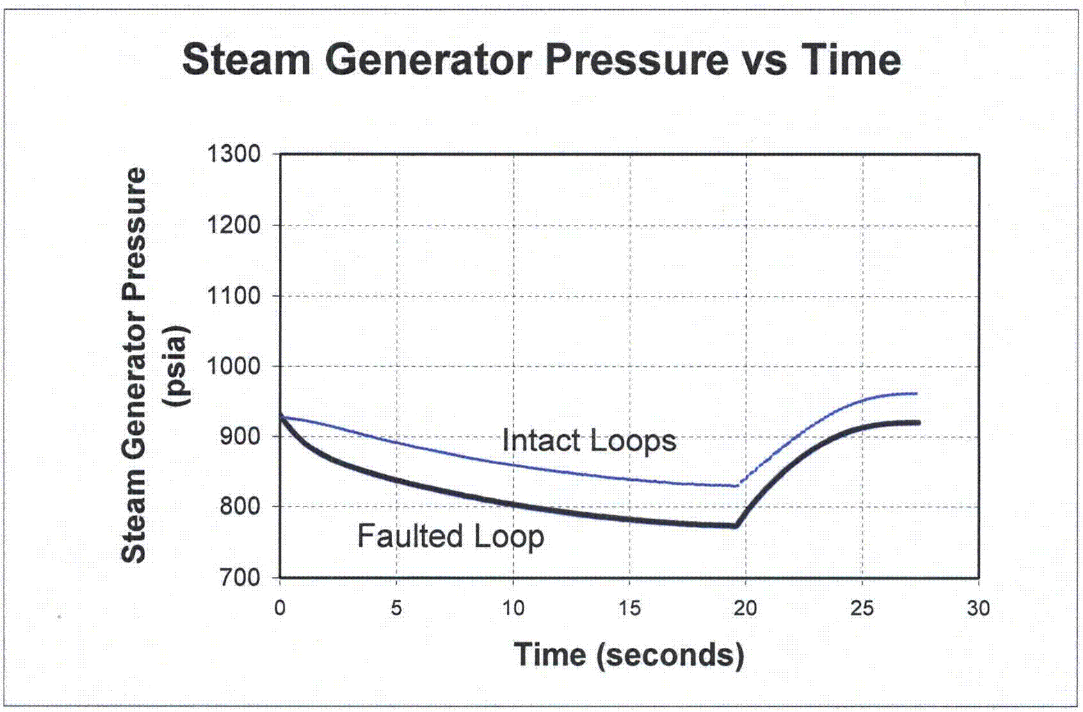


Figure 4.12-6 SG Pressure (Intact and Faulted Loops) versus Time

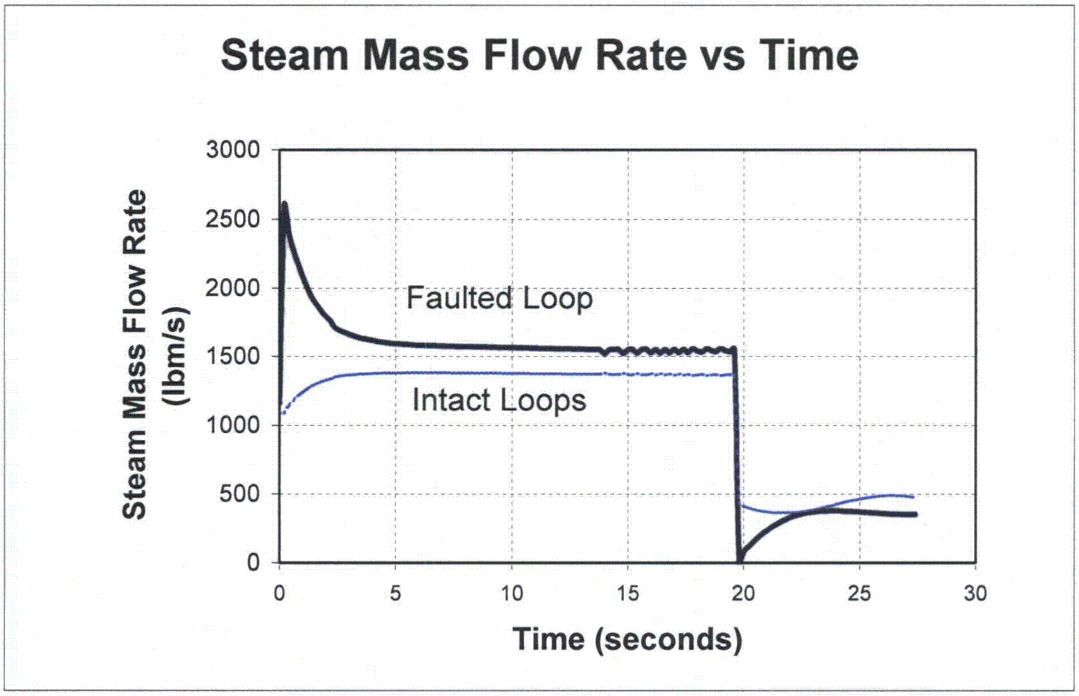


Figure 4.12-7 SG Mass Flow Rate (Intact and Faulted Loops)

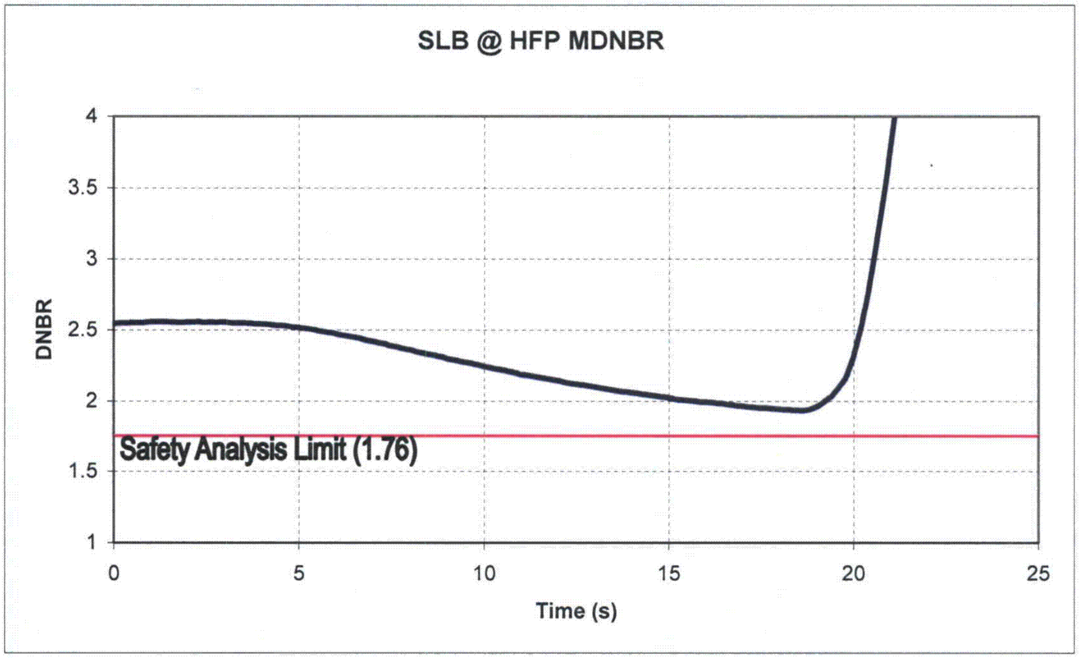


Figure 4.12-8 DNBR versus Time

Steam Line Break at Power Coincident with Rod Withdrawal

The steam line break at power coincident with rod withdrawal is analyzed in response to NRC IE Information Notice No. 79-22, "Qualification of Control System." This notice identified a potential unreviewed safety question concerning non-safety grade equipment, such as the automatic Rod Control System, being subject to an adverse environment from high energy line breaks inside or outside containment.

The postulated accident scenario includes the failure of the Rod Control System as the result of the environment created by a steam line rupture. The analysis assumes the control rod withdrawal occurs at the initiation of the transient. The steam line break causes increased heat removal and a subsequent decrease in primary pressure concurrent with an increase in reactor power and heat flux due to the rod withdrawal. Protection to mitigate the consequences of this event is available from the OPΔT reactor trip signal or the SI signal reactor trip signal (Low Steam Line Pressure). The power and heat flux increases result in a reactor trip on the OPΔT signal. The reactor trip stops the positive reactivity from the withdrawing RCCAs, and the insertion of the control rods provides negative reactivity. After reactor trip, RCS conditions are similar to those of the post-reactor trip portion of a steam line break event, without coincidental rod withdrawal from full power.

The key analysis result is minimum DNBR. In terms of minimum DNBR, the most limiting part of the transient occurs immediately before reactor trip. Therefore, the increase in the MSIV and MFIV closure time would not perceptibly affect the calculated limiting DNBR as the actuations of steam line isolation and feed line isolation occur after the limiting DNBR is reached during rod motion from the reactor trip.

5.0 REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements/Criteria

One main steam isolation valve (MSIV) is installed in each of the four main steam lines outside the containment and downstream of the main steam safety valves. The MSIVs prevent uncontrolled blowdown from more than one steam generator (SG) in the event of a postulated design basis accident. The portion of the Main Feedwater System from the SG to the main feedwater isolation valves (MFIVs) is safety related and is required to function following a design basis accident and to achieve and maintain the plant in a post accident safe shutdown condition. The portion of the Condensate and Feedwater System from the SG to the MFIVs is safety related and is required to function following a design basis accident and to achieve and maintain the plant in a post accident safe shutdown condition.

10 CFR 50, Appendix A, General Design Criteria (GDC) 4, "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that

the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

GDC 10, Reactor design. "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition or normal operations, including the effects of anticipated operational occurrences."

GDC 15, Reactor coolant system design. "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

GDC 16, Containment design. "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

GDC 26, Reactivity control system redundancy and capability. "Two independent reactivity control systems of different design principles shall be provided. ....."

GDC 27, Combined reactivity control systems capability. "The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained."

GDC 28, Reactivity limits. "The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core."

GDC 31, Fracture prevention of reactor coolant pressure boundary. "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized."

GDC 35, Emergency core cooling. "A system to provide abundant emergency core cooling shall be provided."

GDC 50, Containment design basis. "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be: designed so that the Containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience: and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.'"

10 CFR 50.36, Technical specifications, is the regulation that provides requirements regarding the content of technical specifications. Specifically, 10 CFR 50.36(c)(2)(ii) states that: "A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:....."

10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, requires an Emergency Core Cooling System designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to specific criteria for peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling.

10 CFR 100.11, Determination of exclusion area, low population zone, and population center distance, established requirements related to the mitigation of the radiological consequences of an accident.

Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," is NRC guidance for ensuring the adequacy of protection system actuation functions through periodic testing.

The replacement of the Main Steam and Feedwater Isolation System, MSIVs and MFIVs and associated actuators and resulting Technical Specification changes assist in assuring compliance with GDC 4 such that, in the event of a main steam line break or main feedwater line break inside containment, the containment will be appropriately isolated to prevent additional mass and energy from being delivered to the SGs. Compliance with GDC 16 and GDC 50 is maintained in that the increased MSIV and MFIV closure time will not result in exceeding the containment design pressure and violating the environmental qualification envelope of equipment required for mitigation of a design basis accident.

Periodic testing of the reactor trip and engineered safety feature actuation systems, as described in USAR Sections 7.2.2 and 7.3, complies with Regulatory Guide 1.22.

The proposed change affects Technical Specification (TS) 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)." These valves and the Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves have been determined to meet the criteria referred to by 10 CFR 50.36(c)(2)(ii). The proposed change incorporates the MFRVs and MFRV bypass valves into the TSs and specifies the valve isolation times outside of the TSs and into the TS Bases. The TS requirements continue to provide adequate assurance that the Main Steam and Main Feedwater System isolation valves are maintained OPERABLE and that the plant will be operated in a safe manner within the bounds of the applicable accident analyses.

Subsequent accident dose analyses for both steam generator tube rupture scenarios (SG overfill and stuck-open atmospheric relief valve) show that the radiological consequences resulting from a steam generator tube rupture accident remain well within the limiting values specified in 10 CFR Part 100 and Standard Review Plan, Section 15.6.3. A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming a simultaneous loss of offsite power.

The evaluations and analyses discussed in Section 4.0 indicated continued compliance with the requirements in GDC 10, GDC 15, GDC 26, GDC 27, GDC 28, GDC 31, and GDC 35.

Based on the considerations discussed above, 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the Commission's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

5.2 Significant Hazards Consideration

This license amendment request proposes to incorporate changes to TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)." This amendment application proposes to incorporate changes to these specifications based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. Revisions to TS 3.7.3 are made to add the Main Feedwater Regulating Valves (MFRVs) and their associated bypass valves.

Additionally, Surveillance Requirement (SR) 3.7.2.1 and SR 3.7.3.1 are revised to relocate the isolation time limits from the SRs to the TS Bases. This change is consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times."

The MSFIS consists of two independent actuation trains that monitor system inputs and by means of advanced logic matrices, drive actuation relays that energize or deenergize the solenoids required for the appropriate MSIV or MFIV operation. The modified MSFIS performs the same as the current design except that the system is comprised of advanced logic technology. The Solid State Protection System and Reactor Protection System inputs the ESFAS signals, the Main Control Board handswitches input to the MSFIS cabinets, and the same actuation output relays are utilized in the new design.

Wolf Creek Nuclear Operating Corporation (WCNOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," Part 50.92(c), as discussed below:

(1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

Evaluations and/or reanalysis assessing the impact of the replacement MSFIS, MSIVs and MFIVs and actuators, and the increased closure time on non-LOCA transients; SBLOCA transients; main steam line break mass and energy releases inside and outside containment; containment pressure and temperature response to a postulated main steam line break; environmental qualification of equipment; and the steam generator tube rupture transients and associated radiological consequences, were performed. The increase in closure times and the changes to the MSFIS, MSIVs, and MFIVs either do not provide an adverse impact or do not result in accident acceptance criteria being challenged.

The modifications to the MSFIS controls will not affect any design basis accidents since the logic which currently exists will continue to be performed. The replacement controls are functionally the same as the current system since the same logic functions are performed, the same inputs received, and the same outputs produced.

The replacement of the MSFIS controls, replacement of the MSIV and MFIVs, and replacement of the electro-hydraulic actuators with system-medium actuators will not result in a significant increase in the probability or consequence of an accident previously evaluated.

The relocation of the specific isolation times from the TSs to the TS Bases does not impact the design safety function of the valves to close. The TS requirements continue to provide the same level of assurance as before that the MSIVs and MFIVs are capable of performing their intended safety function. The addition of the MFRVs and MFRV bypass valves and extending the Completion Time for one or more MFIVs inoperable, is not an accident initiator and does not change the probability that an accident will occur. The increase in time that the MFIV is unavailable is small and the probability of an event occurring during this time period which would require isolation of the flow path is low. The redundancy provided by the MFRVs and MFRV bypass valves, which have the same actuation signals, provides adequate assurance that automatic feedwater isolation will occur.

Based on all of the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

(2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The increase in MSIV and MFIV closure time as a result of the replacement of the MSFIS controls, MSIVs and MFIVs and associated actuators, will not prevent the Main Steam System, Main Feedwater System, or Auxiliary Feedwater System from performing their safety functions. The increased closure time will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced with the proposed modifications and increased closure times. Although the modification does alter the design of the MSFIS and MSIV and MFIV actuators, it does not prevent the systems, subsystems, and components from performing their safety functions.

The relocation of the specific isolation times from the TSs to the TS Bases and the addition of the MFRVs and MFRV bypass valves and extending the Completion Time for one or more MFIVs inoperable does not affect the assumptions of any accident analysis or the OPERABILITY of plant equipment.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The replacement of the MSFIS controls, replacement of the MSIVs and MFIVs and associated actuators and resulting increased closure time, does not affect the manner in which safety limits or limiting safety system settings are determined, nor will there be any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no significant impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor, nuclear enthalpy rise hot channel factor, LOCA peak cladding temperature, peak local density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Conclusion:

Based on the above evaluation, WCNOC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 ENVIRONMENTAL CONSIDERATION

WCNOC has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. WCNOC has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7.0 REFERENCES

1. USAR Section 10.3.2.2.

2. USAR Section 10.4.7.2.2.

3. USAR Section 7.3.7.1.

4. Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, Revision 2, "Removal of Main Steam and Feedwater Valve Isolation Times."

5. Westinghouse NSAL-03-9, "Steam Generator Water Level Uncertainties," September 22, 2003.

6. WCAP-10079-P-A, "NOTRUMP - A Nodal Transient Small Break and General Network Code," Meyer, P. E., August 1985.

7. WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," Bordelon, F. M., et al., June 1974.

8. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46.

9. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," Lee, N., et al., August 1985.

10. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," Thompson, C. M., et al., July 1997.

11. WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture, " R.E. Land, September 1976.

12. Westinghouse Safety Analysis Standard (SAS) No.12.2, Rev. 8, "Mass and Energy Releases to Containment Following a Steamline Rupture," September 2000.

13. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

14. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-NP-A (Nonproprietary)," April 1984.

15. NAI 8907-02, Revision 17, "GOTHIC Containment Analysis Package User Manual," Version 7.2a(QA), January 2006.

16. NAI 8907-06, Revision 16, "GOTHIC Containment Analysis Package Technical Manual," Version 7.2a(QA), January 2006.

17. NRC letter from Anthony C. McMurtray (NRC) to Thomas Coutu (NMC), Enclosure 2, Safety Evaluation, September 29, 2003.

18. WCAP-10961-P, Revision 1, "Streamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup, J. C. Butler, and Love, D. S., October 1985.

19. IE Information Notice. 84-90: "Main Steam Line Break Effect on Environmental Qualification of Equipment," December 7, 1984.

20. Westinghouse letter SAP 97-116, "Steamline Break Mass and Energy Releases Outside CTMT for Wolf Creek Based on a Revised Shutdown Margin and Auxiliary Feedwater," May 29, 1997.

21. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

22. Westinghouse Safety Analysis Standard (SAS) No. 12.5, Rev. 1, "Mass and Energy Releases Following a Steamline Rupture, Superheated Blowdowns Outside Containment," June 1998.

23. WCAP-9226-P-A, Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," February 1998.

24. Westinghouse Issue Report 03-05-M034: WCAP-9226 Limits of Applicability, transmitted by letter SAP-03-043, April 10, 2003.

25. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," S. L. Humphreys et al., November 1998.

ATTACHMENT II

MARKUP OF TECHNICAL SPECIFICATION PAGES

TABLE OF CONTENTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. ~~One train inoperable.~~ Not Used.	~~-----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE.~~	
	~~H.1 Restore train to OPERABLE status.~~	~~24 hours~~
	~~OR~~	
	~~H.2 Be in MODE 3.~~	~~30 hours~~
I. One channel inoperable.	-----NOTE----- The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels.	
	I.1 Place channel in trip.	72 hours
	OR	
	I.2 Be in MODE 3.	78 hours

(continued)

Table 3.3.2-1 (page 2 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
3. Containment Isolation					
a. Phase A Isolation					
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.13	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
b. Phase B Isolation					
(1) Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3) Containment Pressure - High 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 28.3 psig
4. Steam Line Isolation					
a. Manual Initiation	1,2$^{(i)}$, 3$^{(i)}$	2	F	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2$^{(i)}$, 3$^{(i)}$	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
~~d.~~ d. Containment Pressure - High 2	1,2$^{(i)}$, 3$^{(i)}$	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 18.3 psig
c. Automatic Actuation Logic and Actuation Relays (MSFIS)	1,2$^{(i)}$, 3$^{(i)}$	2 trains	G	SR 3.3.2.3 SR 3.3.2.6	NA

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(i) Except when all MSIVs are closed.

Table 3.3.2-1 (page 3 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
4. Steam Line Isolation (continued)					
e. ~~d.~~ Steam Line Pressure					
(1) Low	1,2(i), 3(b)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 571 psig(c)
(2) Negative Rate - High	3(g)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 125(h) psi
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2(j), 3(j)	2 trains	~~H~~ G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.14	NA
b. Automatic Actuation Logic and Actuation Relays (MSFIS)	1, 2(k), 3(k)	2 trains	G	SR 3.3.2.3 SR 3.3.2.6	NA
c. ~~b.~~ SG Water Level - High High (P-14)	1,2(j)	4 per SG	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 79.7% of Narrow Range Instrument Span
d. ~~c.~~ Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) Above the P-11 (Pressurizer Pressure) Interlock and below P-11 unless the Function is blocked.
(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(g) Below the P-11 (Pressurizer Pressure) Interlock; however, may be blocked below P-11 when safety injection on low steam line pressure is not blocked.
(h) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(i) Except when all MSIVs are closed.
(j) Except when all MFIVs are closed ~~.~~ or and de-activated; or all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; or all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.

(k) Except when all MFIVs are closed and de-activated.

No changes this page - information only

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs and their associated actuator trains shall be OPERABLE.

APPLICABILITY: MODE 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV actuator train inoperable.	A.1 Restore MSIV actuator train to OPERABLE status.	7 days
B. Two MSIV actuator trains inoperable for different MSIVs when the inoperable actuator trains are not in the same separation group.	B.1 Restore one MSIV actuator train to OPERABLE status.	72 hours
C. Two MSIV actuator trains inoperable when the inoperable actuator trains are in the same separation group.	C.1 Restore one MSIV actuator train to OPERABLE status.	24 hours
D. Two actuator trains for one MSIV inoperable.	D.1 Declare the affected MSIV inoperable.	Immediately

(continued)

No changes this page - information only

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Three or more MSIV actuator trains inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A, B, or C not met.	E.1 Declare each affected MSIV inoperable.	Immediately
F. One MSIV inoperable in MODE 1.	F.1 Restore MSIV to OPERABLE status.	8 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 2.	6 hours
H. --------------NOTE-------------- Separate Condition entry is allowed for each MSIV. ----------------------------------- One or more MSIV inoperable in MODE 2 or 3.	H.1 Close MSIV. <u>AND</u> H.2 Verify MSIV is closed.	8 hours Once per 7 days
I. Required Action and associated Completion Time of Condition H not met.	I.1 Be in MODE 3. <u>AND</u> I.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------------------NOTE-------------------------------- Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MSIV is ~~≤ 5 seconds.~~ within limits.	In accordance with the Inservice Testing Program
SR 3.7.2.2	-------------------------------NOTE-------------------------------- Only required to be performed in MODES 1 and 2. -- Verify each actuator train actuates the MSIV to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

LCO 3.7.3 Four MFIVs and their associated actuator trains, four MFRVs and MFRV bypass valves shall be OPERABLE.

APPLICABILITY: MODE~~S~~ 1, ~~2, and 3.~~
MODES 2 and 3 except when:

a. MFIV is closed and de-activated; or
b. MFRV is closed and de-activated or closed and isolated by a closed manual valve; or
c. MFRV bypass valve is closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MFIV actuator train inoperable.	A.1	Restore MFIV actuator train to OPERABLE status.	7 days
B. Two MFIV actuator trains inoperable for different MFIVs when the inoperable actuator trains are not in the same separation group.	B.1	Restore one MFIV actuator train to OPERABLE status.	72 hours
C. Two MFIV actuator trains inoperable when the inoperable actuator trains are in the same separation group.	C.1	Restore one MFIV actuator train to OPERABLE status.	24 hours
D. Two actuator trains for one MFIV inoperable.	D.1	Declare the affected MFIV inoperable.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Three or more MFIV actuator trains inoperable. OR Required Action and associated Completion Time of Condition A, B, or C not met.	E.1 Declare each affected MFIV inoperable.	Immediately
F. --------------NOTE-------------- Separate Condition entry is allowed for each MFIV. ------------------------------------ One or more MFIVs inoperable.	F.1 Close MFIV. AND F.2 Verify MFIV is closed.	~~4~~ 72 hours Once per 7 days
INSERT 3.7-9 ~~G.~~ J. Required Action and associated Completion Time of Condition ~~F~~ G, H, or I not met.	~~G~~J.1 Be in MODE 3. AND ~~G~~J.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	------------------------------NOTE------------------------------ Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MFIV, MFRV and MFRV bypass valve is ~~≤ 5 seconds.~~ within limits.	In accordance with the Inservice Testing Program

(continued)

INSERT 3.7-9

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. --------------NOTE-------------- Separate Condition entry is allowed for each MFRV. ------------------------------------ One or more MFRVs inoperable.	G.1	Close or isolate MFRV.	72 hours
	AND		
	G.2	Verify MFRV is closed or isolated.	Once per 7 days
H. --------------NOTE-------------- Separate Condition entry is allowed for each MFRV bypass valve. ------------------------------------ One or more MFRV bypass valves inoperable.	H.1	Close or isolate MFRV bypass valve.	72 hours
	AND		
	H.2	Verify MFRV bypass valve is closed or isolated.	Once per 7 days
I. Two valves in the same flow path inoperable.	I.1	Isolate affected flow path.	8 hours

SURVEILLANCE REQUIRMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.3.2	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------------------------- Verify each actuator train actuates the MFIV to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.7.3.3	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------------------------- Verify each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

ATTACHMENT III

RETYPED TECHNICAL SPECIFICATION PAGES

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Not Used.		
I. One channel inoperable.	-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----	
	I.1 Place channel in trip.	72 hours
	OR	
	I.2 Be in MODE 3.	78 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
J. One Main Feedwater Pump trip channel inoperable.	--------------------NOTE-------------------- The inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels. --		
	J.1	Place channel in trip.	1 hour
	OR		
	J.2	Be in MODE 3.	7 hours
K. One channel inoperable.	--------------------NOTE-------------------- One additional channel may be tripped for up to 12 hours for surveillance testing. --		
	K.1	Place channel in bypass.	72 hours
	OR		
	K.2.1	Be in MODE 3.	78 hours
	AND		
	K.2.2	Be in MODE 5.	108 hours

(continued)

Table 3.3.2-1 (page 1 of 5)
Engineered Safety Feature Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
1.	Safety Injection					
a.	Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
b.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.13	NA
c.	Containment Pressure - High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 4.5 psig
d.	Pressurizer Pressure - Low	1,2,3(b)	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1820 psig
e.	Steam Line Pressure Low	1,2,3(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 571 psig(c)
2.	Containment Spray					
a.	Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
b.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c.	Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 28.3 psig

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) Above the P-11 (Pressurizer Pressure) interlock and below P-11 unless the Function is blocked.
(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

Table 3.3.2-1 (page 2 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
3. Containment Isolation					
a. Phase A Isolation					
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.13	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
b. Phase B Isolation					
(1) Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3) Containment Pressure - High 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 28.3 psig
4. Steam Line Isolation					
a. Manual Initiation	1,2(i), 3(i)	2	F	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2(i), 3(i)	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic and Actuation Relays (MSFIS)	1,2(i), 3(i)	2 trains	G	SR 3.3.2.3 SR 3.3.2.6	NA
d. Containment Pressure - High 2	1,2(i), 3(i)	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 18.3 psig

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(i) Except when all MSIVs are closed.

Table 3.3.2-1 (page 3 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
4. Steam Line Isolation (continued)					
e. Steam Line Pressure					
(1) Low	1,2(i), 3(b)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 571 psig(c)
(2) Negative Rate - High	3(g)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 125(h) psi
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2(j), 3(j)	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.14	NA
b. Automatic Actuation Logic and Actuation Relays (MSFIS)	1,2(k), 3(k)	2 trains	G	SR 3.3.2.3 SR 3.3.2.6	NA
c. SG Water Level -High High (P-14)	1,2(j)	4 per SG	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 79.7% of Narrow Range Instrument Span
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) Above the P-11 (Pressurizer Pressure) Interlock and below P-11 unless the Function is blocked.
(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(g) Below the P-11 (Pressurizer Pressure) Interlock; however, may be blocked below P-11 when safety injection on low steam line pressure is not blocked.
(h) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(i) Except when all MSIVs are closed.
(j) Except when all MFIVs are closed and de-activated; or all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; or all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.
(k) Except when all MFIVs are closed and de-activated.

Table 3.3.2-1 (page 4 of 5)
Engineered Safety Feature Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
6.	Auxiliary Feedwater					
a.	Manual Initiation	1,2,3	1 per pump	O	SR 3.3.2.8	NA
b.	Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c.	Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)	1,2,3	2 trains	N	SR 3.3.2.3	NA
d.	SG Water Level Low - Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 22.3% of Narrow Range Instrument Span
e.	Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f.	Loss of Offsite Power	1,2,3	2 trains	P	SR 3.3.2.7 SR 3.3.2.10	NA
g.	Trip of all Main Feedwater Pumps	1	2 per pump	J	SR 3.3.2.8	NA
h.	Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	M	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.10 SR 3.3.2.12	≥ 20.53 psia

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	-----------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MSIV is within limits.	In accordance with the Inservice Testing Program
SR 3.7.2.2	-----------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. -- Verify each actuator train actuates the MSIV to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

LCO 3.7.3 Four MFIVs and their associated actuator trains, four MFRVs and MFRV bypass valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when:

a. MFIV is closed and de-activated; or

b. MFRV is closed and de-activated or closed and isolated by a closed manual valve; or

c. MFRV bypass valve is closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MFIV actuator train inoperable.	A.1	Restore MFIV actuator train to OPERABLE status.	7 days
B. Two MFIV actuator trains inoperable for different MFIVs when the inoperable actuator trains are not in the same separation group.	B.1	Restore one MFIV actuator train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two MFIV actuator trains inoperable when the inoperable actuator trains are in the same separation group.	C.1	Restore one MFIV actuator train to OPERABLE status.	24 hours
D. Two actuator trains for one MFIV inoperable.	D.1	Declare the affected MFIV inoperable.	Immediately
E. Three or more MFIV actuator trains inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A, B, or C not met.	E.1	Declare each affected MFIV inoperable.	Immediately
F. --------------NOTE-------------- Separate Condition entry is allowed for each MFIV. ------------------------------------ One or more MFIVs inoperable.	F.1 <u>AND</u> F.2	Close MFIV. Verify MFIV is closed.	72 hours Once per 7 days
G. --------------NOTE-------------- Separate Condition entry is allowed for each MFRV. ------------------------------------ One or more MFRVs inoperable.	G.1 <u>AND</u> G.2	Close or isolate MFRV. Verify MFRV is closed or isolated.	72 hours Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. --------------NOTE-------------- Separate Condition entry is allowed for each MFRV bypass valve. ------------------------------------ One or more MFRV bypass valves is inoperable.	H.1 AND H.2	Close or isolate MFRV bypass valve. Verify MFRV bypass valve is closed or isolated.	72 hours Once per 7 days
I. Two valves in the same flow path inoperable.	I.1	Isolate affected flow path.	8 hours
J. Required Action and associated Completion Time of Condition F, G, H or I not met.	J.1 AND J.2	Be in MODE 3. Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------------------NOTE--------------------------------- Only required to be performed in MODES 1 and 2. -- Verify the isolation time of each MFIV, MFRV and MFRV bypass valve is within limits.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIRMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.3.2	-------------------------------NOTE------------------------------- Only required to be performed in MODES 1 and 2. --- Verify each actuator train actuates the MFIV to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.7.3.3	-------------------------------NOTE------------------------------- Only required to be performed in MODES 1 and 2. --- Verify each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Four ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARV line inoperable for reasons other than excessive leakage.	A.1 Restore required ARV line to OPERABLE status.	7 days
B. Two ARV lines inoperable for reasons other than excessive leakage.	B.1 Restore all but one required ARV line to OPERABLE status.	72 hours
C. Three or more ARV lines inoperable for reasons other than excessive leakage.	C.1 Restore all but two ARV lines to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. With one or more of the ARVs inoperable because of excessive seat leakage.	D.1	Initiate action to close the associated block valve(s).	Immediately
	AND		
	D.2	Restore ARV(s) to OPERABLE staus.	30 days
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ARV.	In accordance with the Inservice Testing Program
SR 3.7.4.2	Verify one complete cycle of each ARV block valve.	18 months

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
LCO 3.0.4b. is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1	Restore steam supply to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable for reasons other than Condition A.	B.1	Restore AFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met. OR Two AFW trains inoperable.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	6 hours 12 hours
D. Three AFW trains inoperable.	D.1	--------------NOTE------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ----------------------------------- Initiate action to restore one AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	-------------------------------NOTE-------------------------------- Not required to be performed for the AFW flow control valves until the system is placed in standby or THERMAL POWER is > 10% RTP. --- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.2	------------------------------NOTE------------------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Test Program
SR 3.7.5.3	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	------------------------------NOTE------------------------------ Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. -- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2 whenever unit has been in MODE 5 or 6 for > 30 days

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST contained water volume shall be ≥ 281,000 gal.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST contained water volume not within limit.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter
	AND	
	A.2 Restore CST contained water volume to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST contained water volume is ≥ 281,000 gal.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	--------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. -----------------------------------	
		Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------------------NOTE------------------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water (ESW) System

LCO 3.7.8 Two ESW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ESW train inoperable.	A.1	-------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by ESW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ESW System. ----------------------------------- Restore ESW train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	-------------------------------NOTE-------------------------------- Isolation of ESW System flow to individual components does not render the ESW System inoperable. --- Verify each ESW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.8.2	Verify each ESW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.8.3	Verify each ESW pump starts automatically on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Plant inlet water temperature of UHS not within limit.	A.1	Verify water level of main cooling lake ≥ 1075 ft. mean sea level.	1 hour <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>		
	A.2	Verify plant inlet water temperature of UHS is ≤ 94°F.	Once per hour
B. Required Action and associated Completion Time not met. <u>OR</u> UHS inoperable for reasons other than Condition A.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.9.1	Verify water level of UHS is ≥ 1070 ft mean sea level.	24 hours
SR 3.7.9.2	Verify plant inlet water temperature of UHS is ≤ 90°F.	24 hours

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

--NOTE--
The control room boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
B. Two CREVS trains inoperable due to inoperable control room boundary in MODES 1, 2, 3, and 4.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1.1	Place OPERABLE CREVS train in CRVIS mode.	Immediately
	<u>AND</u>		
	D.1.2	Verify OPERABLE CREVS train is capable of being powered by an emergency power source.	Immediately
	<u>OR</u>		
	D.2.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	E.2	Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREVS train pressurization filter unit for ≥ 10 continuous hours with the heaters operating and each CREVS train filtration filter unit for ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CREVS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one CREVS train can maintain a positive pressure of ≥ 0.25 inches water gauge, relative to the outside atmosphere during the CRVIS mode of operation.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning System (CRACS)

LCO 3.7.11 Two CRACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRACS train inoperable.	A.1 Restore CRACS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies .	C.1.1 Place OPERABLE CRACS train in operation.	Immediately
	AND	
	C.1.2 Verify OPERABLE CRACS train is capable of being powered by an emergency power source.	Immediately
	OR	
	C.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
D. Two CRACS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D.2 Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CRACS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CRACS train has the capability to remove the assumed heat load.	18 months

3.7 PLANT SYSTEMS

3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

NOT USED

3.7 PLANT SYSTEMS

3.7.13 Emergency Exhaust System (EES)

LCO 3.7.13 Two EES trains shall be OPERABLE.

---NOTE---
The auxiliary building or fuel building boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies in the fuel building.

---NOTE---
The SIS mode of operation is required only in MODES 1, 2, 3, and 4. The FBVIS mode of operation is required only during movement of irradiated fuel assemblies in the fuel building.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EES train inoperable in MODE 1, 2, 3, or 4.	A.1	Restore EES train to OPERABLE status.	7 days
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1	Restore auxiliary building boundary to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two EES trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. One EES train inoperable during movement of irradiated fuel assemblies in the fuel building.	D.1	Place OPERABLE EES train in operation in FBVIS mode.	Immediately
	OR		
	D.2	Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately
E. Two EES trains inoperable due to inoperable fuel building boundary during movement of irradiated fuel assemblies in the fuel building.	E.1	Restore fuel building boundary to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time of Condition E not met. OR Two EES trains inoperable during movement of irradiated fuel assemblies in the fuel building for reasons other than Condition E.	F.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each EES train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.13.2	Perform required EES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each EES train actuates on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.4	Verify one EES train can maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure in the auxiliary building during the SIS mode of operation.	18 months on a STAGGERED TEST BASIS
SR 3.7.13.5	Verify one EES train can maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure in the fuel building during the FBVIS mode of operation.	18 months on a STAGGERED TEST BASIS

3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

NOT USED

3.7 PLANT SYSTEMS

3.7.15 Fuel Storage Pool Water Level

LCO 3.7.15 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ----------------------------------- Suspend movement of irradiated fuel assemblies in the fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the fuel storage pool water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.16 Fuel Storage Pool Boron Concentration

LCO 3.7.16 The fuel storage pool boron concentration shall be ≥ 2165 ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. --	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	<u>AND</u>	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	<u>OR</u>	
	A.2.2 Verify by administrative means that a non-Region 1 fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.16.1	Verify the fuel storage pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 2 or 3 shall be within the Acceptable Domain of Figure 3.7.17-1 or in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 or 3 of the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -------------NOTE-------------- LCO 3.0.3 is not applicable. ------------------------------------ Initiate action to move the noncomplying fuel assembly to Region 1.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 or Specification 4.3.1.1.	Prior to storing the fuel assembly in Region 2 or 3

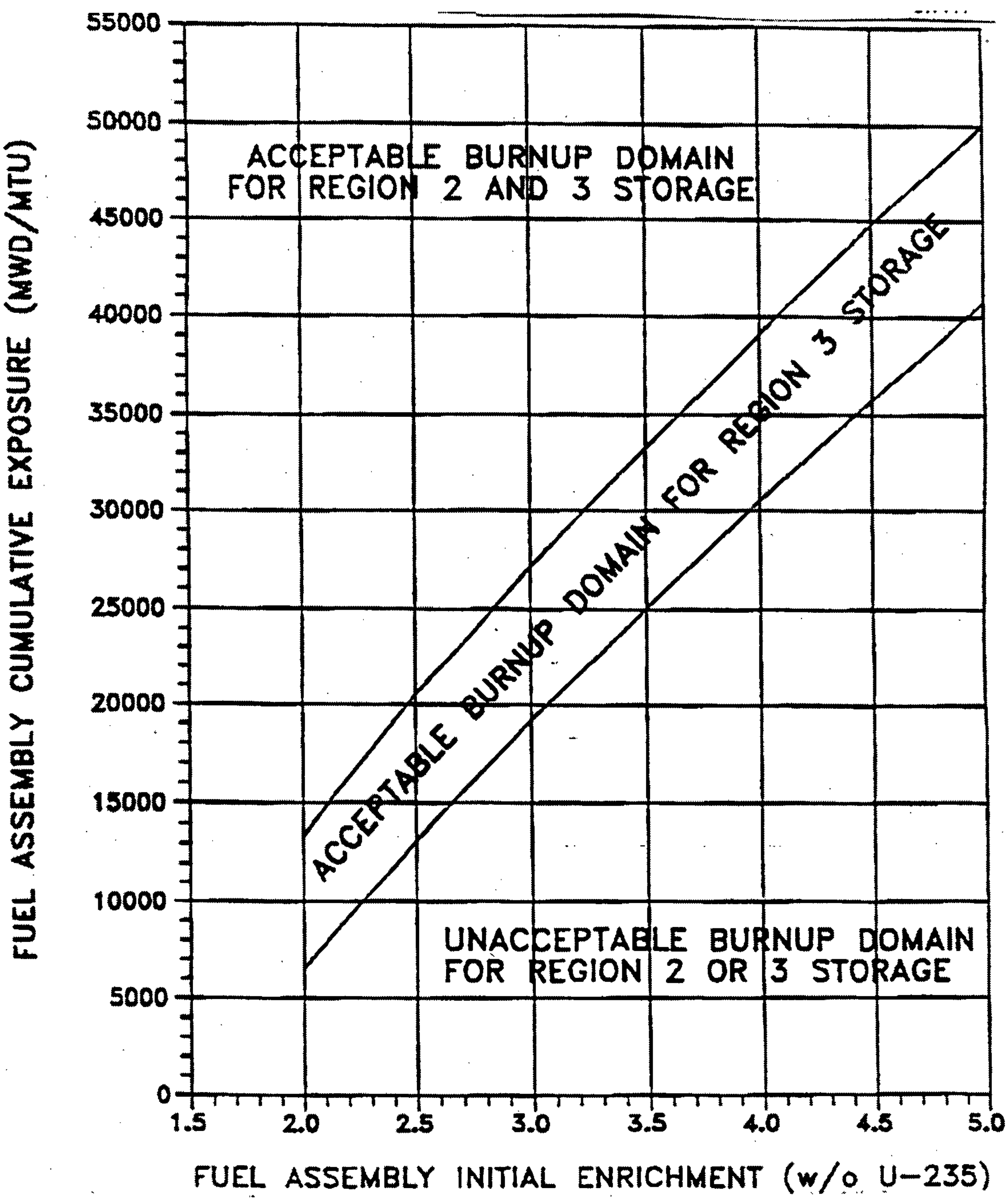


Figure 3.7.17-1 (page 1 of 1)
Minimum Required Fuel Assembly Burnup as a Function of Initial Enrichment to Permit Storage in Regions 2 and 3

3.7 PLANT SYSTEMS

3.7.18 Secondary Specific Activity

LCO 3.7.18 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

ATTACHMENT IV

PROPOSED TS BASES CHANGES
(for information only)

and Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

TABLE OF CONTENTS

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(3) Phase B Isolation - Containment Pressure (continued)

The basis for containment pressure MODE applicability and the Trip Setpoint are as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation (fast close) can be accomplished from the control room. There are two push buttons in the control room and either push button can initiate action to immediately close all MSIVs. The LCO requires two channels to be OPERABLE.

b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays (SSPS)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

INSERT B 3.3.2-19

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the

INSERT B 3.3.2-19

c. Steam Line Isolation – Automatic Actuation Logic and Actuation Relays (MSFIS)

The LCO requires two trains to be OPERABLE. The Steam Line Isolation signal from SSPS is provided to the Main Steam and Feedwater Isolation System (MSFIS) by four actuation signals per separation group. The Steam Line Isolation signals are provided by SSPS slave relays K634A and K634B. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the MSIVs.

4. Steam Line Isolation (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. ~~Steam Line Isolation - Automatic Actuation Logic and Actuation Relays (SSPS) (continued)~~

RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

~~c.~~ d. Steam Line Isolation - Containment Pressure - High 2

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure - High 2 provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties. The Trip Setpoint is ≤ 17.0 psig.

Containment Pressure - High 2 must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed. In MODE 4, the increase in containment pressure following a pipe break would occur over a relatively long time period such that manual actions could reasonably be expected to provide protection. In MODES 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure - High 2 setpoint.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~d~~ e.

Steam Line Isolation - Steam Line Pressure

(1) Steam Line Pressure - Low

Steam Line Pressure - Low provides closure of the MSIVs in the event of an SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure - Low was discussed previously under SI Function 1.e and the Trip Setpoint is the same.

Steam Line Pressure - Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11 unless blocked), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. If not blocked below P-11, the Steam Line Pressure - Low Function must be OPERABLE. When blocked, an inside containment SLB will be terminated by automatic actuation via Containment Pressure - High 2. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure - Negative Rate - High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have a significant effect on required plant equipment.

(2) Steam Line Pressure - Negative Rate - High

Steam Line Pressure - Negative Rate - High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(2) Steam Line Pressure - Negative Rate – High (continued)

limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure - Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure - Negative Rate - High signal is automatically enabled. Steam Line Pressure - Negative Rate - High control functions are isolated from the protective functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy requirements with a two-out-of-three logic.

Steam Line Pressure - Negative Rate - High must be OPERABLE in MODE 3 when the Steam Line Pressure - Low signal is blocked, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure - Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODE 3 unless all MSIVs are closed. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties. The Trip Setpoint is ≤ 100 psi with a rate /lag controller time constant ≥ 50 seconds.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high high setpoint and performs the following functions:

- Trips the main turbine;
- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

This Function is actuated by SG Water Level - High High, or by an SI signal. *The RTS also initiates a turbine trip signal whenever a* reactor trip (P-4) is generated. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was previously discussed.

a. Turbine Trip and Feedwater Isolation - Automatic Actuation Logic and Actuation Relays (SSPS)

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

INSERT B 3.3.2-23

~~b.~~ c. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand

INSERT B 3.3.2-23

b. Feedwater Isolation – Automatic Actuation and Logic and Actuation Relays (MSFIS)

Automatic Actuation Logic and Actuation Relays in the MSFIS consist of the same features and operate in the same manner as described for ESFAS Function 4.c.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~b~~ c.

Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14) (continued)

both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties. The Trip Setpoint is ≤ 78% of narrow range span.

~~e~~ d

Turbine Trip and Feedwater Isolation - Safety Injection

Turbine Trip and Feedwater Isolation are also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these initiation Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

INSERT B 3.3.2-24

~~Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODE 1 and in MODE 2 except when all MFIVs, are closed. In MODES 3, 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.~~

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available when reactor power is less than 2% power. The system has two motor driven pumps and a turbine driven pump, making it available during normal unit operation during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate

INSERT B 3.3.2-24

Turbine Trip and Feedwater Isolation Function 5.c, SG Water Level – High High must be OPERABLE in MODES 1 and 2 except when all MFIVs are closed and de-activated; or all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; or all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves. In MODES 3, 4, 5, and 6, Function 5.c is not required to be OPERABLE. The Automatic Actuation Logic and Actuation Relays (SSPS) Function must be OPERABLE in MODE 1, MODE 2 (except when all MFIVs are closed and de-activated; or all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; or all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves) and MODE 3 (except when all MFIVs are closed and de-activated; or all MFRVs are closed and de-activated or closed and isolated by a closed manual valve; or all MFRV bypass valves are closed and de-activated, or closed and isolated by a closed manual valve, or isolated by two closed manual valves). The Automatic Actuation Logic and Actuation Relays (MSFIS) Function must be OPERABLE in MODE 1, MODE 2 (except when all MFIVs are closed and de-activated), and MODE 3 (except when all MFIVs are closed and de-activated). In MODES 4, 5, and 6, the Automatic Actuation Logic and Actuation Relays (SSPS and MSFIS) are not required to be OPERABLE.

BASES

ACTIONS

H.1 ~~and H.2~~ Not Used.

~~Condition H applies to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation Function.~~

~~This action addresses the train orientation of the SSPS and the master and slave relays for this Function. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the following 6 hours. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference 12 and Reference 15. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. These Functions are no longer required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.~~

~~The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 7) assumption that 4 hours is the average time required to perform train surveillance.~~

I.1 and I.2

Condition I applies to:

- SG Water Level - High High (P-14);

If one channel is inoperable, 72 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-three logic will result in actuation. The 72 hour Completion Time is justified in Reference 12. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit to be placed in MODE 3 within the following 6 hours.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1 (continued)

that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.2.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 13.

SR 3.3.2.3

INSERT B 3.3.2-46

~~SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST using the BOP ESFAS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic. This test is required every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.~~

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but

INSERT B 3.3.2-46

SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST. The BOP ESFAS actuation logic is tested every 31 days on a STAGGERED TEST BASIS, using the BOP ESFAS automatic tester. The MSFIS actuation logic is tested every 31 days on a STAGGERED TEST BASIS, using the MSFIS automatic tester. The continuity check does not have to be performed, as explained in the Note. This SR is applied to the BOP actuation logic and MSFIS actuation logic that do not have circuits installed to perform the continuity check. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate based on industry operating experience, considering instrument reliability and operating history data.

TABLE B 3.3.2-1
(Page 1 of 2)

FUNCTION	TRIP SETPOINT[a]
1. Safety Injection	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Containment Pressure – High-1	≤ 3.5 psig
d. Pressurizer Pressure - Low	≥ 1830 psig
e. Steam Line Pressure - Low	≥ 615 psig
2. Containment Spray	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Containment Pressure - High-3	≤ 27.0 psig
3. Containment Isolation	
a. Phase A Isolation	
(1) Manual Initiation	N.A.
(2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
(3) Safety Injection	See Function 1 (Safety Injection)
b. Phase B Isolation	
(1) Manual Initiation	N.A.
(2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
(3) Containment Pressure - High-3	≤ 27.0 psig
4. Steam Line Isolation	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Automatic Actuation Logic and Actuation Relays (MSFIS)	N.A.
~~c.~~ d. Containment Pressure - High-2	≤ 17.0 psig
~~d.~~ e. Steam Line Pressure	
(1) Low	≥ 615 psig
(2) Negative Rate - High	≤ 100 psi

b. Automatic Actuation Logic and Actuation Relays (MSFIS) N.A.

TABLE B 3.3.2-1
(Page 2 of 2)

FUNCTION	TRIP SETPOINT[a]
5. Turbine Trip and Feedwater Isolation	
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. ~~b.~~ SG Water Level - High High	≤ 78% of narrow range instrument span
d. ~~c.~~ Safety Injection	See Function 1 (Safety Injection)
6. Auxiliary Feedwater	
a. Manual Initiation	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.
d. SG Water Level - Low-Low	≥ 23.5% of narrow range instrument span
e. Safety Injection	See Function 1 (Safety Injection)
f. Loss of Offsite Power	N.A.
g. Trip of all Main Feedwater Pumps	N.A.
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	≥ 21.60 psia
7. Automatic Switchover to Containment Sump	
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
b. Refueling Water Storage Tank (RWST) Level - Low Low	≥ 36% of instrument span
Coincident with Safety Injection	See Function 1 (Safety Injection)
8. ESFAS Interlocks	
a. Reactor Trip, P-4	N.A.
b. Pressurizer Pressure, P-11	≤ 1970 psig

(a) The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value.

Table B 3.3.2-2
(Page 2 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. <u>Steam Line Pressure - Low</u>	
a. Safety Injection (ECCS)	≤ 39[3]/27[4]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ ~~2~~ 2[5]
3) Phase "A" Isolation	≤ 2[5]
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60[1]
6) Containment Cooling	≤ 60[1]
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14[6]
9) Turbine Trip	N.A.
b. Steam Line Isolation	≤ 2[5]
5. <u>Containment Pressure - High-3</u>	
a. Containment Spray	≤ 32[1]/20[2]
b. Phase "B" Isolation	≤ 31.5
6. <u>Containment Pressure - High-2</u>	
Steam Line Isolation	≤ 2[5]
7. <u>Steam Line Pressure - Negative Rate-High</u>	
Steam Line Isolation	≤ 2[5]
8. <u>Steam Generator Water Level - High-High</u>	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ ~~2~~ 2[5]
9. <u>Steam Generator Water Level - Low-Low</u>	
a. Start Motor Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine Driven Auxiliary Feedwater Pumps	≤ 60
10. <u>Loss-of-Offsite Power</u>	
Start Turbine Driven Auxiliary Feedwater Pumps	≤ 60
11. <u>Trip of All Main Feedwater Pumps</u>	
Start Motor Driven Auxiliary Feedwater Pumps	N.A.

Table B 3.3.2-2
(Page 1 of 3)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Purge Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
I. Essential Service Water	N.A.
j Containment Cooling	N.A.
k. Control Room Isolation	N.A.
l. Reactor Trip	N.A.
m. Emergency Diesel Generators	N.A.
n. Component Cooling Water	N.A.
o. Turbine Trip	N.A.
2. Containment Pressure - High-1	
a. Safety Injection (ECCS)	≤ 29[7]/27[4]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ ~~[illegible]~~ 2[5]
3) Phase "A" Isolation	≤ 1.5[5]
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60[1]
6) Containment Cooling	≤ 60[1]
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14[6]
9) Turbine Trip	N.A.
3. Pressurizer Pressure - Low	
a. Safety Injection (ECCS)	≤ 29[7]/27[4]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ ~~[illegible]~~ 2[5]
3) Phase "A" Isolation	≤ 2[5]
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60[1]
6) Containment Cooling	≤ 60[1]
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14[6]
9) Turbine Trip	N.A.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The initial containment conditions are relatively unimportant parameters with respect to the containment pressure temperature analysis. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes developed to predict the containment pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. However, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 14.7 psia (0 psig). The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure results from a MSLB. The maximum containment pressure resulting from the worst case MSLB, 51.5 ~~48.9~~ psig, does not exceed the containment design pressure, 60 psig.

The containment was also designed for an external pressure load equivalent to -3.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 0 psig. This resulted in a minimum pressure inside containment of -2.72 psig, which is less than the design pressure.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the

BASES

APPLICABLE SAFETY ANALYSES (continued)

design basis analyses (Ref. 1) is 120°F. This resulted in a maximum containment air temperature of 386.5°F. The design temperature is 320°F.

360.0°F.

The spectrum of SLBs cases are used to establish the environmental qualification operating envelope for containment. The performance of required safety related equipment, including the containment structure itself, is evaluated against this operating envelope to ensure the equipment can perform its safety function. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature limit is also used in the containment external pressure analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 1).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a MSLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant accident temperature profile assures that the containment structural temperature will be maintained below the containment design temperature and that required safety related equipment within containment will continue to perform its function.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining

---

BASES

BACKGROUND

Containment Cooling System (continued)

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the ESW is an important factor in the heat removal capability of the fan units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regards to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable.

51.5

360.0°F

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 48.9 psig and the peak containment temperature is 386.5°F (experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level ranging to 102%, one containment spray train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -2.72 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators to the break.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators. system-medium

The MSIV is a 28-inch gate valve with ~~dual-redundant hydraulic~~ actuation trains. Either actuation train can independently perform the safety function to fast-close the MSIV on demand. ~~Each actuator train consists of a hydraulic accumulator controlled by solenoid valves on the associated MSIV.~~ For each MSIV, one actuator train is associated with separation group 4 ("yellow"), and one actuator train is associated with separation group 1 ("red").

The MSIVs close on a main steam isolation signal generated by low steam line pressure, high steam line negative pressure rate or High-2 containment pressure. The MSIVs fail as is on loss of control or actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the USAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the USAR, Section 6.2.1.4 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the USAR, Section 15.1.5 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the containment pressure analysis is the SLB inside containment, with initial reactor power at approximately 50% with loss of offsite power and the failure of one emergency diesel generator. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close,the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System. 0%

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSVIs isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting as far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that four MSIVs and their associated actuator trains be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

An MSIV actuator train is considered OPERABLE when it is capable of fast-closing the associated MSIV on demand and within the required isolation time. ~~This includes having adequate accumulator pressure to support fast-closure of the MSIV within the required isolation time and instrument air supply and pressure to the valve regulator is within limits.~~

INSERT B 3.7.2-3

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 due to significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function. The MSIV actuator trains must be OPERABLE in MODES 1, 2, and 3 to support operation of the MSIV.

In MODE 4, the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

INSERT B 3.7.2-3

The MSIVs are considered OPERABLE when isolation times are within the limits of Figure B 3.7.2-1 when given a fast close signal and they are capable of closing on an isolation actuation signal.

BASES

ACTIONS

The LCO specifies OPERABILITY requirements for the MSIVs as well as for their associated actuator trains. The Conditions and Required Actions for TS 3.7.2 separately address inoperability of the MSIV actuator trains and inoperability of the MSIVs themselves.

A.1

With a single actuator train inoperable on one MSIV, action must be taken to restore the inoperable actuator train to OPERABLE status within 7 days. The 7-day Completion Time is reasonable in light of the dual-redundant actuator train design such that with one actuator train inoperable, the affected MSIV is still capable of closing on demand via the remaining OPERABLE actuator train. The 7-day Completion Time takes into account the redundant OPERABLE actuator train to the MSIV, reasonable time for repairs, and the low probability of an event occurring that requires the inoperable actuator train to the affected MSIV.

B.1

With an actuator train on one MSIV inoperable and an actuator train on an additional MSIV inoperable, such that the inoperable actuator trains are not in the same separation group, action must be taken to restore one of the inoperable actuator trains to OPERABLE status within 72 hours. With two actuator trains inoperable on two MSIVs, there is an increased likelihood that an additional failure (such as the failure of an actuation logic train) could cause one MSIV to fail to close. The 72-hour Completion Time is reasonable since the dual-redundant actuator train design ensures that with only one actuator train on each of two affected MSIVs inoperable, each MSIV is still capable of closing on demand.

C.1

With an actuator train on one MSIV inoperable and an actuator train on an additional MSIV inoperable, but with both inoperable actuator trains in the same separation group, action must be taken to restore one of the inoperable actuator trains to OPERABLE status within 24 hours. The 24-hour Completion Time provides a reasonable amount of time for restoring at least one actuator train since the dual-redundant actuator train design for each MSIV ensures that a single inoperable actuator train cannot prevent the affected MSIV(s) from closing on demand. With two actuator trains inoperable in the same separation group, an additional failure (such as the failure of an actuation logic train in the other separation group) could cause both affected MSIVs to fail to close on demand. The 24 hour

BASES

ACTIONS

C.1 (continued)

Completion Time takes into the redundant OPERABLE actuator trains to the affected MSIVs and the low probability of an event occurring that requires the inoperable actuator trains to the affected MSIVs.

D.1

Required Action D.1 provides assurance that the appropriate Action is entered for the affected MSIV if its associated actuator trains become inoperable. Failure of both actuator trains for a single MSIV results in the inability to close the affected MSIV on demand.

E.1

With three or more MSIV actuator trains inoperable or when Required Action A.1, B.1, or C.1 cannot be completed within the required Completion Time, the affected MSIVs may be incapable of closing on demand and must be immediately declared inoperable. Having three actuator trains inoperable could involve two inoperable actuator trains on one MSIV and one inoperable actuator train on another MSIV, or an inoperable actuator train on each of three MSIVs, for which the inoperable actuator trains could all be in the same separation group or be staggered among the two separation groups.

Depending on which of these conditions or combinations is in effect, the condition or combination could mean that all of the affected MSIVs remain capable of closing on demand (due to the dual-redundant actuator train design), or that at least one MSIV is inoperable, or that with an additional single failure up to three MSIVs could be incapable of closing on demand. Therefore, in some cases, immediately declaring the affected MSIVs inoperable is conservative (when some or all of the affected MSIVs may still be capable of closing on demand even with a single additional failure), while in other cases it is appropriate (when at least one of the MSIVs would be inoperable, or up to three could be rendered inoperable by an additional single failure). Required Action E.1 is conservatively based on the worst-case condition and therefore requires immediately declaring all the affected MSIVs inoperable. Declaring two or more MSIVs inoperable while in MODE 1 requires entry into LCO 3.0.3.

BASES

ACTIONS (continued)

F.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs. Condition F is entered when one MSIV is inoperable in MODE 1, including when both actuator trains for one MSIV are inoperable. When only one actuator train is inoperable on one MSIV, Condition A applies.

The 8 hour Completion Time is consistent with that allowed for containment isolation valves that isolate a closed system penetrating containment. This time is reasonable due to the relative stability of the closed system which provides an additional passive means for containment isolation.

G.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition H would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

H.1 and H.2

Condition H is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition F.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid.

BASES

ACTIONS

H.1 and H.2 (continued)

The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

I.1 and I.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

INSERT B 3.7.2-7

~~This SR verifies that MSIV isolation time is ≤ 5.0 seconds on an actual or simulated actuation signal from each actuator train. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.~~

The Frequency is in accordance with the Inservice Testing Program.

This test can be conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each actuator train can close its respective MSIV on an actual or simulated actuation signal. The manual fast close hand switch in the control room provides an acceptable actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

INSERT B 3.7.2-7

This SR verifies that the closure time of each MSIV is within the limits of Figure B 3.7.2-1 from each actuator train when tested pursuant to the Inservice Testing Program. The MSIV isolation time is explicitly assumed in the accident analyses that credit the Steam Line Isolation. Figure B 3.7.2-1 is a curve of the MSIV isolation time limit as a function of steam generator pressure. The acceptance curve for the MSIV stroke time conservatively accounts for potential pressure differential between the steam generator pressure indication and the pressure at the MSIV. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.2 (continued)

The frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 10.3.
2. USAR, Section 6.2.
3. USAR, Section 15.1.5.
4. 10 CFR 100.11.

INSERT FIGURE B 3.7.2-1 behind this page

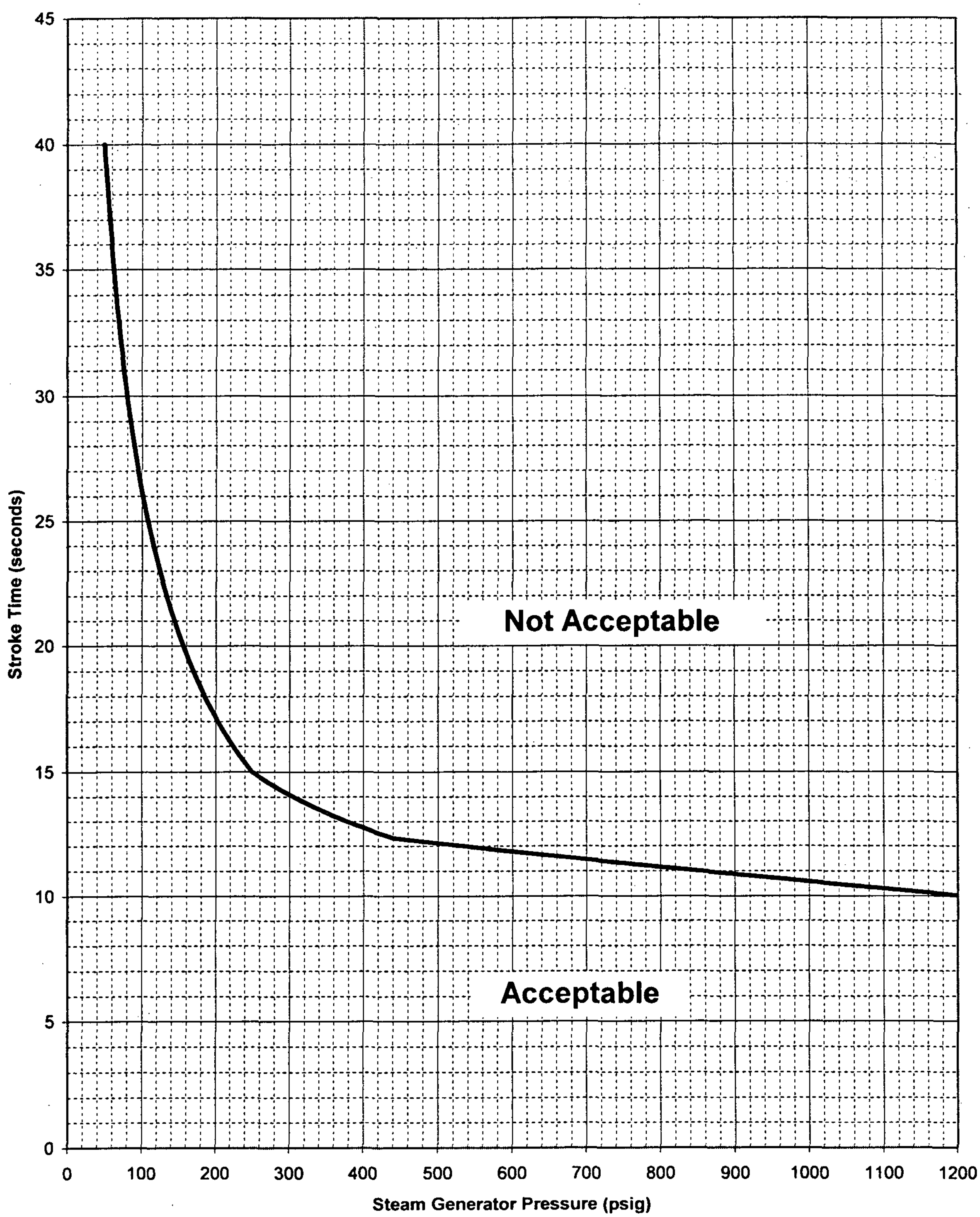


Figure B 3.7.2-1
MSIV Isolation Time vs. Steam Generator Pressure

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs)

and Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The Main Feedwater Regulation Valves (MFRVs) function to control feedwater flow to the SGs.

and MFRV bypass valves

and provide backup isolation of MFW flow in the event an MFIV fails to close.

The MFIV is a 14-inch gate valve with ~~a dual-redundant hydraulic actuator~~. Either actuation train can independently perform the safety function to fast-close the MFIV on demand. ~~Each actuator train consists of a hydraulic accumulator controlled by solenoid valves on the associated MFIV.~~ For each MFIV, ~~on~~ actuator train is associated with separation group 4 ("yellow"), and one actuator trains is associated with separation group 1 ("red").

system-medium actuation trains.

one

The MFRVs are air-operated angle valves used to control feedwater flow to the SGs from between ~~20~~% and full power. The MFRV bypass valves are air-operated globe valves used to control flow to the SGs up to ~~25%~~ power.

30

approximately 30%

Closure of the MFIVs terminates main feedwater flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of main feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

or MFRVs and MFRV bypass valves

The MFIVs isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

and one MFRV are

One MFIV ~~is~~ located on each MFW line, outside but close to containment. The MFIVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV closure. The piping volume from these valves to the steam generators is accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

and MFRVs and MFRV bypass valves

or MFRVs and MFRV bypass valves

INSERT B 3.7.3-1

INSERT B 3.7.3-1

The MFRV bypass valves are located in six inch lines that bypass flow around the MFRVs during low power operations. An MFIV can not be isolated with closed manual valves; the MFRV can be isolated upstream by a closed manual valve; and the MFRV bypass valves can be isolated both upstream and downstream with a closed manual valve.

BASES

BACKGROUND (continued)

and MFRVs and MFRV bypass valves

The MFIVs close on receipt of any safety injection signal, a T_{avg} - Low coincident with reactor trip (P-4), a low-low steam generator level, or steam generator water level - high high signal. ~~They~~ The MFIVs may also be actuated manually. In addition to the MFIVs, a check valve inside containment is available. The check valve isolates the feedwater line, penetrating containment, and ensures the pressure boundary of any intact loop not receiving auxiliary feedwater.

system-medium actuation

The MFIV actuators consist of two separate ~~pneumatic-hydraulic power~~ trains each receiving an actuation signal from one of the redundant ESFAS channels. A single active failure in one power train would not prevent the other power train from functioning. The MFIVs provide the primary success path for events requiring feedwater isolation and isolation of nonsafety related portions from the safety related portion of the system, such as, for auxiliary feedwater addition. INSERT B 3.7.3-2

A description of the MFIVs ~~and~~ MFRVs , and MFRV bypass valves is found in the USAR, Section 10.4.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Credit is taken in accident analysis for the MFIVs to close on demand. The safety function of the MFRVs and associated bypass valves credited in accident analysis is to provide a backup to the MFIVs for the potential failure of an MFIV to close even though the MFRVs are located in the nonsafety related portion of the feedwater system. Further assurance of feedwater flow termination is provided by the SGFP trip function; however, this is not credited in accident analysis. The accident analysis credits the main feedwater check valves as backup to the MFIVs to prevent SG blowdown for pipe ruptures in the non-seismic Category I portions of the feedwater system outside containment.

Criterion 3 of 10 CFR 50.36(c)(2)(ii) indicates that components that are part of the primary success path and that actuate to mitigate an event that presents a challenge to a fission product barrier should be in Technical Specifications. *The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria) so that the plant response to the event remains within appropriate acceptance criteria.* The primary success path does not include backup and diverse equipment. The MFIVs, with their dual-redundant actuators, are the primary success path for feedwater isolation; the MFRVs, MFRV bypass valves, and the SGFP trip are backup and diverse equipment. ~~Therefore, only the MFIVs are incorporated into Technical Specifications.~~ The MFIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The MFRVs and MFRV bypass valves satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii).

INSERT B 3.7.3-2

The MFRV and MFRV bypass valve actuators consist of two separate actuation trains each receiving an actuation signal from one of the redundant ESFAS channels. Both trains are required to actuate to close the valve.

BASES

and MFRVs and MFRV bypass valves

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the steam generators, following an FWLB or main steam line break. ~~These valves~~ The MFIVs will also isolate the nonsafety related portions from the safety related portions of the system.

and four MFRVs and four MFRV bypass valves

This LCO requires that four MFIVs and their associated actuator trains be OPERABLE. ~~The MFIVs are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.~~

An MFIV actuator train is considered OPERABLE when it is capable of fast-closing the associated MFIV on demand and within the required isolation time. ~~This includes having adequate accumulator pressure to support fast closure of the MFIV within the required isolation time and instrument air supply and pressure to the valve regulator is within limits.~~

INSERT B 3.7.3-3a

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, and failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. In MODES 1, 2, and 3, the MFIVs are required to be OPERABLE to perform their isolation function and limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. ~~When the valves are closed, they are already performing their safety function.~~ The MFIV actuator trains must be OPERABLE in MODES 1, 2, and 3 to support operation of the MFIV.

and MFRVs and MFRV bypass valves

INSERT B 3.7.3-3b

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs can be closed since MFW is not required.

ACTIONS

The LCO specifies OPERABILITY requirements for the MFIVs as well as for their associated actuator trains. The Conditions and Required Actions for TS 3.7.3 separately address inoperability of the MFIV actuator trains and inoperability of the MFIVs themselves.

and MFRVs and MFRV bypass valves

INSERT B 3.7.3-3a

The MFIVs are considered OPERABLE when isolation times are within the limits of Figure B 3.7.3-1 when given a fast close signal and they are capable of closing on an isolation actuation signal. The MFRVs and MFRV bypass valves are considered OPERABLE when isolation times are ≤ 15 seconds when given an isolation actuation signal and they are capable of closing on an isolation actuation signal. For the MFRVs and the MFRV bypass valves, the LCO requires only that the trip close function is OPERABLE.

INSERT B 3.7.3-3b

Exceptions to the Applicability are allowed where the valve is assured of performing its safety function as follows:

a. When the MFIV is closed and de-activated, it is performing is safety function. Requiring the valve closed and de-activated provides dual assurance that it is performing its safety function. When the valve is de-activated, power is removed from the actuation solenoids on the valves.

b. When the MFRV is closed and de-activated or is closed and isolated by a closed manual valve, it is performing its safety function. Requiring the valve closed and de-activated provides dual assurance that it is performing its safety function. When the valve is de-activated, power is removed from the actuation solenoids on the valves. Requiring the valve closed and isolated by a closed manual valve also provides dual assurance that it is performing its safety function.

c. When the MFRV bypass valve is closed and de-activated, or is closed and isolated by a closed manual valve, or is isolated by two closed manual valves, it is performing its safety function. Requiring the valve closed and de-activated provides dual assurance that it is performing its safety function. When the valve is de-activated, power is removed from the actuation solenoids on the valves. Requiring the valve closed and isolated by a closed manual valve also provides dual assurance that it is performing its safety function. Finally, there is dual assurance that the safety function is being performed when the MFRV bypass valve is isolated by two closed manual valves.

BASES

ACTIONS (continued)

A.1

With a single actuator train inoperable on one MFIV, action must be taken to restore the inoperable actuator train to OPERABLE status within 7 days. The 7-day Completion Time is reasonable in light of the dual-redundant actuator train design such that with one actuator train inoperable, the affected MFIV is still capable of closing on demand via the remaining OPERABLE actuator train. The 7-day Completion Time takes into account the redundant OPERABLE actuator train to the MFIV, reasonable time for repairs, and the low probability of an event occurring that requires the inoperable actuator train to the affected MFIV.

B.1

With an actuator train on one MFIV inoperable and an actuator train on an additional MFIV inoperable, such that the inoperable actuator trains are not in the same separation group, action must be taken to restore one of the inoperable actuator trains to OPERABLE status within 72 hours. With two actuator trains inoperable on two MFIVs, there is an increased likelihood that an additional failure (such as the failure of an actuation logic train) could cause one MFIV to fail to close. The 72-hour Completion Time is reasonable since the dual-redundant actuator train design ensures that with only one actuator train on each of two affected MFIVs inoperable, each MFIV is still capable of closing on demand.

C.1

With an actuator train on one MFIV inoperable and an actuator train on an additional MFIV inoperable, but with both inoperable actuator trains in the same separation group, action must be taken to restore one of the inoperable actuator trains to OPERABLE status within 24 hours. The 24-hour Completion Time provides a reasonable amount of time for restoring at least one actuator train since the dual-redundant actuator train design for each MFIV ensures that a single inoperable actuator train cannot prevent the affected MFIV(s) from closing on demand. With two actuator trains inoperable in the same separation group, an additional failure (such as the failure of an actuation logic train in the other separation group) could cause both affected MFIVs to fail to close on demand. The 24 hour Completion Time takes into the redundant OPERABLE actuator trains to the affected MFIVs and the low probability of an event occurring that requires the inoperable actuator trains to the affected MFIVs.

BASES

ACTIONS (continued)

D.1

Required Action D.1 provides assurance that the appropriate Action is entered for the affected MFIV if its associated actuator trains become inoperable. Failure of both actuator trains for a single MFIV results in the inability to close the affected MFIV on demand.

E.1

With three or more MFIV actuator trains inoperable or when Required Action A.1, B.1, or C.1 cannot be completed within the required Completion Time, the affected MFIVs may be incapable of closing on demand and must be immediately declared inoperable. Having three actuator trains inoperable could involve two inoperable actuator trains on one MFIV and one inoperable actuator train on another MFIV, or an inoperable actuator train on each of three MFIVs, for which the inoperable actuator trains could all be in the same separation group or be staggered among the two separation groups.

Depending on which of these conditions or combinations is in effect, the condition or combination could mean that all of the affected MFIVs remain capable of closing on demand (due to the dual-redundant actuator train design), or that at least one MFIV is inoperable, or that with an additional single failure up to three MFIVs could be incapable of closing on demand. Therefore, in some cases, immediately declaring the affected MFIVs inoperable is conservative (when some or all of the affected MFIVs may still be capable of closing on demand even with a single additional failure), while in other cases it is appropriate (when at least one of the MFIVs would be inoperable, or up to three could be rendered inoperable by an additional single failure). Required Action E.1 is conservatively based on the worst-case condition and therefore requires immediately declaring all the affected MFIVs inoperable.

F.1 and F.2

Condition F is modified by a Note indicating that separate Condition entry is allowed for each MFIV.

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close ~~or~~ ~~isolate~~ inoperable affected valves within ~~4~~ 72 hours. When these valves are closed, they are performing their required safety function. Condition F is entered when one or more MFIV is inoperable ~~in MODE 1~~, including when both actuator trains for one MFIV are inoperable. When only one actuator train is inoperable on one MFIV, Condition A applies.

and MFRV and MFRV Bypass Valves

BASES

ACTIONS F.1 and F.2 (continued)

72

, the redundancy afforded by the remaining OPERABLE valves,

The ~~4~~ hour Completion Time takes into account the redundancy afforded by the dual-redundant actuators on the MFIVs and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The ~~4~~ hour Completion Time is reasonable, based on operating experience. 72

Inoperable MFIVs that are closed must be verified on a periodic basis that they are closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

INSERT B 3.7.3-6a

~~G.1 and G.2~~ J.1 and J.2

and MFRVs and MFRV bypass valves

If the MFIV(s) cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

INSERT B 3.7.3-6b

~~This SR verifies that the closure time of each MFIV is ≤ 5 seconds on an actual or simulated main feedwater isolation actuation signal from each actuator train. The MFIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. This is consistent with Regulatory Guide 1.22 (Ref. 3)~~

The Frequency for this SR is in accordance with the Inservice Testing Program. Operating experience has shown that these components usually pass the Surveillance when performed at the Inservice Testing Program Frequency. This test is normally conducted in MODE 3 with the unit at nominal operating temperature and pressure, as discussed in Reference 2. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

INSERT B 3.7.3-6a

G.1 and G.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

H.1 and H.2

With one MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRV bypass valves that are closed or isolated must be verified on a periodic basis that they are closed and isolated. This is necessary to ensure that the assumptions of the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

I.1 and I.2

Two inoperable valves in the same flow path is treated the same as a loss of the isolation capability of this flow path. For each feedwater line there are two flow paths, defined as flow through the MFRV/MFIV and flow through the MFRV bypass valves/MFIV. Because the MFIV, MFRV, and MFRV bypass valve are of different designs, a common mode failure of the valves in the same flow path is not likely. However, under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated with 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV and MFRV bypass valve, or otherwise isolate the affected flow path.

INSERT B 3.7.3-6b

This SR verifies that the closure time of each MFIV, MFRV, and MFRV bypass valve is within limits (Figure B 3.7.3-1 for the MFIVs and ≤ 15 seconds for the MFRV and MFRV bypass valves) when tested pursuant to the Inservice Testing Program. The MFIV, MFRV, and MFRV bypass valve closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. This is consistent with Regulatory Guide 1.22 (Ref. 3). The Surveillance may be performed as required for post-maintenance testing of the MFRVs and MFRV bypass valves under appropriate conditions during applicable MODES. In particular, the MFRVs should normally not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the unit generating power. However, when the plant is operating using the MFRV bypass valves (at low power levels during MODE 1), the surveillance for the MFRVs may be performed for post-maintenance testing during such conditions without increasing plant risk.

If it is necessary to adjust stem packing to stop packing leakage and if a required stroke test is not practical in the current plant MODE, it should be shown by analysis that the packing adjustment is within torque limits specified by the manufacturer for the existing configuration of packing, and that the performance parameters of the valve are not adversely affected. A confirmatory test must be performed at the first available opportunity when plant conditions allow testing. Packing adjustments beyond the manufacturer's limits may not be performed without (1) an engineering analysis and (2) input from the manufacturer, unless tests can be performed after the adjustments. (Reference 4)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each actuator train can close its respective MFIV on an actual or simulated actuation signal. The manual close hand switch in the control room provides an acceptable actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage in conjunction with SR 3.7.3.1. However, it is acceptable to perform this Surveillance individually. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated

The frequency of MFIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

INSERT B 3.7.3-7

REFERENCES

1. USAR, Section 10.4.7.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
3. Regulatory Guide 1.22, Rev. 0.
4. NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants."

INSERT FIGURE B 3.7.3-1 behind this page

INSERT B 3.7.3-7

SR 3.7.3.3

This SR verifies that each MFRV and MFRV bypass valve is capable of closure on an actual or simulated actuation signal. The actuation of solenoids locally at the MFRVs and MFRV bypass valves constitutes an acceptable simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage in conjunction with SR 3.7.3.1. However, it is acceptable to perform this Surveillance individually.

The Frequency of MFRV and MFRV bypass valve testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. This Frequency is acceptable from a reliability standpoint. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

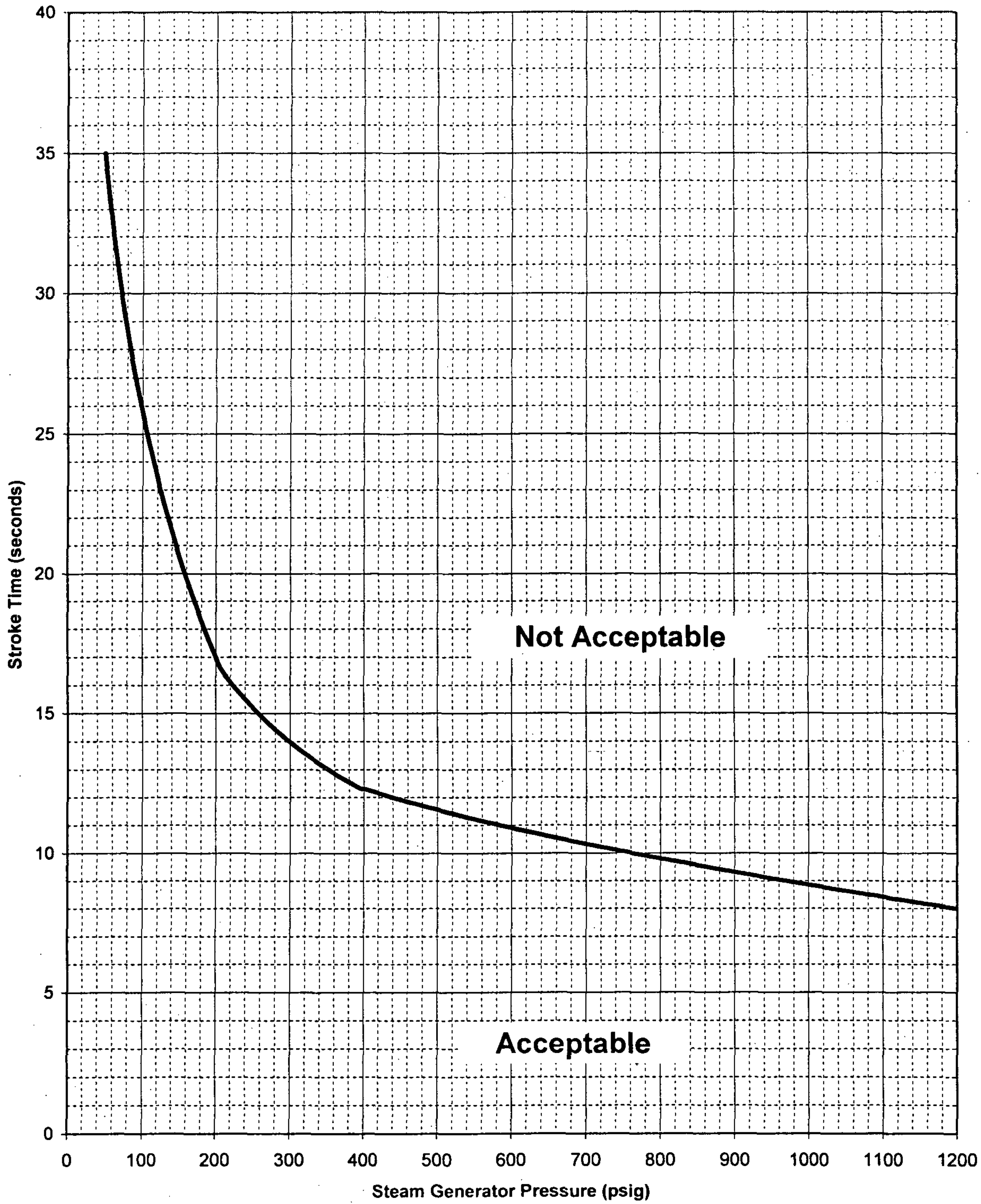


Figure B 3.7.3-1
MFIV Isolation Time Limit vs. Steam Generator Pressure

SUMMARY OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by WCNOC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Kevin Moles at (620) 364-4126.

COMMITMENT	Due Date/Event
The license amendment will be implemented prior to startup from Refueling Outage 16. Final TS Bases changes will be implemented pursuant to TS 5.5.14 at the time the amendment is implemented.	Prior to startup from Refueling Outage 16