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Ladies and Gentlemen: ULNRC- 04592

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
PROPOSED REVISION TO TECHNICAL SPECIFICATION 1.1, "DEFINITIONS"; TECHNICAL SPECIFICATION 3.7.3 "MAIN FEEDWATER ISOLATION VALVES (MFIVs)"; AND STEAM GENERATOR TUBE RUPTURE WITH OVERFILL RE-ANALYSIS
Reference: ULNRC-04845, dated May 9, 2003 (LER 2003-003-00)

Pursuant to 10 CFR 50.90, AmerenUE hereby requests an amendment to the Facility Operating License No. NPF-30 for the Callaway Plant. The amendment request incorporates the attached changes into the Callaway Plant Technical Specifications. Revisions to the Technical Specifications are required based on the impact of plant modifications scheduled for components in the Main Feedwater (MFW) and Auxiliary Feedwater (AFW) systems. In addition, as part of this application, AmerenUE requests NRC review and approval of the re-analysis of the steam generator tube rupture (SGTR) with overfill accident.

As discussed in the referenced letter, which transmitted LER 2003-003-00 to the NRC, during the plant review process for the modifications, AmerenUE identified a potential adverse impact on the SGTR with overfill analysis. In support of the modifications, FSAR Chapter 6.2 and Chapter 15 accidents were evaluated for impact and the radiological and thermal-hydraulic consequences for the SGTR with overfill accident were re-analyzed. The revised analysis uses International Commission on Radiological Protection Publication 30 (ICRP 30) based dose conversion factors (DCFs) and a revised iodine spike model based on Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors; revised operator action times for their responses to a SGTR; and other revised inputs and assumptions consistent with current plant configuration and operation. AmerenUE proposes to incorporate the

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approved re-analysis of the SGTR accident into the Callaway FSAR. In addition, AmerenUE requests NRC approval to use Regulatory Guide 1.195 for other licensing basis dose applications.

This amendment application revises Technical Specification (TS) 3.7.3, "Main Feedwater Isolation Valves (MFIVs)", so that Surveillance Requirement 3.7.3.1 now requires a stroke time of 15 seconds rather than 5 seconds for the MFIVs. The revision is based on a plant modification which will replace the electro-hydraulic actuators currently installed on the MFIVs with system-medium actuators. The new system-medium actuators will improve MFIV reliability, reduce maintenance requirements and reduce personnel exposure to hazardous material. In addition, relaxing the required stroke time reduces the magnitude of Feedwater System pressure transients.

Another modification will replace the existing swing check valve in each AFW motor driven pump discharge line with an automatic recirculation control check (ARC) valve. At Callaway Plant the AFW system consists of two motor driven pumps (MDAFP) and one steam turbine driven pump. Each motor driven pump discharges through a check valve and a locked-open isolation valve to feed two steam generators. By replacing the swing check valve with the ARC valve, a potential for vibration in the AFW system is reduced and AFW flow margin is increased. The new ARC valve maintains minimum flow requirements for the pump, but as pump discharge flow increases, the recirculation line automatically, mechanically, modulates closed, so that all flow is diverted to the steam generators. The ARC valve provides the recirculation flow needed for pump protection at low flow rates and allows full diversion to the steam generators with increased discharge flow. This is an enhancement over the current continuous recirculation design.

The SGTR accident considers two scenarios: (1) SGTR with overfill and (2) SGTR with stuck open atmospheric steam dump valve (ASD). In these scenarios, the plant modifications result in increased main and auxiliary feedwater inventory in the ruptured steam generator. The offsite doses presented in the Callaway FSAR are based on the analysis of the scenario with a stuck open ASD on the ruptured steam generator. In the original submittal, this case was the limiting accident scenario for radiological consequences. The original submittal for the SGTR accident also concluded that SGTR with overfill could not occur at Callaway - a SNUPPS plant. Subsequent to this submittal, Callaway implemented fuel design changes and a power uprate. In response to NRC questions and at NRC request, Union Electric submitted an analysis for SGTR with overfill (ULNRC-1518, dated May 27, 1987). The analysis was based on a "forced" overfill scenario, because it used assumptions beyond the design basis accident to achieve the overfill condition. The NRC approved this analysis in a safety evaluation (SE) issued in 1990. However, it was not considered a design basis accident for Callaway and the analysis was not incorporated into the FSAR. The current content of FSAR Section 15.6.3 concludes that in the SGTR with overfill scenario, water relief is precluded and there is no release of radiation to the overfill scenario, water relief is precluded and there is no release of radiation to the environment. Now, based on the results of the revised analysis for SGTR with overfill and as discussed in LER 2003-003-00 (the referenced letter ULNRC-04845), the FSAR conclusion must be changed and the accident consequences revised in the updated FSAR.

The application of ICRP 30 based DCFs impacts the definition of Dose Equivalent Iodine I-131 described in Callaway Technical Specifications Section 1.1. In support of the new analysis, this amendment application proposes to revise Technical Specification 1.1, "Definitions", to allow the use of exposure-to-thyroid factors (DCFs) derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." The DCF numerical values are taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factor for Inhalation, Submersion, and Ingestion." The current Technical Specification definition allows use of thyroid DCFs from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" (ICRP-2). Use of the ICRP 30 DCFs and iodine spike modeling is desired because the methodology is recognized for its scientific basis; it is recommended in Regulatory Guide 1.195 (Section 4.1.2 and Appendix E) and it has been approved by the NRC for use by other licensees. Because applying the source term methodology of Regulatory Guide 1.195 revises the source term of a current design basis accident, 10 CFR 50.67 requires submittal of a license amendment under 10 CFR 50.59.

The appropriate Technical Specification Bases changes and FSAR changes for the modifications and license amendment requests are attached for information only.

Attachment 1 to this submittal is the required Affidavit. Attachment 2 provides a detailed description, safety analysis of the proposed changes, and the Callaway determination that the proposed changes do not involve a significant hazards consideration. Attachment 3 provides the existing Technical Specification pages marked-up to show the proposed changes. Attachment 4 provides a clean copy of the proposed Technical Specification pages. Attachment 5 provides the existing Technical Specification Bases pages marked-up to show the proposed changes (for information only). Attachment 6 provides FSAR pages marked-up to show the proposed changes (for information only).

This letter identifies actions committed to by AmerenUE and the Callaway Plant in this submittal. Other statements are provided for information purposes and are not considered to be commitments. A summary of the regulatory commitments included in this submittal is provided in Attachment 7. Attachment 8 provides a figure depicting the replacement MFIV actuator (system-medium operated actuator).

It has been determined that this amendment application does not involve a significant hazards consideration as determined per 10 CFR 50.92. In addition, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of this amendment.

AmerenUE requests approval of this proposed License Amendment by February 1, 2004 which is prior to the next refueling outage scheduled for April 2004. Approval of the requested Technical Specification changes prior to the outage will allow planned outage work to proceed in conjunction with the outage. The approved amendment will be implemented prior to restart of the unit following Refueling Outage 13.

Pursuant to 10 CFR 50.91(b)(1), AmerenUE is providing the State of Missouri with a copy of this proposed amendment.

If you should have any questions on the above or attached, please contact Dave Shafer at (314) 554-3104 or Dwyla Walker at (314) 554-2126.

Very truly yours,

David Shafer

David E. Shafer
Acting Manager, Regulatory Affairs

DJW/jdg

Attachments:
1) Affidavit
2) Evaluation
3) Markup of Technical Specification pages
4) Retyped Technical Specification pages
5) Proposed Technical Specification Bases changes (for information only)
6) Proposed Callaway FSAR changes (for information only)
7) Summary of Regulatory Commitments
8) Diagram for MFIV System-Medium Operated Actuator

STATE OF MISSOURI)
) S S
CITY OF ST. LOUIS)

David Shafer, of lawful age, being first duly sworn upon oath says that he is Acting Manager, Regulatory Affairs, for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company 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 David Shafer
David Shafer
Acting Manager, Regulatory Affairs

SUBSCRIBED and sworn to before me this 27th day of June, 2003.

Carol A. Head

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ULNRC-04592

ATTACHMENT 2

EVALUATION

PROPOSED REVISION TO TECHNICAL SPECIFICATION 1.1, "DEFINITIONS"; TECHNICAL SPECIFICATION 3.7.3 "MAIN FEEDWATER ISOLATION VALVES (MFIVs)"; AND STEAM GENERATOR TUBE RUPTURE WITH OVERFILL RE-ANALYSIS

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EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-30 for the Callaway Plant. Revisions to the Technical Specifications are required based on the impact of plant modifications scheduled for components in the main feedwater (MFW) and auxiliary feedwater (AFW) systems. In addition, as part of this application, AmerenUE requests NRC review and approval of the re-analysis of the steam generator tube rupture (SGTR) with overfill accident. As discussed in LER 2003-003-00, during the plant review process for the modifications, AmerenUE identified a potential adverse impact on the SGTR with overfill analysis.

In support of the modifications, FSAR Chapter 6.2 and Chapter 15 accidents were evaluated for impact and the radiological and thermal-hydraulic consequences for the SGTR with overfill accident were re-analyzed. The revised analysis uses International Commission on Radiological Protection Publication 30 (ICRP 30) based dose conversion factors (DCFs) and revised iodine spike modeling based on Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors; revised operator action times for their responses to a SGTR; and other revised inputs and assumptions consistent with current plant configuration and operation. AmerenUE proposes to incorporate the approved re-analysis of the SGTR with overfill accident into the updated Callaway FSAR. In addition AmerenUE requests NRC approval to use Regulatory Guide 1.195 for other licensing basis dose applications.

The appropriate Technical Specification Bases and FSAR changes are also attached (for information only), as appropriate, to reflect the various proposed changes.

2.0 PROPOSED CHANGES

This amendment application would revise Technical Specification (TS) 3.7.3, "Main Feedwater Isolation Valves (MFIVs)", Surveillance Requirement (SR) 3.7.3.1, to require a stroke time of 15 seconds rather than 5 seconds for the MFIVs. The revision is based on a plant modification which will replace the electro-hydraulic actuators currently installed on the MFIVs with system-medium actuators. The new system-medium actuators will improve MFIV reliability and reduce maintenance requirements. In addition, relaxing the required stroke time reduces the magnitude of Feedwater System pressure transients.

The TS Bases are also revised to reflect changes based on the new MFIV actuator design and the TS Bases Surveillance Requirement 3.7.3.1 section is revised to indicate that the closure time of each MFIV must be verified to be less than or equal to 15 seconds from each actuation train, when tested pursuant to the plant Inservice Testing Program.

As a result of the actuator replacements, additional changes are required to the main control board and the Main Steam and Feedwater Isolation System (MSFIS) cabinet and controls. These additional changes include the removal, rewiring and replacement of several relays and modification to affected test switches and indicators. A new program module will be added to the main program for the MSFIS actuation logic. The new programming module will process the inputs and produce desired outputs for the new MFIV actuators. The entire MSFIS actuation logic programming will be verified and validated as required to establish the quality of the new program module and to identify any impact on existing program modules. These additional changes do not require TS revisions; are evaluated and found acceptable under plant review programs performed under 10 CFR 50.59.

Another plant modification replaces the existing swing check valve in each AFW motor driven pump discharge line with an automatic recirculation control check (ARC) valve. The new ARC valve maintains minimum flow requirements for the pump, but as pump discharge flow increases, the recirculation line automatically, mechanically, modulates closed, so that all flow is diverted to the steam generators. Because the impact of increased flow margin to the AFW system, associated with the modification, has been evaluated for impact on FSAR Chapter 6.2 and Chapter 15 accidents, this change is included in the license amendment request package. However, this modification does not require TS revisions, is evaluated and found acceptable under plant review programs performed under 10 CFR 50.59.

The TS Bases are revised based on the impact of the change to the new ARC valve design and the MFIV actuator replacement. TS Bases Surveillance Requirement section 3.7.5.2 is updated to indicate revised MDAFP acceptance criteria for recirculation flow given the change in differential pressure.

In order to determine the impact of the plant modifications and the proposed changes to the Technical Specifications, the consequences to Chapter 6.2 and Chapter 15 accident scenarios have been evaluated. In addition, the SGTR with overfill accident is re-analyzed for thermal-hydraulic and radiological consequences. The revised analysis uses ICRP 30 based DCFs and revised iodine spike modeling based on Regulatory Guide 1.195; revised operator action times for their responses to a SGTR; and other revised inputs and assumptions consistent with current plant configuration and operation.

The application of ICRP 30 based DCFs impacts the definition of Dose Equivalent Iodine I-131 described in Callaway Technical Specifications Section 1.1. In support of the new analysis, this amendment application proposes to revise Technical Specification 1.1, "Definitions", to allow the use of exposure-to-thyroid factors (DCFs) derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." The DCF numerical values are taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factor for Inhalation, Submersion, and Ingestion." The current definition allows use of thyroid DCFs from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" (ICRP-2).

3.0 BACKGROUND

3.1 Main Feedwater Isolation Valve Actuator Replacement

The existing electro-hydraulic actuators for the Main Feedwater Isolation Valves (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, fyrequel, associated with the electro-hydraulic actuators). As a result, the existing electro-hydraulic actuators on the MFIVs will be replaced with system-medium actuators during Refuel 13. To accommodate the new actuators, the existing valve bonnets and yokes will be removed, and a new bonnet installed.

The safety design functions of the MFIV actuators are to close the MFIV within the required time frame and provide an acceptable pressure boundary for the Auxiliary Feedwater System, when closed. The proposed TS change will increase the required MFIV closure time from 5 seconds to 15 seconds.

<u>Description of New Actuators</u>

The new MFIV actuators are shown in Attachment 8. They are simple steam pistons, with the piston shaft attached directly to the valve stem. The new MFIV actuators are operated by system-medium (process fluid) to close the valve and utilize instrument air or process fluid to open the valve. The pressure of the process fluid acts upon the actuator piston to position the valve. The process fluid is directed to the actuator piston chamber by solenoid valves. Each MFIV has three solenoid valves per train with a total of six solenoids per valve. Two of the solenoids per train control the admission of process fluid above the actuator piston (the upper piston chamber), and one solenoid per train controls venting and/or pressurization below the actuator piston (the lower piston chamber). The upper and lower piston chambers are isolated from each other by means of double piston seals. The lower portion of the actuator is the steam chamber, which remains at system pressure. The steam chamber is connected to the inlet of the solenoid valves via internal steam ports in the wall of the actuator cylinder, therefore, there is no external process fluid piping to the solenoid valves. The steam chamber is isolated from the lower piston chamber by means of double stem seals and a leak tight backseat.

Opening the MFIV

To open an MFIV, depress the Open button on the main control board handswitch. This will energize solenoid valves MV1 and MV2 (vented position), energize solenoid valves MV3 and MV4 (closed position), and energize solenoid vent valves MV5 (closed position) and MV6 (pressurized position). If sufficient system pressure (≥ 30 psig) is available the MFIV will open due to the feedwater pressurizing the Lower Piston Chamber (LPC) through MV6. If sufficient system pressure is not available the MFIV can be opened by directing auxiliary medium (air) through manual valves HV3 and HV4 to the LPC. After a 12 minute time delay, solenoid valves MV5 and MV6 will be returned to a de-energized state (vented position). When using auxiliary medium to open an MFIV, manual valves HV5 and HV6 should be closed to prevent venting of the LPC once the 12 minute time delay has expired.

Closing the MFIV

Only one actuation train is required to close an MFIV. Both Upper Piston Chamber (UPC) solenoids within the train, MV1 and MV3 or MV2 and MV4, must be de-energized. When MV1 and MV3 or MV2 and MV4 de-energize, they open to admit feedwater from the valve steam chamber to the UPC. The LPC is vented through solenoids MV5 and/or MV6, which are in the de-energized state (vented position). After a 30 second time delay, solenoid valves MV5 and MV6 will go to an energized state (closed or pressurized position), preventing any leakage from the LPC. Note that the term "close" includes either closure from the main control board handswitch or closure from the Engineered Safety Feature Actuation System (ESFAS).

3.2 Auxiliary Feedwater Pump Discharge Check Valve Replacement

At Callaway Plant the AFW system consists of two motor driven pumps (MDAFP) and one steam turbine driven pump. The two motor driven pumps are driven by ac-powered electric motors. Each horizontal centrifugal pump takes suction from the nonsafety-related condensate storage tank, or alternatively, from the safety-related essential service water system. Pump design capacity includes continuous minimum flow recirculation, which is controlled by restriction orifices. Each motor driven pump discharges through a check valve and a locked-open isolation valve to feed two steam generators. The current pump discharge check valves are "swing style" check valves. Plant experience with these valves has resulted in documentation of their contribution to high vibration on both trains of MDAFP piping systems, where leaking MDAFP discharge flow control valves have been shown the direct causal effect of the vibration. In certain AFW system lineups, use of the discharge swing style check valves contributes to piping vibration and to system hydraulic instability.

Immediate corrective action involved reworking the discharge flow control valves to eliminate leakage and the direct cause of vibrations. Further corrective action to minimize piping vibration and system instability includes replacing the existing swing check valve in each MDAFP discharge line with an ARC valve. Evaluations support a change in design for the pump discharge check valve to eliminate susceptibility for system hydraulic instability and piping vibration.

The ARC valve performs all flow sensing, bypass pressure reduction, reverse flow protection and modulating recirculating flow in an integral three port valve. The valve is flow operated and does not require any instrument air or electrical needs to operate. Upon pump start up and without process demand, the bypass is completely open, recirculating the required minimum flow. When process demand starts, the spring loaded disc is lifted and held in position by flow demand. Until main process flow demand exceeds recommended minimum flow, the valve will modulate. As the process flow demand increases beyond minimum flow, the bypass will close. Due to the specific design for Callaway Plant no calibration is required.

The ARC valve provides the check valve function provided by the existing discharge check valve and directs flow to either the main process flow or to recirculation flow. Because of the ARC valve design, margin for the MDAFPs is increased. Because the ARC valve provides the recirculation flow needed for pump protection at low flow rates and allows full diversion to the steam generators with increased discharge flow, it is an enhancement over the current continuous recirculation design.

3.3 Re-Analysis of Radiological Consequences for SGTR

<u>Background for the Technical Analysis</u>

The method of technical analysis used for evaluating the potential impacts of the modifications and the proposed changes (increased MFIV stroke time from 5 seconds to 15 seconds; the increase in AFW flow margin; and the use of ICRP 30 based DCFs and an iodine spike model based on Regulatory Guide 1.195) has been to review and assess the existing design basis evaluations for the impact of these changes.

The existing design basis evaluations were originally performed by Westinghouse Corporation (Westinghouse), the NSSS supplier and by Bechtel Corporation (Bechtel) the architect engineer, and in some cases as a collaborative effort between them. FSAR Chapter 15 accident analyses and FSAR Chapter 3.9(N).1.1, NSSS design thermal transients, were performed by Westinghouse. The original SGTR analyses were performed for Callaway as a SNUPPS plant.

FSAR Chapter 6.2 Mass and Energy Release Analyses included collaboration between Westinghouse and Bechtel. The post-MSLB containment pressure-temperature analysis for the Callaway Plant was resulted from a collaborative effort between Westinghouse and Bechtel. Westinghouse was responsible for calculating mass and energy releases, while Bechtel was originally responsible for using the mass and energy release data as an input to calculate containment pressure-temperature response. Currently, AmerenUE rather than Bechtel is the containment analyst and performs calculations for containment pressure-temperature response based on Westinghouse calculations of mass and energy release data.

In support of the proposed changes to the Technical Specifications and in order to determine the impact of the plant modifications, AmerenUE and Westinghouse have performed additional evaluations and calculations for the assessments. The consequences to Chapter 6.2 and Chapter 15 accident scenarios have been evaluated and the SGTR with overfill accident was re-analyzed for thermal-hydraulic and radiological consequences.

<u>Description of the Callaway SGTR Accident Analysis</u>

The SGTR accident considers two major scenarios: (1) the potential for an AFW control valve failure leading to overfill of the faulted steam generator with liquid entering the main steam line resulting in liquid relief through a failed open safety valve, and (2) the failure of an atmospheric steam dump (ASD) valve in the open position leading to continued release of steam generator fluid and contained radioactivity.

Current offsite doses presented in the Callaway FSAR are based on the SGTR scenario with the stuck open ASD on the ruptured steam generator. The failure of the ASD in the open position leads to a continued release until operator action stops the release. Based on the original submittal for SGTR accident consequences, this scenario is limiting for postulating the maximum radiological consequences. The doses calculated are acceptably below the applicable offsite dose limits from Standard Review Plan (SRP) 15.6.3.

In the overfill case, the AFW control valve (an automatically positioned flow control valve commonly referred to as a "smart" valve) for the ruptured SG fails open. Primary to secondary break flow plus the maximized AFW flow forces the ruptured SG towards a water-solid condition. As postulated, the steam lines may fill, and water relief through a Main Steam Safety Valve (MSSV) could occur. The MSSVs are not designed for water relief, and following water relief, the MSSV may fail to reseat for some period of time.

Callaway's initial Operating License required Union Electric to submit the analysis for the SGTR accident for NRC review and approval. The original submittal was made on behalf of both SNUPPS plants (SLNRC 86-01, dated January 8, 1986). The SNUPPS analysis concluded that the SNUPPS units would not overfill. Subsequent to this submittal, Callaway Plant implemented fuel design changes and a power uprate, rendering the plant different from the original SNUPPS analysis. In response to NRC questions, Union Electric submitted an additional analysis for SGTR with SG overfill (ULNRC-1518, dated May 27, 1987). The analysis was considered to be based on a "forced" overfill scenario, because it used assumptions beyond the design basis accident to achieve the overfill condition. The NRC approved this analysis in a safety evaluation (SE) issued in 1990. The SE specifically included confirmatory calculations for the overfill scenario's radiological consequences. However, because it was not considered a design basis accident, Union Electric did not incorporate the scenario into the FSAR. The current content of FSAR Section 15.6.3 concludes that in the SGTR with overfill scenario, water relief is precluded and there is no release of radiation to the environment. Now, based on the results of the revised analysis for SGTR with overfill, and based on LER 2003-003-00 transmitted in ULNRC-04845, dated May 9, 2003, the FSAR conclusion must be changed.

Description of the Revised Analysis for SGTR with Overfill

The revised SGTR analysis uses ICRP 30 based DCFs and a revised iodine spiking model, based on Regulatory Guide 1.195 (different from the current method and DCFs described for SGTR with a stuck open ASD presented in the FSAR Table 15A-4); revised operator action times for their responses to a SGTR; and other revised inputs and assumptions consistent with current plant configuration and operation. The application of ICRP 30 based DCFs impacts the definition of Dose Equivalent Iodine-131 described in Callaway Technical Specifications Section 1.1. The current definition allows use of thyroid DCFs from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" (ICRP-2). The revised Technical Specification 1.1, "Definitions", would allow the use of the DCFs derived from data provided in the ICRP Publication 30. The revised analysis uses the exposure-to-thyroid factors (DCFs) derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers."

The new analysis also uses a revised iodine spike modeling factor of 335, based on Regulatory Guide 1.195, which differs from the factor of 500 used in the current analysis of record. Together these changes reduce the magnitude of the accident source term and result in lower thyroid doses than would be obtained using the current methodology of record. These methods are found acceptable and recommended in Regulatory Guide 1.195, Section 4.1.2 and in Appendix E.

Use of these revisions is desired because the methodology is recognized for its scientific basis; the methods are recommended by Regulatory Guide 1.195 and the methodology has been approved by the NRC for use by other licensees (see Section 7.0 for precedents). Because applying the source term methodology of Regulatory Guide 1.195 revises the source term of a current design basis accident, 10 CFR 50.67 requires submittal of a license amendment under 10 CFR 50.90.

4.0 TECHNICAL ANALYSIS

The safety implications associated with the plant modifications and with the use of Regulatory Guide 1.195 source term methodology and other methodology changes incorporated into the SGTR with overfill re-analysis, have been evaluated.

The replacement of the electro-hydraulic MFIV actuators with system-medium actuators and the proposed increase in MFIV isolation time has been evaluated for impact on the accident analyses presented in FSAR Chapters 6.2 and 15. MFIV isolation time is not explicitly modeled in all accident sequences. Several accidents take credit for either MFIV isolation time or the addition of a minimum flowrate of AFW. The accidents that are applicable for this review are listed below and the evaluations are discussed in the specific accident summaries provided below.

The safety implications associated with use of the ARC valve and the associated increase in AFW delivered to the steam generators are evaluated for adverse impact on FSAR Chapter 6.2 and Chapter 15 analyses for which maximum AFW flow is limiting. For the other accidents, the increase in AFW flow margin has no impact or is beneficial to mitigation of the accident. The events evaluated include the SGTR with overfill; inadvertent opening of a steam generator safety valve; steam system piping failure and steamline break inside containment. The evaluations are discussed in the specific accident summaries provided below.

Finally, the safety implications of using Regulatory Guide 1.195 source term methodology changes to re-analyze the SGTR with overfill accident are evaluated. This includes use of thyroid DCFs based on ICRP-30; use of an iodine spike model that yields a different factor than used in the analysis of record; and use of different operator response times. This evaluation is discussed below in the SGTR with overfill accident.

ANALYSIS

MFIV isolation time is not explicitly modeled in the following Chapter 15 accident sequences or categories of sequences (i.e., MFIV position is not an essential analysis consideration):

- Excessive Increase in Secondary Steam Flow (FSAR 15.1.3)
- Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow (FSAR 15.2.1)
- Loss of External Electrical Load (FSAR 15.2.2)
- Turbine Trip (FSAR 15.2.3)
- Inadvertent Closure of Main Steam Isolation Valves (FSAR 15.2.4)
- Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip (FSAR 15.2.5)
- Decrease in Reactor Coolant System Flow Rate Accidents (FSAR 15.3, other than Locked Rotor Steam releases discussed later)
- Reactivity and Power Distribution Anomaly Accidents (FSAR 15.4, other than RCCA Ejection Steam releases as discussed later)
- Increase in Reactor Coolant Inventory Accidents (FSAR 15.5)
- Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR 15.6.1)

- Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment (FSAR 15.6.2)
- Large Break Loss-of-Coolant Accident (LBLOCA) (FSAR 15.6.5)
- Radioactive Release from a Subsystem or Component Accidents (FSAR 15.7)
- ATWS (FSAR 15.8)

The following accidents are applicable to the review for impact of the proposed amendments and are discussed in subsequent paragraphs:

- Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (FSAR 15.1.1). Reviewed for increased MFIV stroke time.

- Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (FSAR 15.1.2). Reviewed for increased MFIV stroke time.

- Inadvertent Opening of a Steam Generator Relief or Safety Valve (FSAR 15.1.4) Reviewed for increased MFIV stroke time and for additional AFW flow.

- Steam System Piping Failure (FSAR 15.1.5) Reviewed for increased MFIV stroke time and for additional AFW flow.

- Loss of Nonemergency AC Power to the Station Auxiliaries (FSAR 15.2.6) Reviewed for increased MFIV stroke time.

- Loss of Normal Feedwater Flow (FSAR 15.2.7) Reviewed for increased MFIV stroke time.

- Feedwater System Pipe Break (FSAR 15.2.8) Reviewed for increased MFIV stroke time.

- Steam Generator Tube Rupture (SGTR) (FSAR 15.6.3) Reviewed for increased MFIV stroke time, additional AFW flow, and Regulatory Guide 1.195 source term methodology and other methodology changes.

- Small Break Loss-of Coolant Accident (SBLOCA) (FSAR 15.6.5) Reviewed for increased MFIV stroke time.

In addition to the Chapter 15 accident sequences listed above, the FSAR Section 6.2 Mass and Energy Release Analyses, FSAR Section 3.9(N).1.1, NSSS design thermal transients; and the steam releases assumed in radiological analyses were also reviewed for potential impact.

Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

This event is analyzed at the limiting hot full power (HFP) conditions. Feedwater isolation is assumed to occur from a steam generator high-high water level signal. The HFP feedwater malfunction cases analyzed for Callaway cover possible system malfunctions that could result in either increased feedwater flow or decreased feedwater temperature.

The steam generator high-high water level setpoint and MFIV stroke time are explicitly modeled in the Feedwater Malfunction Temperature Decrease case. However, in this case, the reactor is tripped on an overpower ΔT signal, and the turbine is then tripped by the P-4 signal. The minimum Departure from Nucleate Boiling Ratio (DNBR) occurs approximately at the time of reactor trip with feedwater isolation occurring well after this time. Therefore, feedwater isolation does not play a role in this case. It can be concluded, therefore, that changes in the time of feedwater isolation do not impact the limiting DNBR results obtained for this event.

The analysis for this accident sequence continues to satisfy the acceptance criteria. As such, it can be concluded that an increase in MFIV stroke time to 15 seconds is acceptable with respect to the Feedwater System Malfunctions that result in a decrease in feedwater temperature.

Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

This event is analyzed at hot zero power (HZP) and hot full power (HFP) conditions. Feedwater isolation is assumed to occur from a steam generator high-high water level signal in the HFP Feedwater Malfunction cases.

In the HFP Feedwater Flow Increase case, the safety analysis steam generator high-high water level setpoint is assumed to be the Safety Analysis Limit (SAL) of 100% Narrow Range Span (NRS). Results of plant-specific sensitivity studies indicate that a slight delay in reaching a turbine trip and/or feedwater isolation signal, due to minor changes in the steam generator high-high water level setpoint, has little effect on the minimum calculated DNBR for this case. From the results of these sensitivities one can conclude that an increase of 10 seconds in MFIV stroke time would also have little impact on the DNBR transient for this event. Based on this, it can be concluded that the 10-second change in the time for main feedwater isolation does not significantly impact the DNBR results obtained for this event.

The acceptance criterion for this sequence is a minimum DNBR of 1.69. The current Analysis of Record for Callaway results in a minimum DNBR of 2.078, which is well above the minimum acceptable DNBR.

The analyses for this accident sequence continue to satisfy their acceptance criteria. As such, it can be concluded that an increase in MFIV stroke time to 15 seconds is acceptable with respect to the Feedwater System Malfunctions that result in an increase in feedwater flow.

Inadvertent Opening of a Steam Generator Relief or Safety Valve

The inadvertent opening of a steam generator relief or safety valve accident sequence is bounded by the analyses for the main steam line break. Refer to the discussion below under "Steam System Piping Failure".

Steam System Piping Failure

Although main feedwater would not be available during no-load conditions, main feedwater is conservatively assumed to increase to a nominal (full power) mass flow rate at the beginning of the transient for the hypothetical Steamline Break (SLB). This maximizes the initial feedwater flow to increase the heat removal capacity of the steam generators. This feedwater flow is maintained until feedwater isolation occurs. Thus, the SLB analyses credit feedwater isolation to limit the cooldown of the RCS.

The limiting SLB cases analyzed for Callaway also utilize a very conservative auxiliary feedwater (AFW) flow model, also aimed at maximizing the cooldown of the RCS. This model preferentially directs the AFW flow to the faulted steam generator from the start of the transient resulting in very significant asymmetric RCS loop temperatures. It is this asymmetric temperature distribution in the core that makes the SLB a limiting DNBR transient.

Impact of Increased MFIV Stroke Time

The proposed increase in MFIV stroke time represents a symmetric as opposed to an asymmetric change in feedwater flow to the RCS loops. It has been determined that extending the symmetric addition of main feedwater for an additional 10 seconds, to account for an increased MFIV stroke time, has little effect on the DNBR transient calculated for this event. The feedwater added during the additional 10 seconds does not significantly change the resulting RCS cooldown calculated using the conservative modeling assumptions detailed above. Furthermore, the increase in the stroke time of these valves will have no effect on the delivery of auxiliary feedwater to the steam generators. As such, the increase in MFIV stroke time to 15 seconds is acceptable with respect to the SLB event.

Impact of Increased AFW Flow Margin

The maximum AFW flow that can be delivered to the faulted steam generator during a steamline break or inadvertent opening of a steam generator safety/relief valve event has increased. The AFW flows assumed to be delivered to the three intact steam generators, during these events, have also changed. Even though the accident assumptions preferentially direct the AFW flow to the faulted steam generator from the start of the transient, resulting in very significant asymmetric RCS loop temperatures, the design of the AFW system prevents all of the AFW flow from going into the faulted steam generator. The remainder of the total flow is distributed to the three intact steam generators. Updated flow rates were evaluated for the impact of this change on the steamline break or inadvertent opening of a steam generator safety/relief valve event. The flow to the faulted steam generator was increased to 1493.50 gpm from 1417.7 gpm. The flow to the three intact steam generators was changed to 302.28 gpm from 199.7 gpm for one intact SG and 439.3 gpm for each of the other intact steam generators. Based on previous evaluations of AFW flow increases and general sensitivity studies documented in WCAP-9226, Revision 1, the magnitude of the changes in the AFW flow has an insignificant effect on the DNBR results for the steamline break and inadvertent secondary side depressurization events.

Conclusion

The analyses for these accident sequences continue to satisfy their acceptance criteria and the conclusions presented in the FSAR for these Non-LOCA events remain valid. The minimum DNBR for SLB is calculated for each fuel cycle. Fuel Cycle 13 has a minimum DNBR of 2.072 as compared to the acceptance criteria of 1.50. This significant margin, combined with the determination that DNBR is insensitive to the change in MFIV stroke time or the change in increased AFW flow margin, demonstrates that the proposed modification to the MFIV actuators and the proposed installation of the ARC valves do not represent an unacceptable adverse impact on the SLB analyses.

Loss of Nonemergency AC Power to the Station Auxiliaries

In contrast to the typical Westinghouse NSSS design, the feedwater line check valves at the Callaway Plant are located downstream (inside containment) of the AFW injection point rather than upstream (outside containment). This configuration demands a protection system functional requirement to ensure that the main feedwater system is isolated due to a low-low steam generator water level signal. Although feedwater isolation is not explicitly modeled as such in the Loss of Nonemergency AC Power to the Station Auxiliaries analyses, the assumed AFW purge volume (specifically, the AFW purge volume is assumed to be the total volume of piping between the feedwater isolation valve and the steam generator inlet) implies main feedwater isolation occurs. When main feedwater is lost, as is the case in this event, the purge volume fills with residual main feedwater at a high temperature. This fill water must be displaced before colder AFW can reach the steam generators. Time for the displacement (before cold AFW reaches the steam generators) represents an additional delay beyond an assumed 60-second AFW actuation time delay. The 60-second actuation delay accounts for the time it takes from receipt of the low-low steam generator water level signal until the AFW pumps reach full speed. The time for displacement accounts for the delay it takes to fill the piping volume between the feedwater isolation valve and the feedwater line check valve. Thus, based on the current Analysis of Record for this event, main feedwater isolation must occur prior to crediting AFW initiation. An increase in the MFIV stroke time to 15 seconds remains well below the combined 60 seconds assumed AFW actuation time and the time for displacement. Therefore, an increase in MFIV stroke time from 5 seconds to 15 seconds is acceptable with respect to the Loss of Nonemergency AC Power to the Station Auxiliaries event.

The SNUPPS Auxiliary Feedwater System Reliability Evaluation (Reference 8.14) performed for the limiting Condition II events is based on the availability of only one Motor Driven Auxiliary Feedwater Pump (MDAFP) (FSAR Section 18.2.7.3). To be consistent with this assumption, the analyses of the FSAR Chapter 15 Condition II events conservatively assume only one of the two MDAFPs is available.

The most limiting single failure for the Loss of Nonemergency AC Power to the Station Auxiliaries, which is considered a Condition II event, is the failure of a single protection train, which would prevent one MDAFP from starting. In this scenario, with the new system-medium actuators installed on the MFIVs, the single protection train failure would result in one train of solenoid valves not going to their safety position. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 10 gpm of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoids. If the Auxiliary Feedwater System is in operation, this potential diversion flow could be Auxiliary Feedwater for the steam generators. For analytical conservatism, an additional 50% was added to this potential flow diversion, which results in 15 gpm per actuator, or 30 gpm total. For this event it is also assumed the Turbine Driven Auxiliary Feedwater Pump (TDAFP) is also not available. Based on MDAFP surveillance data, the weakest MDAFP is capable of providing 532.0 gpm flow to two steam generators at 1221 psig. Accounting for a potential 30 gpm flow diversion results in a net flow of 502.0 gpm (532.0 gpm – 30 gpm) to two steam generators. Therefore, the limiting MDAFP will provide sufficient flow beyond the required 480 gpm assumed in the accident analysis. Increased AFW flow associated with the new MDAFP ARC valve will be beneficial.

If a single failure of the TDAFP is assumed (instead of a single protection train) and one MDAFP is not available, all solenoid valves on the MFIV system-medium actuators are assumed to function as designed. Therefore, the solenoid valves will provide a complete pressure boundary for the AFW system and no diversion of AFW through the MFIV actuators will occur.

<u>Loss of Normal Feedwater Flow</u>

As discussed in the evaluation for the Loss of Nonemergency AC Power to the Station Auxiliaries, the configuration of the Callaway Plant's main feedwater system demands a protection system functional requirement to ensure that the main feedwater system is isolated due to a low-low steam generator water level signal. Although main feedwater isolation is not explicitly modeled as such in the Loss of Normal Feedwater analyses, the assumed AFW purge volume (see discussion under "Loss of Nonemergency AC Power to the Station Auxiliaries") implies, by the fact it accumulates, that main feedwater isolation occurs. Thus, based on the current analysis of record for this event, main feedwater isolation must occur prior to crediting AFW initiation. As discussed above under "Loss of Nonemergency AC Power to the Station Auxiliaries", an increase in the MFIV stroke time from 5 seconds to 15 seconds will still allow for main feedwater system isolation to occur as originally analyzed. It can be concluded, therefore, that an increase in MFIV stroke time to 15 seconds is acceptable with respect to the Loss of Normal Feedwater Flow event.

As also discussed in the evaluation for the Loss of Nonemergency AC Power to the Station Auxiliaries, the SNUPPS Auxiliary Feedwater System Reliability Evaluation performed for the limiting Condition II events is based on the availability of only one MDAFP. Since the Loss of Normal Feedwater Flow event is considered a Condition II event, it also assumes only one of the two MDAFPs is available.

Similar to the Loss of Nonemergency AC Power event, the most limiting single failure for the Loss of Normal Feedwater Flow, is the failure of a single protection train, which would prevent one MDAFP from starting. Again as described in the previous accident scenario, the TDAFP is also not available. With the new system-medium actuators installed on the MFIVs, assuming the weakest MDAFP is providing flow to two steam generators and accounting for the flow diversion from the MFIV actuators, results in a net flow of 502.0 gpm (532.0 gpm – 30 gpm) to the steam generators. Once again, the limiting MDAFP will provide flow above and beyond the required 480 gpm assumed in the accident analysis. Increased AFW flow associated with the new MDAFP ARC valve will be beneficial.

When the single failure of the TDAFP is assumed (instead of a single protection train) and one MDAFP is not available, all solenoid valves on the MFIV system-medium actuators will function as designed. Therefore, the solenoid valves will provide a complete pressure boundary for the AFW system and no diversion of AFW through the MFIV actuators will occur.

<u>Feedwater System Pipe Break</u>

As explained in the Loss of Normal Feedwater Flow section, Callaway Plant's configuration of the AFW system is unique. Although feedwater isolation is not explicitly modeled in the feedwater line break (FLB) analysis, the assumed AFW delivery time implies feedwater isolation occurs. Again, as discussed above, the total AFW system delivery is assumed in the analysis to be composed of two parts. The first part is a delay of 60 seconds to account for the time from receipt of the low-low steam generator water level signal until the AFW pumps reach full speed. The second part of the total AFW system delivery time is the delay to account for the time it takes to fill the piping volume between the feedwater isolation valve and the feedwater line check valve. Thus, based on the current Analysis of Record for this event, main feedwater isolation must occur prior to crediting AFW initiation.

An increase in the MFIV stroke time from 5 seconds to 15 seconds will still allow main feedwater system isolation to occur as originally analyzed. Since feedwater isolation will still occur prior to AFW initiation, it can be concluded that an increase in MFIV stroke time to 15 seconds is acceptable with respect to the FLB event.

A FLB event is considered a Condition IV event, which also assumes a single failure of one protection train. The accident analysis for the Feedwater System Pipe Break event was performed assuming a single protection train failure and, as a result, only takes credit for the TDAFP and one MDAFP. This accident analysis requires a total

AFW flow of 563.3 gpm to three intact Steam Generators, 470 gpm provided by the TDAFP and 93.3 gpm provided by the MDAFP, which is limited by the flow control valve in the MDAFP discharge line. A hydraulic analysis of the AFW System for this accident, using the latest pump surveillance data, found the pumps would always deliver at least 691.4 gpm. As previously stated, through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 10 gpm of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoids. If the Auxiliary Feedwater System is in operation, this potential diversion flow could be Auxiliary Feedwater for the steam generators. For analytical conservatism, an additional 50% was added to this potential flow diversion. Since this accident would use three MFIVs as part of the pressure boundary, a potential diversion flow of up to 15 gpm per actuator, 45 gpm total, of AFW flow could occur. This would result in a net flow of 646.4 gpm (691.4 gpm – 45 gpm) to three Steam Generators, which exceeds the FSAR required flow of 563.3 gpm for this accident. The increased AFW flow associated with the MDAFP ARC valves will be beneficial.

The Feedline Break accident sequence does not have radiological consequences explicitly reported in the FSAR. This accident sequence is considered to be bounded by the Main Steam Line Break sequence. The Licensing Bases Main Steam Line Break radiological consequences presented in the FSAR are based on the break of a Main Steam Line outside of Containment, and upstream of the Main Steam Isolation Valve. The radiological consequences are based on the release of the entire inventory of one steam generator direct to the atmosphere.

In the event of a Feedline break, and the failure of the solenoid valves associated with one logic train to function, there is the potential for a very small leak path through the new MFIV actuators (i.e. approximately 2 mm diameter). Release of mass through the MFIV actuators, driven by Containment Pressure, would continue to be bounded by the release through a double ended guillotine break of a main steam line, driven by Steam Generator secondary side pressure. Therefore, the radiological consequences of a Feedline Break accident continue to be bounded by the radiological consequences of a Main Steam Line Break accident.

Steam Generator Tube Rupture (SGTR)

The impact of the modifications and the associated increase in MFIV stroke time and AFW margin affects the analysis for the SGTR accident by introducing additional MFW and AFW inventory into the ruptured steam generator (SG). The SGTR accident considers two scenarios: (1) SGTR with overfill and (2) SGTR with stuck open atmospheric steam dump (ASD) valve. The review for impact resulted in the following evaluation for the SGTR with stuck open ASD and the re-analysis for the SGTR with overfill scenario.

SGTR with Stuck Open ASD

Current offsite doses presented in the Callaway FSAR are based on the SGTR scenario with the stuck open ASD on the ruptured SG. The failure of the ASD in the open position leads to a continuous release until operator action stops the release. Based on the Analysis of Record, this scenario is limiting for postulating the maximum radiological consequences. The MFIV stroke time increase and the increased AFW flow margin do not adversely impact the SGTR with stuck open ASD analysis. Quicker isolation of main feedwater flow causes an increase in break flow flashing and steam releases, resulting in more radioactivity released to the atmosphere and higher radiological consequences. As a result, the proposed increase in MFIV stroke time does not invalidate the results of the current SGTR with stuck-open ASD analysis. The MDAFP ARC valve and the associated increase in maximum AFW flow are beneficial to mitigation of the event.

The SGTR event is considered a Condition IV event. The worst single failure assumed for this event is a stuck open atmospheric steam dump valve. Because the single failure assumed during this event is a stuck open ASD valve, the ARC valve and all solenoid valves on the MFIV system-medium actuators function as designed. The MFIV actuator solenoid valves provide a complete pressure boundary for the AFW system and there is no diversion flow of AFW through the MFIV actuators.

SGTR with Overfill

The subject modifications will result in an additional 10 seconds of MFW flow and increased AFW flow into the faulted SG in this scenario. As a result, the additional liquid relief through a MSSV will affect the dose consequences of the event. A re-analysis of the SGTR with overfill event was performed to verify that the applicable offsite dose limits of Standard Review Plan 15.6.3 continue to be met.

The SGTR with overfill analysis is divided into two portions: (1) Thermal-hydraulic accident analysis, and (2) Offsite dose consequence analysis. Operator action times are important to the thermal-hydraulic portion of the SGTR reanalysis. These times determine the duration (i.e. minutes) for key stages of the SGTR accident. The stages, shown in the table below, are established as part of the accident analysis methodology due to the impact each has on the mass of contaminated liquid (i.e. source term) in the secondary side of the ruptured steam generator (SG).

TIME	KEY STAGE DESCRIPTION
T_1 (20 min.)	AFW flow to the ruptured S/G isolated
T_2 (30 min.)	Initiate RCS cooldown
T_3 (40 min.)	Complete RCS depressurization
T_4 (45 min.)	SI terminated
T_5 (60 min.)	RCS and S/G pressure equalized

Note: These revised operator times are incorporated into emergency procedures and operating crews are currently being trained to meet the re-analysis times.

Simulated control room exercises were performed in 2003 for this accident. The exercises have demonstrated that the operator action times that serve as inputs to the thermal-hydraulic analysis have increased above the times originally analyzed in the SGTR with overfill analysis presented to the NRC in ULNRC-1518, dated May 27, 1987 (See FSAR Table 15.6-1 draft mark-ups in Attachment 6). The current re-analysis for the SGTR with overfill accident is presented to the NRC via this license amendment request. The thermal-hydraulic analysis concludes that the ruptured SG overfills and releases more mass to the environment as a result of the combined effects from the plant modifications and the increased operator action times.

As a result, the offsite doses projected at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are higher than previously calculated due to the larger source term released to the environment during overfill. The following tables provide a comparison of re-analysis results. The results provide doses based on the reactor coolant iodine concentration corresponding to (1) a pre-accident iodine spike and (2) a concurrent (accident-initiated) iodine spike:

THYROID DOSE

	Location	A Current FSAR Failed-open ASD Case Value [REM]	B Analysis of Record Overfill Case Value [REM]	C New W Overfill Case Value [REM]	D Regulatory Limit [REM]
Pre-accident	EAB	34.3	24.48	46.2	300
	LPZ	3.43	2.52	4.71	300
Accident Initiated	EAB	22.9	5.64	13.4	300
	LPZ	2.29	0.64	1.43	300

WHOLE BODY DOSE

	Location	A Current FSAR Failed-open ASD Case Value [REM]	B Analysis of Record Overfill Case Value [REM]	C New W Overfill Case Value [REM]	D Regulatory Limit [REM]
Pre-accident	EAB	0.324	0.047	0.362	25
	LPZ	0.0367	0.0056	0.0385	25
Accident Initiated	EAB	0.643	0.035	0.396	25
	LPZ	0.0685	0.0043	0.0424	25

Note that the initial iodine inventory in the reactor coolant system (RCS) and the SGs are assumed to be at the maximum concentrations permitted by the TS. The initial noble gas inventory in the RCS is based on fuel damage equivalent to 1.0 percent failed fuel. In the pre-accident case, the iodine spike occurs just before the SGTR and the RCS iodine inventory is at 60 μCi/gm dose equivalent I-131. In the accident-initiated case, the SGTR event initiates an iodine spike that increases the rate at which iodine is released from the fuel to the RCS (335 times the normal iodine appearance rate).

As shown in the tables above, the SGTR with overfill accident dose radiological consequences are presented as dose to the thyroid and dose to the whole body for locations at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ). Because the increase in dose values between the re-analysis values and the Analysis of Record values are less than 10% of the margin to the regulatory limits, the recalculated dose values represent "minimal increases" in 10 CFR 50.59 context. Therefore, the new values do not require prior NRC approval.

The re-analysis of the radiological consequences uses Regulatory Guide 1.195 source term methodology to calculate the accident radioiodine levels and doses to the thyroid. This involved changing two elements of the Analysis of Record methodology:

(1) The reanalysis uses thyroid DCFs based on the 1979 International Commission on Radiation Protection Publication 30 (ICRP-30), which is a departure from the more conservative ICRP-2 method of evaluation for DCFs used in the Analysis of Record, and

(2) 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 Analysis of Record.

Both of these methodology changes reduce the magnitude of the accident source term and result in lower thyroid 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. In summary, even though the re-analysis results in an insignificant increase in thyroid and whole body doses received at the EAB, all doses remain well below the limits established in the Standard Review Plan (NUREG 0800) for SGTR events.

<u>Small Break Loss-of Coolant Accident (SBLOCA)</u>

The small break LOCA Analysis of Record modeled a MFIV stroke time of 5 seconds. The effect of small changes on the secondary side of the steam generator has a minimal impact on the SBLOCA analysis results. It has been concluded that the proposed change will have a negligible impact on reported peak clad temperatures for the SBLOCA sequence. The current reported peak clad temperature for SBLOCA is 1687 °F. The regulatory limit is 2200 °F.

A SBLOCA event is considered a Condition III event. The SBLOCA was analyzed only taking credit for the TDAFP providing AFW to four Steam Generators. Therefore, the MDAFP ARC valve modification and associated increase in maximum AFW flow has no impact on this accident. A hydraulic analysis of the AFW System using the latest pump surveillance data for the TDAFP found this pump could provide 1054 gpm during a SBLOCA. Assuming a single protection train failure as required, a diversion of up to 15 gpm per actuator, 60 gpm total, of AFW flow could occur during the SBLOCA. This would result in a net flow of 994 gpm (1054 gpm – 60 gpm) to four Steam Generators, which exceeds the required flow of 940 gpm assumed in the accident analysis.

When the single failure assumed during this event is the loss of one diesel generator, all solenoid valves on the MFIV system-medium actuators will function as designed. Therefore, the solenoid valves will provide a complete pressure boundary for the AFW system and no diversion of AFW through the MFIV actuators will occur.

<u>LOCA Mass and Energy Release Analysis</u>

The short-term and long-term LOCA mass and energy releases, which are discussed in FSAR Section 6.2.1.3, do not credit nor model the main feedwater isolation valve closure. An instantaneous isolation is assumed, since quicker isolation produces limiting results based on an increased superheating of the LOCA mass/energy release. The increase of the MFIV closure time to 15 seconds has no adverse effect on these analyses.

Therefore, the current short-term and long-term LOCA mass and energy releases presented in the FSAR remain valid.

The LOCA mass and energy release analysis models auxiliary feedwater flow. Therefore, the MDAFP ARC valve and associated increase in maximum AFW flow would have an impact on this event. The increase in AFW flow cools the SG secondary side and reduces the energy released. The MDAFP ARC valve and associated increase in maximum AFW flow is beneficial to the mitigation of this event.

Main Steam Line Break Mass and Energy Release Analysis

The increase to the MFIV stroke time has the potential to affect the main steam line break (MSLB) mass and energy (M&E) releases inside and outside containment.

Main Steam Line Break Outside Containment

The analysis of the MSLB M&E releases outside containment (FSAR Section 3B.4.2) assumes main feedwater isolation coincident with reactor trip, with no delays associated with instrumentation or valve stroke. This is a conservative assumption for this event. Quicker isolation of main feedwater flow produces more limiting Main Steam Tunnel pressure-temperature results due to minimized total mass addition to the SGs and resultant higher levels of superheat in the blowdown Mass and Energy (M&E) releases (page 2-10 of Reference 8.18). Therefore, an increase in the MFIV closure time will not adversely impact this analysis. The MDAFP ARC valve and associated increase in maximum AFW flow would be beneficial in the mitigation of this event. The increase in AFW flow cools the steam generator secondary side and reduces the energy released.

Main Steam Line Break Inside Containment

The key parameter in the MSLB inside containment analysis affected by the proposed change to MFIV isolation time is steam generator dry-out time. This parameter is addressed in the original pressure-temperature calculations. Although slower valve closure time impacts mass and energy releases in general, the proposed stroke time increase of 10 seconds does not specifically impact the original calculated Mass and Energy (M&E) Releases. The original portion of the analysis is unaffected by the proposed change in MFIV isolation time.

The analysis of the MSLB M&E releases inside containment (FSAR Section 6.2.1.4) for Callaway is limiting at part-power conditions, resulting from a split rupture in a steam line. The analysis supporting the limiting MSLBs for containment response is a generic calculation performed by the NSSS supplier for the Model F steam generator design. The MSLB M&E releases for split ruptures are generic with no specific assumptions regarding time for main feedwater isolation (as well as other critical protection functions). It has been confirmed that the generic assumptions made in the original Mass and Energy Releases analysis bound the Callaway Plant proposed MFIV stroke time of 15 seconds.

Adjustments have been made to the generic M&E release values for specific Callaway Plant conditions. As previously stated, the key parameter affected by longer MFIV isolation time is steam generator dry-out time.

Post-accident steam generator dry-out is defined as the time when flow into the affected generator is equal to flow out of the generator, after the break has occurred. (In order to reach dry-out, the initial inventory must be depleted, break flow is then a function only of flow into the generator. Following dry-out, the magnitude of the break flow is not influenced by the secondary side water inventory). If dry-out occurs after the termination of AFW flow to the faulted steam generator, the mass release rate is set to zero following dry-out. If dry-out occurs prior to AFW termination, the mass release rate is set to the AFW flow rate. The mass release rate is then subsequently set to zero once AFW flow is terminated.

The proposed increase in MFIV stroke time would result in additional main feedwater mass being introduced into the affected steam generator. The additional mass would then be released to containment, which would delay dry-out of the affected generator. This would then provide the potential to lead to higher post-MSLB pressures or temperatures inside containment.

AmerenUE has performed a calculation to quantify the impact of the additional 10 seconds of main feedwater flow to the steam generators following initiation of the accident sequence. This calculation quantified the additional steam generator secondary side mass inventory, following a MSLB inside containment. CONTEMPT (Reference 8.15) runs were then executed to determine the impact of the additional mass on post-MSLB containment pressures and temperatures. These CONTEMPT runs found that the proposed MFIV actuator replacement and associated increase in MFIV stroke time caused no adverse impact on post-MSLB containment pressures and temperatures. The calculated pressure and temperature profiles remain bounded by the analysis envelopes originally calculated by Bechtel.

The MDAFP ARC valve and associated increase in maximum AFW flow affects the MSLB inside containment analysis due to the increased flow to the steam generators. This has the same impact as the delayed isolation of the MFIVs and the impact of this modification is also bounded by the analysis originally calculated by Bechtel.

Impact of the Proposed Change on other Analyses

NSSS Design Thermal Transients

The impact of the proposed change to MFIV isolation time on the NSSS design thermal transients has been evaluated. It has been concluded that the transient profiles and the number of occurrences remain valid for the increase in MFIV isolation time.

Steam Releases Assumed in Radiological Analyses

The Callaway Plant NSSS supplier is responsible for those analyses that establish the steam mass releases used for the radiological consequence analyses for the following accident sequences:

Loss of AC Power
Locked RCP Rotor
Main Steam Line Break
RCCA Ejection

Timing of Main Feedwater isolation is not an input parameter in the analyses that calculated the steam mass release values. Therefore, it has been concluded that the steam releases currently assumed in the radiological analyses for these sequences remain valid. The proposed change in MFIV isolation time and increased AFW flow therefore will not adversely impact the radiological consequence analyses for these sequences.

RISK SIGNIFICANCE

MFIV Actuator Replacement and Increased MFIV Stroke Time

The risk significance of replacing the existing MFIV actuators has been evaluated based on different design, different actuating media, and the increase in the closing stroke time for the valve. The evaluation concludes that the availability of the feedwater isolation function is unchanged for the new actuator design. Although there is a slight decrease in risk of a reactor trip for the new actuator, the decrease is not quantifiable because trip frequency data do not distinguish the causes of the trip.

In addition, based on the simplicity of the new actuator design and based on the extent of the verification and validation to be performed on the modified MSFIS actuation logic software (for MFIV inputs and outputs only), common mode software failures are no more likely with the new actuator design than they are with the current digital design.

In conclusion, use of the new MFIV actuators does not result in an increase in the risk of a reactor trip or an increase in the unavailability of the feedwater isolation function.

The new actuators improve reliability and reduce required maintenance on the MFIVs. Relaxing the required stroke time from 5 to 15 seconds reduces the magnitude of feedwater system pressure transients. The proposed changes do not significantly impact the Callaway core damage frequency nor do they impose risk significant changes to the plant.

MDAFP ARC Valve and Increased Maximum AFW Flow

The MDAFP ARC valves replace the existing swing check valves. Elimination of the existing valves also eliminates the need for the existing flow orifices in each recirculation line. In the unlikely event that the armature that connects recirculation flow control and the lifting check disc were to fail (break), the result would be that the recirculation line of the ARC valve would fail open. In essence, the system would return to what is currently installed at Callaway, with flow through the recirculation line being continuous, although restricted by the bypass pressure reducer orifice of the ARC valve. This proposed change does not impact the Callaway core damage frequency nor impose risk significant changes to the plant.

Use of ICRP 30 Based DCFs and Iodine Spike Model based on Regulatory Guide 1.195 in Re-Analysis of SGTR with Overfill

Incorporating alternate methodologies in the re-analysis of design basis accidents are not assessed for risk significance to core damage. The use of Regulatory Guide 1.195 for determining the source term is considered a methodology that is recognized for its scientific basis and has been approved by the NRC for use by other licensees. The use of the revised DCFs based on ICRP 30 and the revised iodine spike factor are recommended methodologies based on Section 4.1.2 and Appendix E of Regulatory Guide 1.195. There are precedents for use of the revised methods and it is warranted to incorporate the methodology changes into plant analyses. The proposed change does not impact the Callaway core damage frequency nor impose risk significant changes to the plant.

CONCLUSION

Based on the evaluation results and the re-analysis of SGTR with overfill, the proposed increase in MFIV stroke time, the increased MDAFP flow margin, the use of Regulatory Guide 1.195 source term methodology, do not adversely impact plant safety. The proposed amendment is acceptable for continued safe operation. The analyses summarized above demonstrate that the modifications and associated changes do not adversely impact the adequacy of existing plant system design features, plant procedures, or existing analyses for postulated accidents and their consequences. The proposed amendment request is in compliance with all existing regulatory requirements and criteria such that, 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 to the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

This license amendment request proposes to change the Technical Specifications to increase MFIV stroke time from 5 seconds to 15 seconds and to revise the definition of DOSE EQUIVALENT I-131 to include ICRP 30 based DCFs. Other changes include methodology revisions based on the Regulatory Guide 1.195 source term methodology and the re-analysis of the SGTR with overfill accident. AmerenUE has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   MFIV Actuator Replacement and Increased MFIV Stroke Time

As discussed above, the increase in MFIV stroke time does not adversely impact the NSSS design transients evaluated for the Callaway Plant. The increase in MFIV stroke time will result in a slightly longer normal post trip cool down. Although the plant post trip cool down is expected to be slightly longer for the increased MFIV stroke time, the plant response does not significantly deviate from its current evaluated response following a normal reactor trip.

Evaluations assessing the impact of the change in MFIV actuators and the increase in MFIV stroke time on LOCA mass and energy releases; main steamline break mass and energy releases; LOCA and LOCA related transients; non-LOCA transients; LOCA hydraulic forces and steam releases used for radiological consequence calculations were also performed. The increase in isolation time and change in MFIV actuators either do not provide an adverse impact or have no impact. Except for the SGTR with overfill accident, the results presented in the FSAR remain valid. The increase in MFIV stroke time was evaluated for impact on the SGTR with overfill accident. As discussed below, the results from the re-analysis of the SGTR with overfill accident confirm that there is no significant increase in the probability or consequences of an accident previously evaluated.

The replacement of the existing electro-hydraulic MFIV actuators with system-medium actuators and the increase in MFIV stroke time from 5 seconds to 15 seconds will not result in a significant increase in the probability or consequences of an accident previously evaluated.

MDAFP ARC Valve and Increased Maximum AFW Flow

The replacement of the existing MDAFP discharge check valves with the ARC valves results in increased maximum AFW flow to the steam generators. In many accident scenarios the increase in AFW flow to the SGs is beneficial to mitigation of the event. The evaluations above demonstrate that in those accident scenarios where maximum AFW flow is limiting, except for the SGTR with overfill accident, the increase in AFW flow remains bounded by FSAR analyses. The increase in maximum AFW flow was evaluated for impact on the SGTR with overfill accident. As discussed below, the results from the re-analysis of the SGTR with overfill accident confirm that there is no significant increase in the probability or consequences of an accident previously evaluated. The AFW system is not the initiator of any accident and there is no possibility of a significant increase in the probability of an accident or malfunction previously evaluated.

Use of the ARC valve is an enhancement and the associated increase in the maximum AFW flow will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Use of Revised Methods in Re-Analysis of SGTR with Overfill

The reanalysis of the design basis accident for SGTR with overfill does not significantly increase the probability or consequences of an accident previously evaluated. The re-analysis of an accident is not an initiator. The SGTR accident is classified as an ANS Condition IV Event, Limiting Faults, and is only postulated and not expected to occur. The reanalysis activity being evaluated does not change the ANS classification for this design basis event. The re-analysis does provide dose consequences that are minimal increases to the doses in the Analysis of Record.

However, the doses remain well below regulatory limits. In support of this methodology the proposed TS definition for DOSE EQUIVALENT I-131 will allow the use of ICRP 30 based DCFs. Section 4.1.2 and in Appendix E of Regulatory Guide 1.195 find acceptable and recommend the method revisions.

In summary, using the proposed revised methods for the re-analysis of the SGTR with overfill does not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

MFIV Actuator Replacement and Increased MFIV Stroke Time

The change in MFIV actuators and associated increase in MFIV stroke time will not prevent the main feedwater or auxiliary feedwater systems from performing their safety functions. The proposed increase will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the increase. Although the modification does alter the design of the MFIV actuators, it does not prevent the main feedwater or AFW systems from performing their safety functions.

Therefore, the proposed changes do not create a new or different kind of accident from any accident previously evaluated.

MDAFP ARC Valve and Increased Maximum AFW Flow

The new MDAFP ARC valve and associated increase in maximum AFW flow, will not prevent the AFW system from performing its safety function. The proposed increase in AFW system flow margin will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the increase in AFW system flow margin. Although the modification alters the design of the MDAFP discharge check valves, it does not prevent the AFW system from performing its safety functions.

Therefore, the proposed changes do not create a new or different kind of accident from any accident previously evaluated.

Use of Revised Methods in Re-Analysis of SGTR with Overfill

The revision to the Technical Specifications to allow the use of ICRP 30 based DCFs is based on methodologies found acceptable to the NRC and recommended for use as described in Section 4.1.2 of Regulatory Guide 1.195. The reanalysis of the design basis accident for SGTR with overfill and the use of recommended analysis methods acceptable to the NRC does not introduce the possibility of a new accident. Accident re-analysis is not an initiator of any accident and no new failure modes are introduced. In summary, there is no increase in the possibility of an accident of a different type.

Therefore, the proposed changes do not create 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.

MFIV Actuator Replacement and Increased MFIV Stroke Time

The replacement of the MFIV actuator and the associated increase in the MFIV stroke 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 (F_Q), nuclear enthalpy rise hot channel factor (F-delta-H), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power 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 changes do not involve a significant reduction in the margin of safety.

MDAFP ARC Valve and Increased Maximum AFW Flow

The use of the MDAFP ARC valve and the associated increase in AFW system flow margin 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 (F_Q), nuclear enthalpy rise hot channel factor (F-delta-H), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power 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 changes do not involve a significant reduction in the margin of safety.

Use of Revised Methods in Re-Analysis of SGTR with Overfill

Use of revised methods in the re-analysis for the SGTR with overfill accident 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 is no significant impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (F-delta-H), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met. The re-analysis of the SGTR with overfill confirms that both the thermal-hydraulic and radiological consequences are within the regulatory requirements and does not result in a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

CONCLUSION

Based on the above evaluations, AmerenUE concludes that the activities associated with the changes described above present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

MFIV Actuator Replacement and Increased Stroke Time

The regulatory basis for TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)", is the isolation of main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Closure of the MFIVs terminates flow to the steam generators, terminating the event for feedwater line breaks (FLBs) occurring upstream of the MFIVs. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FLBs inside containment and reducing the cooldown effects for SLBs. For FLBs inside containment or SLBs, isolation of main feedwater limits the high energy fluid to the broken loop and provides a path for the addition of auxiliary feedwater to the three intact steam generators.

The MFIVs isolate nonsafety related portions of the feedwater and condensate systems from the safety related portions. In the event of a secondary side pipe rupture inside containment, the valves (1) limit the quantity of high energy fluid that enters containment through the break, and (2) provide an acceptable pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

The safety related functions of the MFIV actuators are to close an MFIV within the specified time frame and to provide an acceptable pressure boundary for the delivery of auxiliary feedwater. A single failure of any active component cannot prevent the actuators from performing their safety functions. Current Technical Specifications provide a specified valve stroke time of less than or equal to 5 seconds. The proposed amendment would increase this time to less than or equal to 15 seconds.

The portion of the feedwater system from the steam generators to the MFIVs is safety related and is required to function following a design basis accident (DBA), and to achieve and maintain the plant in a safe shutdown condition. General Design Criteria (GDC) 2, "Design Bases for Protection against Natural Phenomena", requires that the safety related portion of the feedwater system be protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles.

GDC 3, "Fire Protection", and GDC 4, "Environmental and Dynamic Effects Design Bases", requires that the safety related portion of the feedwater system be designed to remain functional after a safe shutdown earthquake (SSE), and to perform its intended function following postulated hazards of fire, internal missiles, or pipe break.

GDC 34, "Residual Heat Removal", requires that safety functions of the feedwater system can be performed assuming a single active component failure coincident with the loss of offsite power. GDC 34 also requires that for a main feedwater line break upstream of the MFIVs (outside containment), the feedwater system is designed to prevent the blowdown of any one steam generator and to provide a path for the addition of auxiliary feedwater for reactor cooldown under emergency shutdown conditions.

The change in mass for the new main feedwater isolation valve operators has been evaluated to ensure the feedwater piping system does not exceed ASME Code Class 2 piping design limits.

Considering the postulated plant and external conditions, the main feedwater isolation valves are designed and assumed to close in 15 seconds or less upon receipt of an automatic or manual close signal.

MDAFP ARC Valve and Associated Increase in Maximum AFW Flow

Requirements for the AFW system are identified in FSAR Section 10.4.9.1. Based on the modifications and proposed changes, the applicable regulatory requirement for the AFW system is to provide feedwater (in conjunction with the condensate storage tank (nonsafety-related and not credited for accident mitigation) or essential service water system, which is credited for accident mitigation) to maintain sufficient steam generator level to ensure heat removal from the reactor coolant system in order to achieve a safe shutdown following a main feedwater line break, a main steam line break, or an abnormal plant situation requiring shutdown. The AFW system is capable of delivering full flow when required, after detection of any accident requiring auxiliary feedwater.

Use of Regulatory Guide 1.195 Source Term Methodology in Re-Analysis of SGTR with Overfill

The regulatory requirements for use of source term methodology in the re-analysis of the SGTR with overfill apply to the reactor coolant iodine concentration corresponding to (1) pre-accident iodine spike and (2) a concurrent (accident-initiated) iodine spike. Criteria for acceptable analyses of radiological consequences require meeting the acceptance criteria of the following:

- The relevant dose guidelines provided in 10 CFR Part 100
- Standard Review Plan (NUREG 0800) Sections 15.6.3
- General Design Criterion (GDC) – 19 as it relates to mitigating the radiological consequences of SGTR accidents.
- Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Plants.

6.0 ENVIRONMENTAL CONSIDERATION

AmerenUE 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. AmerenUE has evaluated the proposed changes and has determined that the changes do 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.

As discussed above, the proposed changes do not involve a significant hazards consideration and the analysis demonstrates that the consequences from the postulated accidents are well within the 10 CFR 100 limits. Accordingly, the proposed changes 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 PRECEDENTS

The proposed amendment request associated with the MFIV actuator replacement is specific to the Callaway Plant. AmerenUE does not reference a precedent for increasing the MFIV stroke time based on the replacement of MFIV actuators.

Use of the ARC valve in the MDAFP discharge line is not a new application for the valve design. Similar ARC valves have been installed at other plants (such as McGuire and South Texas) in similar applications and with acceptable results. Further, a similar functioning valve is used at Callaway as the start-up main feed pump auto recirculation control valve.

Use of the alternate source term methodology in the re-analysis of the SGTR with overfill accident has precedents for this application. The methodology is recognized by the nuclear industry as having a better scientific basis and has been approved by the NRC for use by Diablo Canyon (NRC Safety Evaluation (SE) for Diablo Canyon Units 1 & 2 Operating License Amendment No. 156 to Facility Operating License No. DPR-80 and Amendment No. 156 to Facility Operating License No. 82, dated February 20, 2003).

8.0 REFERENCES

8.1 Callaway Plant Technical Specification, Section 1.1, Definitions, DOSE EQUIVALENT I-131.

8.2 Callaway Plant Technical Specification Surveillance Requirement 3.7.3.1.

8.3 FSAR Section 10.4.7, Condensate and Feedwater System.

8.4 FSAR Section 10.4.9, Auxiliary Feedwater System.

8.5 FSAR Section 6.2, Containment Systems

8.6 FSAR Section 15, Accident Analysis

8.7 ASME, Boiler and Pressure Vessel Code, Section XI.

8.8 FSAR, Table 7.3-14, NSSS Instrument Operating Conditions for Isolation Functions.

8.9 ULNRC-01518, dated May 27, 1987, SGTR Analysis.

8.10 SLNRC 86-01, January 8, 1986, SGTR Analysis.

8.11 Hollingsworth, S.D. and Wood, D. C., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226, Revision 1, (Proprietary), January 1978, and WCAP-9227, Revision 1, (Non-Proprietary), January 1978.

8.12 1979 International Commission on Radiation Protection Publication 30 (ICRP-30)

8.13 Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" (ICRP-2).

8.14 SLNRC 81-44, dated June 8, 1981, "Reliability Analysis of the SNUPPS Auxiliary Feedwater System".

8.15 NUREG/CR 0255, "CONTEMPT-LT/028-A Computer Program For Predicting Containment Pressure-Temperature Response To A Loss-of-Coolant Accident", March 1979.

8.16 Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors.

8.17 Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factor for Inhalation, Submersion, and Ingestion."

8.18 "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," WCAP-10961, Revision 1, October 1985.

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ATTACHMENT 3

MARKUP OF TECHNICAL SPECIFICATION PAGES

1.1 Definitions (continued)

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with ~~Specification 5.6.5.~~ Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." INSERT A
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

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(continued)

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INSERT A

or those derived from the data provided in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers, " 1979.

OL 1226, FOR INFORMATION ONLY
NO CHANGES

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs)

LCO 3.7.3 Four MFIVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIVs inoperable.	A.1 Close MFIV. AND A.2 Verify MFIV is closed.	4 hours Once per 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.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 closure time of each MFIV is $\leq$ 5 seconds. 15	In accordance with the Inservice Testing Program
SR 3.7.3.2	-------NOTE------- Only required to be performed in MODES 1 and 2. ------------------ Verify each MFIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

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ATTACHMENT 4

RETYPED TECHNICAL SPECIFICATION PAGES

1.1 Definitions (continued)

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or those derived from the data provided in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979.
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------------NOTE------------------------- Only required to be performed in MODES 1 and 2. --- Verify the closure time of each MFIV is ≤ 15 seconds.	In accordance with the Inservice Testing Program
SR 3.7.3.2	-------------------------NOTE------------------------- Only required to be performed in MODES 1 and 2. --- Verify each MFIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

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ATTACHMENT 5

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

(for information only)

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater ~~Isolation Valves~~ (MFIVs)

System-medium

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 MFRVs function to control feedwater flow to the SGs.

The MFIV is a 14-inch gate valve with a ~~dual-redundant hydraulic~~ actuator. The assumed single active failure of one of the redundant MFIV ~~actuators~~ will not prevent the MFIV from closing.

actuation trains

TSB CN 01-009

Closure of the MFIVs terminates 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 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.

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.

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.

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 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.

The MFIV actuators consist of two separate ~~pneumatic-hydraulic power~~ trains each receiving an actuation signal from one of the redundant

System-medium actuation

(continued)

TSB CN 01-009

BASES

BACKGROUND (continued)

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 non-safety-related portions from the safety-related portion of the system, such as, for auxiliary feedwater addition.

A description of the MFIVs and MFRVs is found in the FSAR, Section 10.4.7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Credit is taken in accident analysis for the MFIVs to close on demand. The function of the MFRVs and associated bypass valves as discussed in the accident analysis is to provide a diverse backup function to the MFIVs for the potential failure of an MFIV to close even though the MFRVs are located in the non-safety-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.

actuation trains,

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, bypass valves, and the SGFP trip function 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).

TSB CN 01-009

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 will also isolate the nonsafety related portions from the safety related portions of the system.

This LCO requires that four MFIVs be OPERABLE. The MFIVs are considered OPERABLE when isolation times are within limits when given a fast close signal and they are capable of closing on an isolation actuation signal.

(continued)

BASES

LCO ~~(continued)~~

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~~.

TSB CN 01-009

APPLICABILITY

The MFIVs must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1,2, and 3, the MFIVs are required to be OPERABLE to 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 MFIVs are closed they are performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs are not required to mitigate the effects of a feedwater or steamline break in these MODES.

need space TSB CN 01-009

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

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 hours. When these valves are closed, they are performing their required safety function.

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.

actuation trains
TSB CN 01-009

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.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

15

This SR verifies that the closure time of each MFIV is ≤ ~~5~~ seconds from each ~~actuator~~ train when tested pursuant to the Inservice Testing Program. 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. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power.

actuation

TSB 01-009

The Frequency for this SR is in accordance with the Inservice Testing Program.

This test is 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.

SR 3.7.3.2

Each actuation train must be tested separately.

TSB CN 01-009

This SR verifies that each MFIV is capable of closure on an actual or simulated actuation signal. The manual fast close handswitch in the Control Room provides an acceptable 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 MFIV 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.

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.7.5.2 (continued)

The required differential pressure for the AFW pumps when tested in accordance with the Inservice Testing Program is:

a. The acceptance criteria for the MDAFPs have been calculated using a limiting performance curve. The acceptance criteria, given as a table below, have been determined based on the Loss of Normal Feedwater (LONF) or Loss of Non-emergency AC Power (LOAC) events.

MOTOR DRIVEN PUMPS
ACCEPTANCE CRITERIA
(using performance curve)

Insert A

Recirc. Flow (gpm)	Diff. Pressure (psid)
~~≤75~~	~~≥1529~~
~~≤80~~	~~≥1533~~
~~≤85~~	~~≥1537~~
~~≤90~~	~~≥1540~~
~~≤95~~	~~≥1544~~
~~≤100~~	~~≥1548~~
~~≤105~~	~~≥1552~~
~~≤110~~	~~≥1555~~
~~≤115~~	~~≥1559~~

INSERT B

TSB CN 01-009

~~b. Turbine Driven Pump ≥1610 psid at ≥120 gpm~~

~~This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.~~

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

(continued)

TSB CN 01-009

INSERT A

Recirc. Flow (gpm)	Diff. Pressure (psid)
≥ 130	≥ 1543
≥ 150	≥ 1542
≥ 160	≥ 1539
≥ 170	≥ 1537

INSERT B

b. The acceptance criteria for the TDAFP has been calculated using a limiting performance curve. The acceptance criteria, given as a table below, have been determined based on the Small Break Loss of Coolant Accident (SBLOCA) event.

TURBINE DRIVEN PUMP
ACCEPTANCE CRITERIA
(using performance curve)

Recirc. Flow (gpm)	Diff. Pressure (psid)
≥ 120	≥ 1628
≥ 140	≥ 1626

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

ULNRC- 04592

ATTACHMENT 6

PROPOSED CALLAWAY FSAR CHANGES

(for information only)

FSAR CN 01-036
System - Medium

TABLE 3.9(B)-16 (Sheet 3)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AE-FV-039	Feedwater	~~Hydraulic~~	14.0	Gate/2	Open	3
AE-FV-040	Feedwater	~~Hydraulic~~	14.0	Gate/2	Open	3
AE-FV-041	Feedwater	~~Hydraulic~~	14.0	Gate/2	Open	3
AE-FV-042	Feedwater	~~Hydraulic~~	14.0	Gate/2	Open	3
AE-V-120	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-121	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-122	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-123	Feedwater	ΔP	14.0	Check/2	NA	3
AE-V-124	Feedwater	ΔP	4.0	Check/2	NA	4
AE-V-125	Feedwater	ΔP	4.0	Check/2	NA	4
AE-V-126	Feedwater	ΔP	4.0	Check/2	NA	4
AE-V-127	Feedwater	ΔP	4.0	Check/2	NA	4
AL-HV-005	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-006	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-007	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-008	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-009	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-010	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-011	Auxiliary Feedwater	Electric Motor	4.0	Globe/2	Open	4
AL-HV-012	Auxiliary Feedwater	Air Cylinder	4.0	Globe/2	Open	4
AL-HV-030	Auxiliary Feedwater	Electric Motor	6.0	Butterfly/3	Closed	4

TABLE 3.9(B)-16 (Sheet 4)

VALVE LOCATOR NO.	SYSTEM	ACTUATED BY	SIZE (IN.)	TYPE/CLASS	NORMAL POSITION	BASIS
AL-HV-031	Auxiliary Feedwater	Electric Motor	6.0	Butterfly/3	Closed	4
AL-HV-032	Auxiliary Feedwater	Electric Motor	8.0	Butterfly/3	Closed	4
AL-HV-033	Auxiliary Feedwater	Electric Motor	8.0	Butterfly/3	Closed	4
AL-HV-034	Auxiliary Feedwater	Electric Motor	8.0	Gate/3	Open	6
AL-HV-035	Auxiliary Feedwater	Electric Motor	8.0	Gate/3	Open	6
AL-HV-036	Auxiliary Feedwater	Electric Motor	10.0	Gate/3	Open	6
AL-V-001	Auxiliary Feedwater	ΔP	10.0	Check/3	NA	6
AL-V-002	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	6
AL-V-003	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	6
AL-V-006	Auxiliary Feedwater	ΔP	6.0	Check/3	NA	4
AL-V-009	Auxiliary Feedwater	ΔP	6.0	Check/3	NA	4
AL-V-012	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	4, 6
AL-V-015	Auxiliary Feedwater	ΔP	8.0	Check/3	NA	4, 6
AL-V-029	Auxiliary Feedwater	ΔP	2.0	Check/3	NA	4
AL-FV-030 ~~AL-V-030~~	Auxiliary Feedwater	ΔP	6.0	Check/recirc/3 ~~Check/3~~	NA	4
AL-V-033	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-036	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-041	Auxiliary Feedwater	ΔP	2.0	Check/3	NA	4
AL-FV-042 ~~AL-V-042~~	Auxiliary Feedwater	ΔP	6.0	Check/recirc/3 ~~Check/3~~	NA	4
AL-V-045	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4
AL-V-048	Auxiliary Feedwater	ΔP	4.0	Check/2	NA	4

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safety analysis will be controlled in accordance with AmerenUE's software control procedures.

3. 2.3 Training and Qualification of Licensee Personnel

Training and qualification of personnel performing Safety Analysis calculations will be accomplished in accordance with AmerenUE's Engineering Support Personnel training program.

4. 2.4 Comparison Calculations

Comparison and benchmark calculations will be performed in accordance with approved procedural controls. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

5. 2.5 Quality Assurance and Change Control

Safety Analysis calculations will be performed in accordance with the AmerenUE OQAP, which implements 10CFR50, Appendix B Criterion III. Computer codes used for safety analysis will be controlled in accordance with AmerenUE's software control procedures.

INSERT A1

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FSAR CN 03-023

INSERT A1

<u>REGULATORY GUIDE 1.195</u> <u>REVISION 0</u> <u>DATED 5/03</u>

Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors

DISCUSSION:

The recommendations of this regulatory guide are met as described in the analysis of FSAR design basis accidents and their radiological consequences.

effects on equipment in both compartments. The Case 1b peak temperatures exceed the qualification requirements previously established for equipment in these rooms based on a 1.4 ft^2 break without superheat; therefore, the surface temperature response of the equipment was evaluated to demonstrate the proper operation of equipment before it was calculated to be heated above its qualified temperature. Failure modes and effects analyses were also employed, when required, to evaluate certain electrical circuits and determine equipment performance.

The surface temperature response was calculated for various representative pieces of equipment and components which may be required following an MSLB in the steam tunnel. The most severe room conditions (those for the break compartment) and the flow characteristics from the FLUD calculation results were used in the calculation of the time dependent equipment surface temperatures.

At any given time, the greater of four times the Uchida condensing heat transfer rate (based on the compartment air to steam mass ratio) or the convective heat transfer rate was used to evaluate the transient surface temperature response of the selected equipment. The Hilpert correlation, for flow past an object in a fluid stream, with consideration for system turbulence, was used to calculate the convective heat transfer coefficient. In the evaluation of the heat transfer coefficient for a component, the characteristic velocity was taken as the time dependent average velocity of the flow between the east and west rooms of the steam tunnel. The film properties used in the evaluation of the Hilpert equation were based on the state of the air and steam in the stream.

As only the outside casing of equipment was modelled~~, the lumped-capacity~~ method was used to calculate the surface temperature response of the equipment. This approach is justified by the thinness and the high thermal conductivity of the modelled external casings. For the ~~main steam~~ isolation valve ~~and main feedwater isolation valve~~ terminal blocks located in terminal boxes on the valve actuators and the solenoid valves mounted on the actuators, more detailed, two-dimensional thermal lag analyses were performed to determine equipment temperatures. Details of the equipment modelling are provided in References 8 and 9.

The surface temperature analysis showed that, with certain exceptions, the equipment surface temperature did not exceed the qualified temperature limits prior to the time when a steam line isolation signal (SLIS) or feedwater isolation signal (FWIS) was initiated. Surface temperature values at the time of SLIS are provided in Reference 12. The exceptions are the main steam pressure transmitter instrument cable, and air-operated valve control cable. A failure modes and effects analysis showed that failure of these cables will not affect the ability to safely shut down the plant following a main steam line break as the affected equipment either fails safe or alternative capability is provided. Further analysis showed that the failure of equipment subsequent to its actuation will not result in equipment repositioning or in misleading the plant operators. Reference 9 provides additional details concerning failure modes and effects analysis. Discussions of qualification margins may be found in References 7 and 9.

Insert A

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INSERT A

Surface temperatures at the time of a FWIS are lower since feedwater isolation was conservatively assumed to occur coincident with reactor trip (see Reference 14).

6.2.1.4.3 Containment Response Analysis

The COPATTA computer code (Ref. 1), which is discussed in Section 6.2.1.1.3, was used to determine the containment responses following the postulated main steam line breaks. The CONTEMPT computer code (Ref. 11), which is also discussed in Section 6.2.1.1.3, was used in June, 2000 to assess the impact of reduced containment cooler performance on containment pressure and temperature results by re-analyzing containment response following limiting main steam line breaks. The following assumptions were made to obtain these responses.

6.2.1.4.3.1 Initial Conditions

The initial containment conditions are the same as those used in the containment response analysis for the postulated reactor coolant system pipe ruptures (see Table 6.2.1-5).

6.2.1.4.3.2 Mass and Energy Release Data

The tables contained in Reference 7 present the mass and energy release data used to determine the containment pressure-temperature responses for the spectrum of breaks analyzed. The basis for these tables is provided in Reference 6, along with a discussion of the methods used to modify the data to reflect the specific plant design. The specific plant design input which was assumed is provided for each case in Table 6.2.1-57. Tables 6.2.1-57A and 6.2.1-57B provide the mass and energy release data for the cases which resulted in the highest temperature and pressure, respectively.

The rate of auxiliary feedwater addition represents the maximum runout flowrate to a fully depressurized steam generator. The value given for mass added by feedwater pumping assumes that no reduction in feedwater pump turbine speed occurs following a MSLB and prior to main feedwater isolation. Feedwater isolation for the full and partial double-ended ruptures is dependent on signals generated by the primary protection system, which results in isolation times ranging between 7.0 and 8.9 seconds for these cases. Feedwater isolation for the split breaks was based on the time required to reach the containment High-1 pressure setpoint which generates an SIS, which then results in the feedwater isolation signal. Determination of feedwater flowrates prior to isolation assumed that the feedwater control valve in the broken loop goes wide open while those in the intact loops remain in their pre-break positions. INSERT B FSAR CN 01-036

6.2.1.4.3.3 Containment Pressure-Temperature Results

Figures 6.2.1-79 through 6.2.1-82 provide curves of the resultant containment pressure-temperature transients for the cases producing the highest peak containment pressure and temperature. Table 6.2.1-58 summarizes the results of all the cases analyzed and indicates the times at which dryout occurs and the various containment pressure setpoints are reached. The sequence of events following a postulated main

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It should be noted the main feedwater isolation time was increased from 5 seconds to 15 seconds by Amendment XXX to the Operating License in support of the plant design change which installed system-medium actuators on the Main Feedwater Isolation Valves. This 10 second increase in the isolation time results in an increase in the amount of main feedwater added to the steam generators for each break case. Calculations performed in support of the design change to install the system-medium actuators determined that the previously evaluated steam generator dry-out time and total mass release values remain bounding.

TABLE 6.2.1-57 SPECIFIC PLANT DESIGN INPUT FOR MSLB ANALYSIS

Case	1	2	3	4	5	6
Initial steam generator inventory, lbm	124,000	124,000	124,000	134,000	134,000	134,000
Initial steam pressure, psia	983	983	983	1,013	1,013	1,013
Mass added by feedwater pumping, lbm ****	19,814	17,604	35,863	19,000	17,000	32,000
Mass added by feedwater flashing, lbm	10,432	10,432	10,432	10,748	10,748	10,748
Steam piping blowdown inventory, ft^3	800	800	800	800	800	800
Auxiliary feedwater addition rate, lbm/hr	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5
Main steam line isolation time, sec	7.3	8.9	68.1	7.0	8.7	63.6
Main feedwater line isolation time, sec ****	7.3	8.9	23.1	7.0	8.7	20.9
Termination of auxiliary feedwater addition, sec	1800***	1800***	1800***	1800***	1800***	1800***

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TABLE 6.2.1-57 (Sheet 2)

Case	7	8	9	10	11	12
Initial steam generator inventory, lbm	145,000	145,000	145,000	156,000	156,000	156,000
Initial steam pressure, psia	1,036	1,036	1,036	1,063	1,063	1,063
Mass added by feedwater pumping, lbm ****	19,000	16,000	35,000	20,000	16,000	35,000
Mass added by feedwater flashing, lbm	11,020	11,020	11,020	11,508	11,508	11,508
Steam piping blowdown inventory, ft^3	800	800	800	800	800	800
Auxiliary feedwater addition rate, lbm/hr	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5
Main steam line isolation time, sec	7.0	8.5	65.8	7.0	8.1	81.1
Main feedwater line isolation time ****	7.0	8.5	21.3	7.0	8.1	24.0
Termination of auxiliary feedwater addition, sec	1800***	1800***	1800***	1800***	1800***	1800***

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TABLE 6.2.1-57 (Sheet 3)

Case	13	14	15	16
Initial steam generator inventory, lbm	168,000	168,000	168,000	124,000
Initial steam pressure, psia	1,106	1,106	1,106	983
Mass added by feedwater pumping, lbm ****	5,000	5,000	12,000	19,814
Mass added by feedwater flashing, lbm	12,500	12,500	12,500	10,432
Steam piping blowdown inventory, ft^3	800	800	800	9300 *
Auxiliary feedwater addition rate, lbm/hr	6.9×10^5	6.9×10^5	6.9×10^5	6.9×10^5
Main steam line isolation time, sec	7.0	7.5	193.8	7.3 **
Main feedwater line isolation time, sec ****	7.0	7.5	36.0	7.3
Termination of auxiliary feedwater addition, sec	1800***	1800***	1800***	1800***

* In addition, a steam release rate of 10 lbm/sec was added to the total blowdown to account for releases from unisolated branch lines off the main steam piping.

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TABLE 6.2.1-57 (Sheet 4)

** Intact loop isolation valves only. Broken loop isolation valve is assumed to fail open.

*** Termination of auxiliary feedwater at 600 seconds is assumed for the re-analysis performed in June, 2000 to assess a reduction in containment cooler performance. Since results for inputs above are bounding, the Table values are conservative and have not been revised. Also refer to Section 6.2.1.4.

**** INSERT C

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INSERT C

All main Feed isolation times listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feed Water Isolation Valves.

This would result in an increase in the amount of main feedwater added to the steam generators for each break case discussed in this table. Calculations performed in support of the system-medium actuator design determined that the previously evaluated steam generator dry-out and total mass release values remain bounding.

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TABLE 6.2.1-58 SUMMARY OF RESULT FOR MSLB CONTAINMENT PRESSURE-TEMPERATURE ANALYSIS** , ***

Case No.	Power Level (%)	Break Size ft²	Break Type	Max Press @ Time (psig @ Time)	Max Vapor Temp @ Time (F @ sec)	Dryout Time (sec)	6 psig @ Time (sec)	20 psig @ Time (sec)	30 psig @ Time (sec)
1*	102	Full	D.E.	39.6 @ 1800	348.0 @ 110	186	2.13	11.86	78.5
2*	102	0.60	D.E.	36.8 @ 1800	338.0 @ 310	355.6	10.7	120.2	275.0
3*	102	0.80	Split	45.6 @ 1800	336.0 @ 204	529	16.6	61.6	169.0
4	75	Full	D.E.	38.5 @ 1800	313.9 @ 150	182	2.48	29.30	116.0
5	75	0.55	D.E.	37.8 @ 1800	363.9 @ 370	496	17.30	160.92	335.0
6	75	0.84	Split	45.0 @ 1800	384.9 @ 135	676	14.38	57.12	104.0
7	50	Full	D.E.	39.5 @ 1800	308.1 @ 145	213	2.61	25.15	114.0
8	50	0.45	D.E.	39.7 @ 1800	361.2 @ 455	719	24.81	213.94	425.0
9	50	0.80	Split	47.8 @ 1800	382.6 @ 145	1475	14.78	59.33	114.0
10	25	Full	D.E.	40.7 @ 1800	311.5 @ 160	236	2.51	28.58	125.0
11	25	0.33	D.E.	43.9 @ 1800	357.0 @ 685	1433	46.59	348.58	650.0
12	25	0.66	Split	48.1 @ 1800	380.0 @ 195	908	17.54	74.57	160.0
13	HS	Full	D.E.	39.3 @ 1800	312.1 @ 180	243	2.44	37.97	149.0
14	HS	0.2	D.E.	36.6 @ 2670	328.2 @ 1400	2672	115.10	766.20	1380.0
15	HS	0.40	Split	47.1 @ 2170	363.4 @ 510	2165	29.40	187.30	490.0
16*	102	Full	D.E.	34.6 @ 186	352.0 @ 45	186	2.13	8.46	51.4

* Cases 1-3 and 16 revised per ULNRC-1471 dated 3/31/87.

** Cases 6, 12 and 14 were re-analyzed in June, 2000 assuming termination of auxiliary feedwater at 600 seconds, 8% revaporization of blowdown condensate and a reduction in containment cooler performance. Since the results above are bounding, the Table values are conservative and have not been revised. Also refer to Section 6.2.1.4.

*** ← INSERT D →

FSAR CN 01-036

FSAR CN 01-036

INSERT D

The main feedwater isolation times listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feed Water Isolation Valves.

This would result in an increase in the amount of main feedwater added to the steam generators for each break case discussed in this table. Calculations performed in support of the system-medium actuators design determined that the previously evaluated steam generator dry-out and total mass release values remain bounding.

TABLE 6.2.1-59 SEQUENCE OF EVENTS FOR CASE 12

PEAK CALCULATED CONTAINMENT PRESSURE CASE FOR MSLB**

Time (sec)	Event
0.0	Break occurs, blowdown from all four steam generators
17.5	Containment pressure setpoint for isolation of main feedwater lines reached (6 psig)
24.0 ***	Main feedwater line isolation valves closed
60.0	Air coolers start
74.6	Containment pressure setpoint for isolation of main steam lines reached (20 psig)
81.1	Main steam line isolation valves closed, blowdown from broken loop steam generator and unisolated steam piping only
160.0	Containment pressure setpoint for actuation of containment sprays reached (30 psig)
190.0*	Containment sprays start
190.0	Peak containment vapor temperature of 380°F is reached.
908.0	Dryout occurs, blowdown equals auxiliary feedwater addition rate
1800.0	Auxiliary feedwater addition is terminated
1800.0	Peak containment pressure of 48.1 psig is reached

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* The 30-second delivery time considers the simultaneous opening of the containment isolation valve and the starting of the containment spray pumps.

** Case 12 was re-analyzed in June, 2000 assuming termination of auxiliary feedwater at 600 seconds, 8% revaporization of blowdown condensate and a reduction in containment cooler performance. Since the results above are bounding, the Table values are conservative and have not been revised. Also refer to Section 6.2.1.4.

*** ← INSERT D →

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The main feedwater isolation times listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feed Water Isolation Valves.

This would result in an increase in the amount of main feedwater added to the steam generators for each break case discussed in this table. Calculations performed in support of the system-medium actuators design determined that the previously evaluated steam generator dry-out and total mass release values remain bounding.

TABLE 6.2.1-60 SEQUENCE OF EVENTS FOR CASE 6

PEAK CALCULATED CONTAINMENT TEMPERATURE FOR MSLB**

Time (sec)	Event
0.0	Break occurs, blowdown from all steam generators
14.4	Containment pressure setpoint for isolation at main feedwater lines reached (6 psig)
20.9 ***	Main feedwater line isolation valves closed
57.1	Containment pressure setpoint for isolation of main steam lines reached (20 psig)
60.0	Air coolers start
63.6	Main steam line isolation valves closed, blowdown from broken loop steam generator and unisolated steam piping only
103.3	Containment pressure setpoint for actuation of containment sprays reached (30 psig)
133.3*	Containment sprays start
140.0	Peak containment vapor temperature of 384.9°F is reached
676.0	Dryout occurs, blowdown equals auxiliary feedwater addition rate
1800.0	Auxiliary feedwater addition is terminated
1800.0	Peak containment pressure of 45.0 psig is reached

FSAR CN 01-036

* The 30-second delivery time considers the simultaneous opening of the containment isolation valve and the starting of the containment spray pumps.

** Case 6 was re-analyzed in June, 2000 assuming termination of auxiliary feedwater at 600 seconds, 8% revaporization of blowdown condensate and a reduction in containment cooler performance. Since the results above are bounding, the Table values are conservative and have not been revised. Also refer to Section 6.2.1.4.

*** ← INSERT DI →

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INSERT D1

The main feedwater isolation times listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feed Water Isolation Valves.

This would result in an increase in the amount of main feedwater added to the steam generators for the break case discussed in this table. Calculations performed in support of the system-medium actuators design determined that the previously evaluated steam generator dry-out and total mass release values remain bounding.

TABLE 6.2.1-61 MASS AND ENERGY BALANCE

FSAR CN 01-036

PEAK CALCULATED CONTAINMENT PRESSURE CASE FOR MSLB*, **

ENERGY BALANCE (x 10^6 Btu)

	Initial 0 sec	Peak Pressure 1800 sec	End of Blowdown 1800 sec
Containment atmosphere	6.5	275.2	275.2
Containment sump	0.0	247.9	247.9
Total Energy	6.5	523.1	523.1
Initial energy	6.5	6.5	6.5
Energy added by blowdown	0.0	730.3	730.3
Energy added by sprays	0.0	48.2	48.2
Energy removed by air coolers	0.0	94.4	94.4
Energy removed by heat sinks	0.0	167.4	167.4
Total Energy	6.5	523.2	523.2

MASS BALANCE (x 10^3 lbm)

	Initial 0 sec	Peak Pressure 1800 sec	End of Blowdown 1800 sec
Containment atmosphere	6.2	248.1	248.1
Containment sump	0.0	1073.1	1073.1
Total Mass	6.2	1321.2	1321.2
Initial Mass	6.2	6.2	6.2
Mass added by blowdown	0.0	606.8	606.8
Mass added by sprays	0.0	708.2	708.2
Total Mass	6.2	1321.2	1321.2

* Case 6 was re-analyzed in June, 2000 assuming termination of auxiliary feedwater at 600 seconds, 8% revaporization of blowdown condensate and a reduction in containment cooler performance. Since the results above are bounding, the Table values are conservative and have not been revised. Also refer to Section 6.2.1.4.

** INSERT D1

FSAR CN 01-036

FSAR CN 01-036

INSERT D1

The main feedwater isolation times listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feed Water Isolation Valves.

This would result in an increase in the amount of main feedwater added to the steam generators for the break case discussed in this table. Calculations performed in support of the system-medium actuators design determined that the previously evaluated steam generator dry-out and total mass release values remain bounding.

FSAR CN 01-036

TABLE 6.2.1-62 MASS AND ENERGY BALANCE

PEAK CALCULATED CONTAINMENT TEMPERATURE FOR MSLB*, **

ENERGY BALANCE (x 10^6 Btu)

	Initial 0 sec	Peak Pressure 1800 sec	End of Blowdown 1800 sec
Containment atmosphere	6.5	294.5	294.5
Containment sump	0.0	246.1	246.1
Total Energy	6.5	540.6	540.6
Initial energy	6.5	6.5	6.5
Energy added by blowdown	0.0	754.5	754.5
Energy added by sprays	0.0	46.5	46.5
Energy removed by air coolers	0.0	95.6	95.6
Energy removed by heat sinks	0.0	171.4	171.4
Total Energy	6.5	540.5	540.5

MASS BALANCE (x 10^3 lbm)

	Initial 0 sec	Peak Pressure 1800 sec	End of Blowdown 1800 sec
Containment atmosphere	6.2	265.3	265.3
Containment sump	0.0	1052.1	1052.1
Total Mass	6.2	1317.4	1317.4
Initial Mass	6.2	6.2	6.2
Mass added by blowdown	0.0	626.9	626.9
Mass added by sprays	0.0	684.3	684.3
Total Mass	6.2	1317.4	1317.4

* Case 12 was re-analyzed in June, 2000 assuming termination of auxiliary feedwater at 600 seconds, 8% revaporization of blowdown condensate and a reduction in containment cooler performance. Since the results above are bounding, the Table values are conservative and have not been revised. Also refer to Section 6.2.1.4.

** INSERT D1

FSAR CN 01-036

FSAR CN 01-036

INSERT D1

The main feedwater isolation times listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feed Water Isolation Valves.

This would result in an increase in the amount of main feedwater added to the steam generators for the break case discussed in this table. Calculations performed in support of the system-medium actuators design determined that the previously evaluated steam generator dry-out and total mass release values remain bounding.

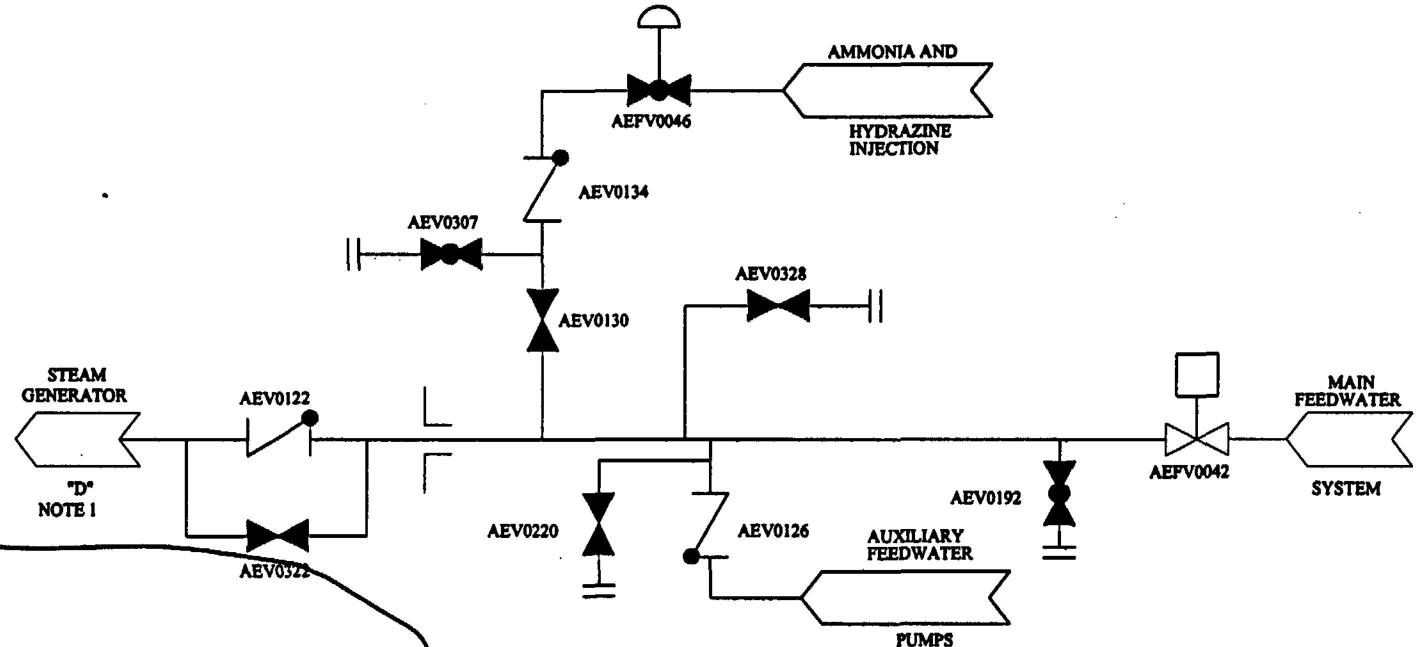
FSAR CN 01-036

VALVE NO.	LINE/ VALVE SIZE,IN.	INSIDE/ OUTSIDE CONT.	NORMAL FLOW DIRECTION	VALVE TYPE	VALVE OPERATOR	POWER SOURCE	PRIMARY ACTUATION SIGNAL	SECONDARY ACTUATION SIGNAL	MAXIMUM CLOSURE TIME(SEC)	VALVE POSITION				
										NORMAL	SHUTDOWN	POWER FAILURE	POST ACCIDENT PRIMARY	POST ACCIDENT SECONDARY
AEFV0042	14/14	OUTSIDE	IN	GATE	HYDRAULIC (2)	1,4	FWIS	NONE	~~5~~ 15	OPEN	OPEN	CLOSED	CLOSED	N/A
AEV0126	4/4	OUTSIDE	IN	CHECK	N/A	N/A	N/A	N/A	N/A	CLOSED	N/A	N/A	OPEN	N/A
AEFV0046	1/1	OUTSIDE	IN	GLOBE	AIR	4	FWIS	NONE	5	CLOSED	CLOSED	CLOSED	CLOSED	N/A
AEV0307	3/4 / 3/4	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0220	1/1	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0328	3/3	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0192	1/1	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
VALVES BELOW NOT SHOWN IN SKETCH														
AEV0714	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0716	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A

ENGINEERED SAFETY FEATURE SYSTEM YES NO	☒ YES ☐ NO
FLUID CONTAINED:	WATER
LENGTH OF PIPING TO OUTERMOST ISOLATION VALVE:	21.8 FT.
APPLICABLE GDC NO.	NONE

GENERAL COMMENTS:

THE CONTAINMENT PENETRATIONS ASSOCIATED WITH THE STEAM GENERATORS ARE NOT SUBJECT TO GDC-56, SINCE THE CONTAINMENT BARRIER INTEGRITY IS NOT BREACHED. THE BOUNDARY OR BARRIER AGAINST FISSION PRODUCT LEAKAGE TO THE ENVIRONMENT IS THE INSIDE OF THE STEAM GENERATOR TUBES AND THE OUTSIDE OF THE LINES EMANATING FROM THE STEAM GENERATOR SHELLS.



NOTE 1: THE STEAM GENERATOR AND LINES EMANATING FROM THE STEAM GENERATOR SHELLS ARE NOT SHOWN HERE. REFERENCE FIG. 10.4-6 FOR DETAILS OF THE PIPING, INSTRUMENTATION AND VALVE CONFIGURATION/ IDENTIFICATION.

Note 2: System - medium utilizing process fluid

APPENDIX J REQUIREMENT

- TYPE A ☒
- B ☐
- C ☐
- NONE ☐

FSAR CN 01-036

REFERENCE SECTION 10.4.7

CONTAINMENT PENETRATION NO. P-5
DESCRIPTION:
MAIN FEEDWATER LINE

CONTAINMENT PENETRATIONS
FIGURE 6.2.4-1
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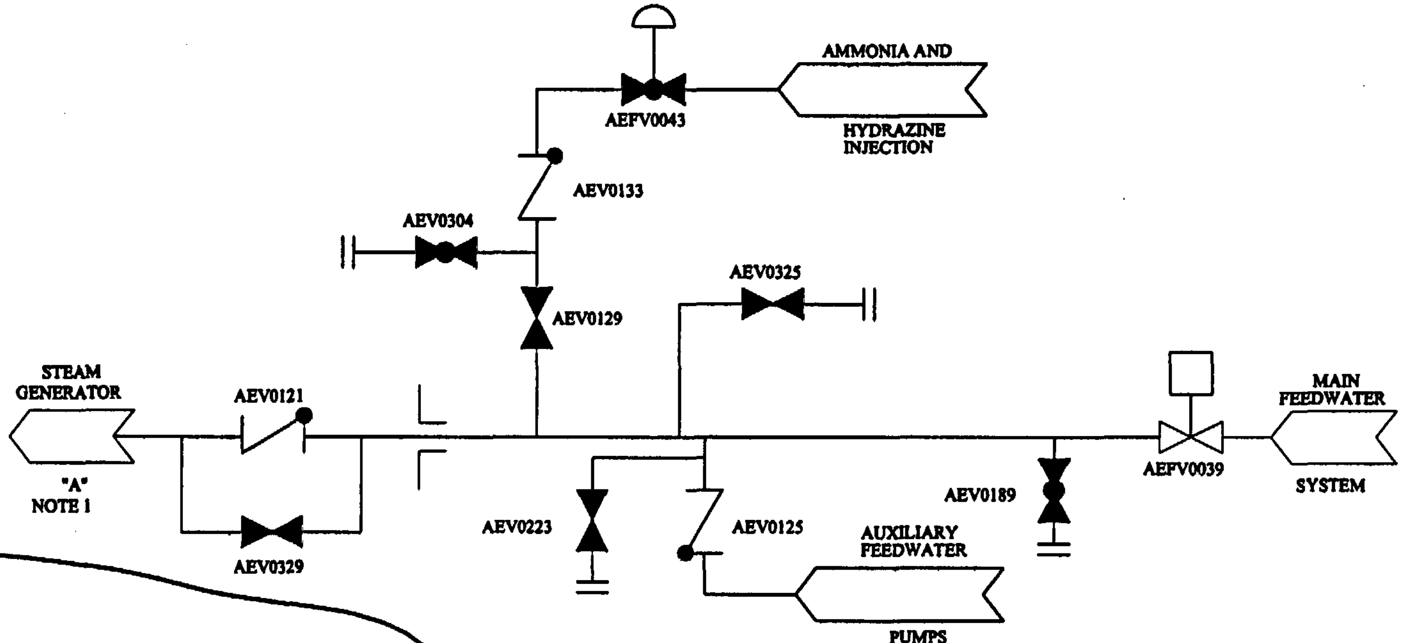
FSAR CN 01-036

VALVE NO.	LINE/ VALVE SIZE,IN.	INSIDE/ OUTSIDE CONT.	NORMAL FLOW DIRECTION	VALVE TYPE	VALVE OPERATOR	POWER SOURCE	PRIMARY ACTUATION SIGNAL	SECONDARY ACTUATION SIGNAL	MAXIMUM CLOSURE TIME(SEC.)	VALVE POSITION				
										NORMAL	SHUTDOWN	POWER FAILURE	POST ACCIDENT PRIMARY	POST ACCIDENT SECONDARY
AEFV0039	14/14	OUTSIDE	IN	GATE	HYDRAULIC (2)	1,4	FWIS	NONE	15	OPEN	OPEN	CLOSED	CLOSED	N/A
AEV0125	4/4	OUTSIDE	IN	CHECK	N/A	N/A	N/A	N/A	N/A	CLOSED	N/A	N/A	OPEN	N/A
AEFV0043	1/1	OUTSIDE	IN	GLOBE	AIR	1	FWIS	NONE	5	CLOSED	CLOSED	CLOSED	CLOSED	N/A
AEV0223	1/1	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0325	3/3	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0189	1/1	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0304	3/4 / 3/4	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
VALVES BELOW NOT SHOWN IN SKETCH														
AEV0702	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0704	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A

ENGINEERED SAFETY FEATURE SYSTEM	YES ☒ NO ☐
FLUID CONTAINED:	WATER
LENGTH OF PIPING TO OUTERMOST ISOLATION VALVE:	21.1 FT.
APPLICABLE GDC NO.	NONE

GENERAL COMMENTS:

THE CONTAINMENT PENETRATIONS ASSOCIATED WITH THE STEAM GENERATORS ARE NOT SUBJECT TO GDC-56, SINCE THE CONTAINMENT BARRIER INTEGRITY IS NOT BREACHED. THE BOUNDARY OR BARRIER AGAINST FISSION PRODUCT LEAKAGE TO THE ENVIRONMENT IS THE INSIDE OF THE STEAM GENERATOR TUBES AND THE OUTSIDE OF THE LINES EMANATING FROM THE STEAM GENERATOR SHELLS.



NOTE 1: THE STEAM GENERATOR AND LINES EMANATING FROM THE STEAM GENERATOR SHELLS ARE NOT SHOWN HERE. REFERENCE FIG. 10.4-6 FOR DETAILS OF THE PIPING, INSTRUMENTATION AND VALVE CONFIGURATION/ IDENTIFICATION.

Note 2: System-medium utilizing process fluid

FSAR CN 01-036

APPENDIX J REQUIREMENT

TYPE A ☒
B ☐
C ☐
NONE ☐

REFERENCE SECTION 10.4.7

CONTAINMENT PENETRATION NO. P-6
DESCRIPTION:
MAIN FEEDWATER LINE

CONTAINMENT PENETRATIONS
FIGURE 6.2.4-1
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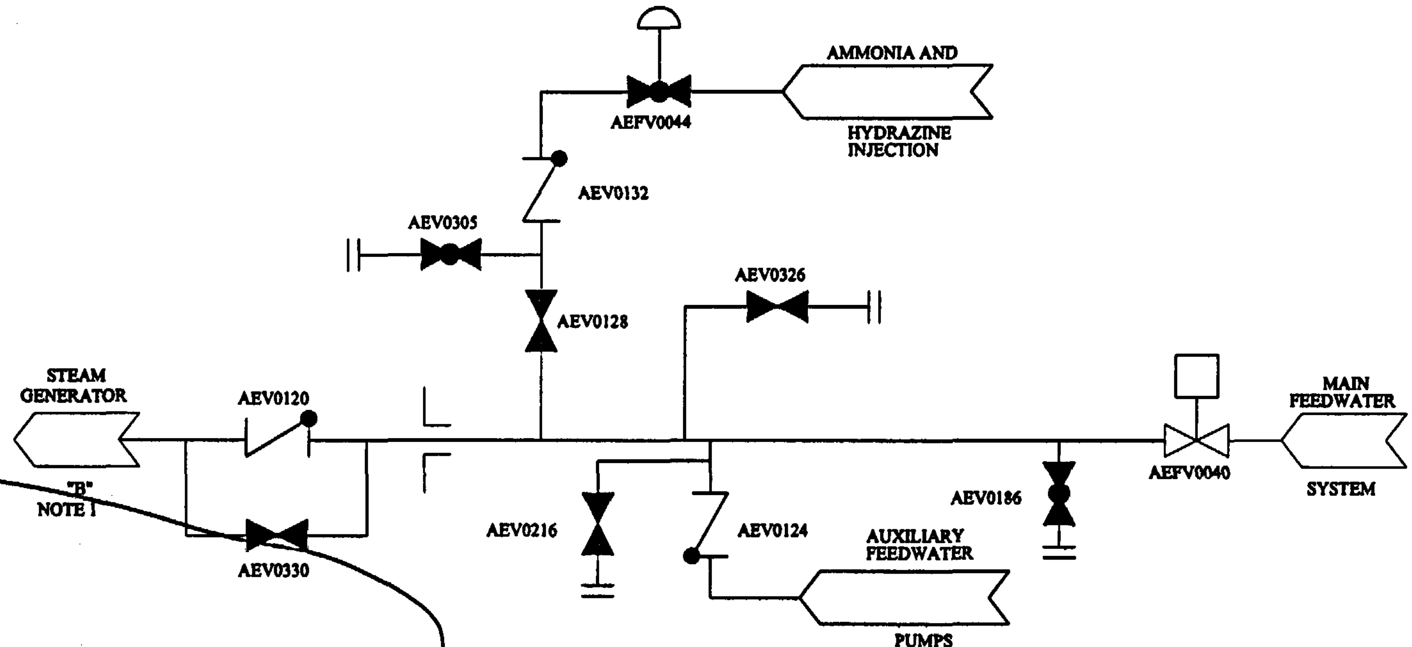
VALVE NO.	LINE/ VALVE SIZE,IN.	INSIDE/ OUTSIDE CONT.	NORMAL FLOW DIRECTION	VALVE TYPE	VALVE OPERATOR	POWER SOURCE	PRIMARY ACTUATION SIGNAL	SECONDARY ACTUATION SIGNAL	MAXIMUM CLOSURE TIME(SEC.)	VALVE POSITION				
										NORMAL	SHUTDOWN	POWER FAILURE	POST ACCIDENT PRIMARY	POST ACCIDENT SECONDARY
AEFV0040	14/14	OUTSIDE	IN	GATE	HYDRAULIC (2)	1,4	FWIS	NONE	~~8~~ 15	OPEN	OPEN	CLOSED	CLOSED	N/A
AEV0124	4/4	OUTSIDE	IN	CHECK	N/A	N/A	N/A	N/A	N/A	CLOSED	N/A	N/A	OPEN	N/A
AEFV0044	1/1	OUTSIDE	IN	GLOBE	AIR	4	FWIS	NONE	5	CLOSED	CLOSED	CLOSED	CLOSED	N/A
AEV0305	3/4 / 3/4	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0326	3/3	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0216	1/1	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0186	1/1	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
VALVES BELOW NOT SHOWN IN SKETCH														
AEV0706	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0708	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A

ENGINEERED SAFETY FEATURE SYSTEM YES ☒ NO ☐	
FLUID CONTAINED:	WATER
LENGTH OF PIPING TO OUTERMOST ISOLATION VALVE:	21.1 FT.
APPLICABLE GDC NO.	NONE

GENERAL COMMENTS:

THE CONTAINMENT PENETRATIONS ASSOCIATED WITH THE STEAM GENERATORS ARE NOT SUBJECT TO GDC-56, SINCE THE CONTAINMENT BARRIER INTEGRITY IS NOT BREACHED. THE BOUNDARY OR BARRIER AGAINST FISSION PRODUCT LEAKAGE TO THE ENVIRONMENT IS THE INSIDE OF THE STEAM GENERATOR TUBES AND THE OUTSIDE OF THE LINES EMANATING FROM THE STEAM GENERATOR SHELLS.



NOTE 1: THE STEAM GENERATOR AND LINES EMANATING FROM THE STEAM GENERATOR SHELLS ARE NOT SHOWN HERE. REFERENCE FIG. 10.4-6 FOR DETAILS OF THE PIPING, INSTRUMENTATION AND VALVE CONFIGURATION/ IDENTIFICATION.

Note 2: system-medium utilizing process fluid

APPENDIX J REQUIREMENT

TYPE A ☒
B ☐
C ☐
NONE ☐

FSAR CN 01-036

REFERENCE SECTION 10.4.7

CONTAINMENT PENETRATION NO. P-7 DESCRIPTION: MAIN FEEDWATER LINE
CONTAINMENT PENETRATIONS FIGURE 6.2.4-1 PAGE 7 OF 84 REV. OL-10 11/98

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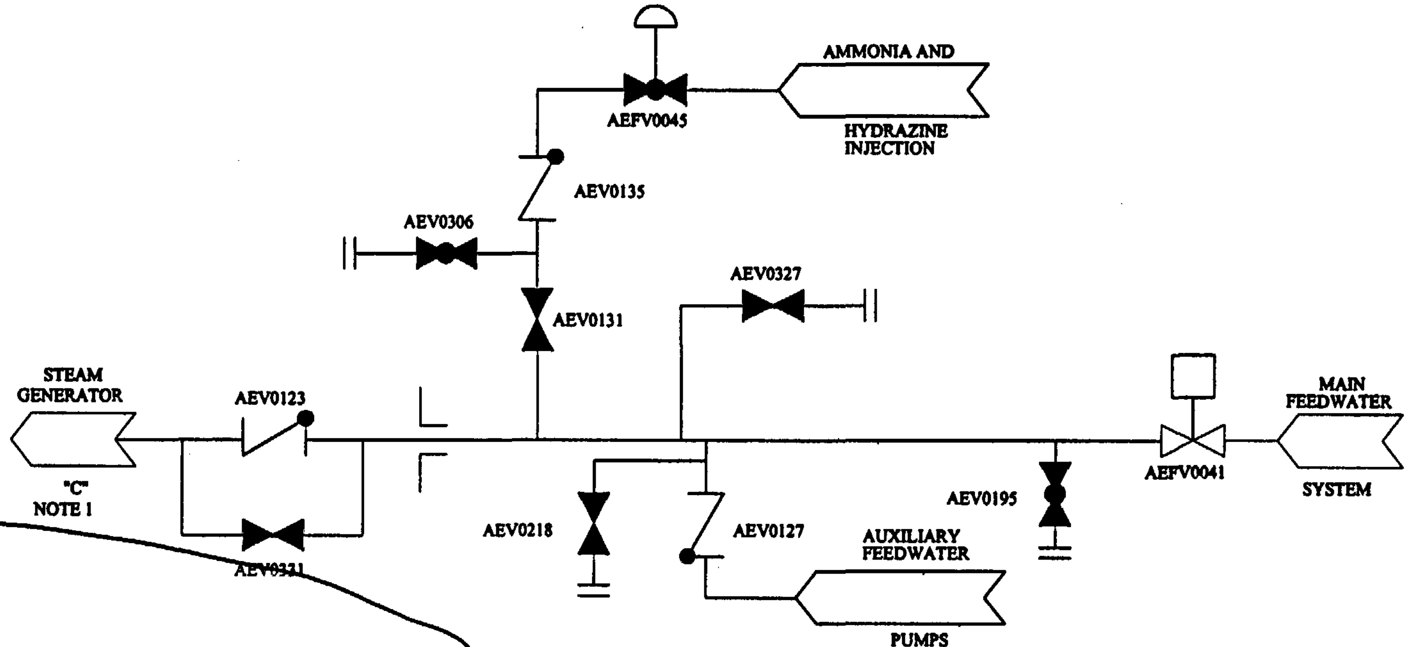
VALVE NO.	LINE/ VALVE SIZE,IN.	INSIDE/ OUTSIDE CONT.	NORMAL FLOW DIRECTION	VALVE TYPE	VALVE OPERATOR	POWER SOURCE	PRIMARY ACTUATION SIGNAL	SECONDARY ACTUATION SIGNAL	MAXIMUM CLOSURE TIME(SEC.)	VALVE POSITION				
										NORMAL	SHUTDOWN	POWER FAILURE	POST ACCIDENT	
													PRIMARY	SECONDARY
AEFV0041	14/14	OUTSIDE	IN	GATE	HYDRAULIC (2)	1,4	FWIS	NONE	~~8~~ 15	OPEN	OPEN	CLOSED	CLOSED	N/A
AEV0127	4/4	OUTSIDE	IN	CHECK	N/A	N/A	N/A	N/A	N/A	CLOSED	N/A	N/A	OPEN	N/A
AEFV0045	1/1	OUTSIDE	IN	GLOBE	AIR	1	FWIS	NONE	5	CLOSED	CLOSED	CLOSED	CLOSED	N/A
AEV0306	3/4 / 3/4	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0218	1/1	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0327	3/3	OUTSIDE	N/A	GATE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0195	1/1	OUTSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
VALVES BELOW NOT SHOWN IN SKETCH														
AEV0710	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A
AEV0712	3/4 / 3/4	INSIDE	N/A	GLOBE	MANUAL	N/A	N/A	N/A	N/A	CLOSED	CLOSED	N/A	CLOSED	N/A

ENGINEERED SAFETY FEATURE SYSTEM YES NO	☒ YES ☐ NO
FLUID CONTAINED:	WATER
LENGTH OF PIPING TO OUTERMOST ISOLATION VALVE:	24.2 FT.
APPLICABLE GDC NO.	NONE

GENERAL COMMENTS:

THE CONTAINMENT PENETRATIONS ASSOCIATED WITH THE STEAM GENERATORS ARE NOT SUBJECT TO GDC-56, SINCE THE CONTAINMENT BARRIER INTEGRITY IS NOT BREACHED. THE BOUNDARY OR BARRIER AGAINST FISSION PRODUCT LEAKAGE TO THE ENVIRONMENT IS THE INSIDE OF THE STEAM GENERATOR TUBES AND THE OUTSIDE OF THE LINES EMANATING FROM THE STEAM GENERATOR SHELLS.



NOTE 1: THE STEAM GENERATOR AND LINES EMANATING FROM THE STEAM GENERATOR SHELLS ARE NOT SHOWN HERE. REFERENCE FIG. 10.4-6 FOR DETAILS OF THE PIPING, INSTRUMENTATION AND VALVE CONFIGURATION/ IDENTIFICATION.

Note 2: System-medium utilizing process fluid

APPENDIX J REQUIREMENT

- TYPE A ☒
- B ☐
- C ☐
- NONE ☐

FSAR CN 01-036

REFERENCE SECTION 10.4.7

CONTAINMENT PENETRATION NO. P-8
DESCRIPTION:
MAIN FEEDWATER LINE

CONTAINMENT PENETRATIONS
FIGURE 6.2.4-1
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TABLE 7.1-1 INSTRUMENTATION SYSTEMS IDENTIFICATION

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Safety-Related Systems or Categories	Designer: Westinghouse	Designer: Bechtel	Similar To Plant: Comanche Peak	Similar To Plant: W. B. McGuire and Watts Bar	Similar To Plant: Other	Remarks
1. Reactor trip system	X		X	X		
2. Engineered safety feature actuation systems						
a. Main steam and feedwater isolation	X	X	~~X~~	~~X~~		New (see 7.3.7)
b. Containment combustible gas control		X			Millstone Unit 2	
c. Containment purge isolation		X				New (see 7.3.2)
d. Fuel building ventilation isolation		X			—	New (see 7.3.3)
e. Control room ventilation isolation		X			—	New (see 7.3.4)
f. Auxiliary feedwater supply	X	X	X	X	—	New supply configuration (see 7.3.6)
g. NSSS ESFAS	X		X			
3. Systems required for safe shutdown						
a. Hot standby	X	X	X	X		
b. Cold shutdown	X	X	X	X		
c. Safe shutdown from outside control room	X	X	X		—	New (see 7.4.3)
4. Safety-related display instrumentation						
a. Reactor trip system	X		X	X		
b. Engineering safety feature actuation systems	X	X	X	X		
c. Systems required for safe shutdown	X	X	X	X		
5. Other instrumentation systems required for safety						
a. Instrumentation and control power supply system		X			Trojan	
b. Residual heat removal system isolation valve interlocks	X		X			
c. Refueling interlocks	X		X			

7.3.7 MAIN STEAM AND FEEDWATER ISOLATION

7.3.7.1 Description

The signals that initiate automatic closure of the main steam and feedwater isolation valves are generated in the ESFAS ~~described in Section 7.3.8. The logic diagrams for the generation of these signals are shown in Figure 7.2-1 (Sheets 8 and 13). The remainder of this section concentrates on the non-Westinghouse portion of the main steam and feedwater isolation system (MSFIS).~~

instrument

The main steam and feedwater isolation valves are operated by hydraulic actuators. The actuators are powered by compressed air accumulators, which are controlled by electrically operated solenoid valves. Each valve has two actuators. Each actuator is controlled from a separate Class IE electrical system, and each is capable of closing the valve independently of the other.

INSERT E

The non-Westinghouse MSFIS consists of two independent Class IE actuation trains. Within each train, three Programmable Logic Controllers (PLCs) produce a 2 out of 3 logic configuration for each actuation relay per valve. The use of the same software in the PLCs in each train can produce the possibility of a Common Mode Software Failure (CMSF). Consequently, a diverse backup means to fast close the main steam isolation valves through the use of an Emergency Override Panel and Fast Close toggle switches is included in each train to mitigate the consequences of the CMSF.

7.3.7.1.1 System Description

a. Initiating circuits

The main steam and feedwater isolation valves close automatically upon receipt of an automatic close signal from the Westinghouse solid state protection system (SSPS). Manual operation is also provided.

INSERT F

b. Logic

~~See drawings referenced in Section 1.7. In addition to the manual and automatic trip modes of operation, manual controls are provided for the slow opening or closing of each valve, for checking the accumulator pressures, and for a valve operational check that closes the valve 10 percent and then opens it again.~~

INSERT G

c. Bypass

See Section 7.3.8.

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d. Interlocks

FSAR CN 01-036

INSERT E

The main feedwater isolation valves are operated by system-medium actuators. The actuators are powered by the system-medium, which is controlled by electrically operated solenoid valves. Each main feedwater isolation valve has six solenoid valves, three in each actuation train. Each actuation train is powered from a separate Class 1E electrical system and is capable of closing the valve independent of the opposite actuation train.

INSERT F

Two manual Fast Close switches, one per each actuation train, are provided for the main steam isolation valves. Either switch will close all four associated valves.

Two manual Fast Close switches are provided for the feedwater isolation valves. Each switch has the capability to actuate both actuation trains associated with all four valves. Train isolation at the switches is assured by fire retardant sleeving on the wire going to the switches, and by qualification testing that was performed on the switches. This feature is provided to conserve feedwater that would be lost to the condenser in the event of a single train actuation. Refer to Table 7.1-5 Position 5.

INSERT G

In addition to the manual and automatic trip modes of operation, manual controls are provided for the slow opening or closing, and for checking the accumulator pressures for each main steam isolation valve. For the main feedwater isolation valves, in addition to the manual and automatic trip modes of operation, manual controls are provided for opening and closing the valves.

~~See Section~~ 7.3.8.

e. Redundancy

For the main steam isolation valves, two

Two complete actuation systems are provided for each valve operator. Each system is capable of closing the valve as required, regardless of the state of the other.

INSERT H

f. Diversity

See Section 7.3.8 for a discussion of diversity with regard to the automatic actuation signal.

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g. Actuated devices

The actuated devices are the main steam and feedwater isolation valves.

h. Supporting systems

The system makes use of the Class IE dc power system and of the compressed air system.

i. Portions of the system not required for safety

main steam isolation

One operator for each valve includes provisions for opening or slow closing the valve. Instrumentation is provided for measuring the accumulator pressures. Neither of these provisions is required for safety.

INSERT I

7.3.7.1.2 Design Bases

The design bases for the main steam and feedwater isolation actuation system are provided in Section 7.3.8. The design bases for the remainder of the main steam and feedwater isolation system are that the system isolates the main steam and feedwater when required, and that no single failure can prevent any valve from performing its required function. See Section 7.3.8 for additional discussion.

In addition, Section 7.3.1.1.2 is applicable to the control system components.

7.3.7.1.3 Drawings

See Figures 7.2-1 (Sheet 8), 7.3-2, and 7.3-3. Other drawings pertaining to this system are included in the introductory material for this section.

7.3.7.2 Analysis

a. Conformance to NRC general design criteria

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INSERT H

For the main feedwater isolation valves, two complete actuation trains are provided for each actuator. Each actuation train consists of three solenoid valves and is capable of closing the main feedwater isolation valve regardless of the state of the opposite actuation train.

INSERT I

For the main feedwater isolation valves, each actuation train includes provisions to 'Normal Close' the valves, while both actuation trains are required to remotely 'Open' the valves. These functions are not required for safety. The vent lines downstream of the safety related rupture disk, which include manual isolation valves, along with the actuation position indication, are also not required for safety.

See Section 7.3.8.

b. Conformance to IEEE Standard 279-1971

The design of the valve control system conforms to the applicable requirements of IEEE Standard 279-1971, as listed and discussed in Section 7.3.1.2, except that the system is automatically actuated. The setpoints are provided in the Callaway Technical Specifications.

c. Conformance to NRC regulatory guides

See Section 7.3.8.

d. Failure modes and effects analysis

See ~~Section 7.3.8.~~ Table 10.4-7.

e. Periodic testing

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The valve control system includes provisions for verifying the proper operation of the electronic logic circuits, For the main steam isolation valves, it also includes provisions for checking the accumulator pressure in each actuator. The frequency of actuation system testing is provided in the Callaway Technical Specifications. The mechanical system testing provisions are given in Technical Specification 3.7 and FSAR Section S 10.3.4 and 10.4.7.4.

Note that each valve can be closed within the appropriate time limit by either actuator side. Testing is administratively controlled to ensure that both sides of a given actuator will not be set to "TEST" mode simultaneously.

7.3.8 NSSS ENGINEERED SAFETY FEATURE ACTUATION SYSTEM

7.3.8.1 Description

The Westinghouse solid state protection system (SSPS) consists of two parts: the reactor trip system (RTS), which is described in Section 7.2, and the engineered safety feature actuation system (ESFAS), which is described here. The ESFAS monitors selected plant parameters and, if predetermined safety limits are exceeded, transmits signals to logic matrices sensitive to combinations indicative of primary or secondary system boundary ruptures (Condition III or IV events). When certain logic combinations occur, the system sends actuation signals to the appropriate engineered safety feature components. The ESFAS meets the requirements of GDCs 13, 20, 21, 22, 23, 24, 25, 27, 28, 34, 35, 37, 38, 40, 41, 43, 44, 46, 54, 55, and 56.

7. Feedwater isolation (see Section 10.4.7)

8. Ventilation isolation valves and damper actuator (see Section 6.4)

9. Steam line isolation valve actuators (see Section 7.3.7 and Section 10.3)

10. Containment spray pump and valve actuators (see Section 6.2.2)

If an accident is assumed to occur coincident with a loss of offsite power, the engineered safety feature loads must be sequenced onto the diesel generators to prevent overloading them. This sequence is discussed in Section 8.3. The design meets the requirements of GDC-35.

i. Support systems

The following systems are required for support of the engineered safety features:

1. Essential service water system - heat removal (see Section 9.2.1)

2. Component cooling water system - heat removal (see Section 9.2)

3. Electrical power distribution systems (see Section 8.3)

4. Essential HVAC systems (see Section 9.4)

Table 7.3-12 provides a list of the auxiliary support ESF systems.

j. Portion of system not required for safety

The system produces annunciator, status light, and computer input signals to indicate individual channel status. The system provides signals to the reactor trip annunciators for sequence of events indication, and indicates the condition of blocks and permissives. Semiautomatic testing features are provided for on-line testing. All monitoring for the testing is at the protection system cabinets. Equipment used to accomplish these functions is isolated from the protection functions and is not required for the safety of the plant. Section 7.3.7.1.1.i discusses individual steam and feedwater isolation control switches ~~(slow close)~~ that are not required for safety.

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7.3.8.1.2 Design Bases

The functional diagrams presented in Figure 7.2-1 (Sheets 5, 6, 7, and 8) provide a graphic outline of the functional logic associated with requirements for the ESFAS.

TABLE 7.3-14 (Sheet 2)

No.	Functional Unit	No. of Channels	No. of Channels to Trip
	f. Manual	2*	1*

* ~~Manual actuation of either train closes all main feedwater isolation valves or all main steamline isolation and bypass valves. It is also possible to operate these valves individually. However, those controls are provided for convenience only (slow close switches), and they do not meet the requirements of safety-related controls.~~

Note 1: The feedwater line will isolate on low T_{avg} only in conjunction with reactor trip (P-4). This feedwater isolation signal may be bypassed for normal reactor startups and shutdowns.

Note 2: The feedwater line will isolate on reactor trip in conjunction with low T_{avg} (see Note 1), high-high steam generator level, or safety injection.

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Manual actuation of either switch closes all main feedwater isolation valves or all main steam isolation and bypass valves. It is also possible to operate these valves with individual switches. However, those controls are provided for normal operation only.

TABLE 9.3-2 SAFETY-RELATED PNEUMATICALLY OPERATED VALVES

Valve Number	Description	Safe Position	Failure Mode on Loss of Air Supply	Safety Function	Notes
AB-HV-05	Loop 2 Steam Supply to AFW Pump Turbine	Open	Open	Admit steam to AFW pump turbine and secondary side pressure boundary isolation	
AB-HV-06	Loop 3 Steam Supply to AFW Pump Turbine	Open	Open		
AB-HV-11	Main Steam Iso. Valve Loop 4	Closed	NA	Secondary side pressure boundary isolation	3
AB-HV-14	Main Steam Iso. Valve Loop 1	Closed	NA		3
AB-HV-17	Main Steam Iso. Valve Loop 2	Closed	NA		3
AB-HV-20	Main Steam Iso. Valve Loop 3	Closed	NA		3
AB-HV-12	Main Steam Iso. Bypass Valve Loop 4	Closed	Closed	Secondary side pressure boundary isolation and steam line warmup	
AB-HV-15	Main Steam Iso. Bypass Valve Loop 1	Closed	Closed		
AB-HV-18	Main Steam Iso. Bypass Valve Loop 2	Closed	Closed		
AB-HV-21	Main Steam Iso. Bypass Valve Loop 3	Closed	Closed		
AB-HV-48	Loop 2 Steam Supply to AFW Turbine Bypass	Closed	Closed	Secondary side pressure boundary isolation and steam line keep warm	
AB-HV-49	Loop 3 Steam Supply to AFW Turbine Bypass	Closed	Closed		
AB-LV-07	Main Steam Line Drain Valve Loop 3	Closed	Closed	Secondary side pressure boundary isolation and condensate drain	
AB-LV-08	Main Steam Line Drain Valve Loop 2	Closed	Closed		
AB-LV-09	Main Steam Line Drain Valve Loop 1	Closed	Closed		
AB-LV-10	Main Steam Line Drain Valve Loop 4	Closed	Closed		
AB-PV-01	Steam Gen. A Atm. Relief Valve	Closed	Closed	Secondary side pressure boundary isolation, secondary side heat removal, and pressure relief	2
AB-PV-02	Steam Gen. B Atm. Relief Valve	Closed	Closed		2
AB-PV-03	Steam Gen. C Atm. Relief Valve	Closed	Closed		2
AB-PV-04	Steam Gen. D Atm. Relief Valve	Closed	Closed		2
AE-FV-39	Feedwater Iso. Valve Loop 1	Closed	NA	Secondary side pressure boundary isolation	~~3~~ 4
AE-FV-40	Feedwater Iso. Valve Loop 2	Closed	NA		~~3~~ 4
AE-FV-41	Feedwater Iso. Valve Loop 3	Closed	NA		~~3~~ 4
AE-FV-42	Feedwater Iso. Valve Loop 4	Closed	NA		~~3~~ 4
AE-FV-43	Steam Gen. A Chemical Control	Closed	Closed	Secondary side pressure boundary isolation and chemistry control	
AE-FV-44	Steam Gen. B Chemical Control	Closed	Closed		
AE-FV-45	Steam Gen. C Chemical Control	Closed	Closed		
AE-FV-46	Steam Gen. D Chemical Control	Closed	Closed		

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TABLE 9.3-2 (Sheet 4)

Valve Number	Description	Safe Position	Failure Mode on Loss of Air Supply	Safety Function	Notes
EP-HV-8879C	Acc. Tank C to SIS Test Line Iso. Valve	Closed	Closed	System pressure boundary isolation	
EP-HV-8879D	Acc. Tank D to SIS Test Line Iso. Valve	Closed	Closed	System pressure boundary isolation	
EP-HV-8880	Ctmt. Iso. Valve - N_2 Supply to Accum.	Closed	Closed	Containment isolation	
FC-FV-310	AFP Steam Trap Isolation Valve	Closed	Closed	Condensate removal	
GT-HZ-04	Ctmt. Iso. Valve - Ctmt. Mini Purge	Closed	Closed	Containment isolation	
GT-HZ-05	Ctmt. Iso. Valve - Ctmt. Mini Purge	Closed	Closed	Containment isolation	
GT-HZ-06	Ctmt. Iso. Valve - Ctmt. Large Vol.	Closed	Closed	Containment isolation	
GT-HZ-07	Ctmt. Iso. Valve - Ctmt. Large Vol.	Closed	Closed	Containment isolation	
GT-HZ-08	Ctmt. Iso. Valve - Ctmt. Large Vol.	Closed	Closed	Containment isolation	
GT-HZ-09	Ctmt. Iso. Valve - Ctmt. Large Vol.	Closed	Closed	Containment isolation	
GT-HZ-11	Ctmt. Iso. Valve - Ctmt. Mini Purge	Closed	Closed	Containment isolation	
GT-HZ-12	Ctmt. Iso. Valve - Ctmt. Mini Purge	Closed	Closed	Containment isolation	
HB-HV-7126	Ctmt. Iso. Valve - RCDT to Waste Gas Comp.	Closed	Closed	Containment isolation	
HB-HV-7136	Ctmt. Iso. Valve - RCDT to Recy. Holdup Tank	Closed	Closed	Containment isolation	
HB-HV-7150	Ctmt. Iso. Valve - RCDT to Recy. Holdup Tank	Closed	Closed	Containment isolation	
HB-HV-7176	Ctmt. Iso. Valve - RCDT to Waste Gas Comp.	Closed	Closed	Containment isolation	
KA-FV-29	Ctmt. Iso. Valve - Inst. Air Line	Closed	Closed	Containment isolation	
LF-FV-96	Ctmt. Iso. Valve - Sump to Floor Drain Tank	Closed	Closed	Containment isolation	

NOTES:

(1) Provided with backup compressed gas supply to open for safety functions

(2) Provided with backup compressed gas supply to modulate valve as required during cooldown from hot shutdown condition to cold shutdown

(3) Each MSIV ~~and MFIV~~ is provided with single-failure-proof (redundant) hydraulic actuators. Each hydraulic cylinder is driven by a separate pressurized nitrogen storage container. On loss of electrical control power the MSIVs fail as-is ~~and the MFIVs fail closed~~.

(4) Each MFIV is provided with redundant actuation trains. On loss of electrical control power, the MFIVs Fail closed.

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TABLE 9.3-11 SERVICE GAS REQUIREMENTS

Component Serviced with Nitrogen	Service Gas Function
Safety injection accumulator tanks	Cover gas, source of potential energy
Pressurizer relief tank	Cover gas
Volume control tank	Purge gas (during shutdown)
Spent resin tanks	Sluice spent resins to solid radwaste system
Gas decay tanks	Maintenance during shutdown
Feedwater heaters	Purge and cover gas during layup
Steam generator (shell side)	Purge and cover gas during layup
Auxiliary steam generator and reboiler	Purge and cover during layup
Chilled water expansion tank	Cover gas
Chemical addition tanks	Cover gas
Electrical penetration assemblies	Testing
Steam generator blowdown system	Chemical mixing in steam generators
Hydrogen recombiners	Purge gas
~~Back up compressed~~ gas system accumulators	Source of potential energy
~~FWIV &~~ MSIV accumulators	Cover gas, source of potential energy (from 6 494 SCF local cylinders in the turbine building)
Condensate storage tank	Purge gas

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A.23.7 Analysis

A.23.7.1 Fire Suppression

A fire in this area will be detected and alarmed by the detection system. The fire can be extinguished manually, using the portable extinguishers and/or the hose station in Room 1506. The safe shutdown instrumentation in this area has watertight enclosures.

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The hydraulic fluid in the actuators of the main steam ~~and feedwater~~ isolation valves is contained in a totally enclosed system. The trade name for the fluid is Fyrquel 220. This is a synthetic phosphate ester fluid. The fluid does not easily sustain combustion (due to its self-extinguishing property). A prolonged exposure to an ignition source is required to initiate and maintain combustion. It is incredible to postulate such an ignition source in these rooms. All the safe shutdown equipment in these rooms is qualified to a steam environment and an ambient temperature of 320°F.

Since the access to this area is controlled and limited, the transient combustibles introduced into these rooms will be those associated with the maintenance of equipment located in these rooms. Any major maintenance work on the isolation valves or the power-operated relief valves will require a plant shutdown. (Maintenance on these valves will normally be done during refueling outage). Any postulated transient fire will not damage the 2-foot-thick concrete wall. Due to the low combustible loading and the strict control on access and transient combustibles, this area is judged to require a degree of fire protection equivalent to the reactor building. Therefore, the provisions of Appendix R, Section III.G.2.f are applied to this area.

A.23.7.2 Safe Shutdown Capability

Loops 1 and 4 main steam piping (steam generators A and D) and associated isolation valves, relief valves, power-operated relief valves, and pressure transmitters are located in Room 1508. The corresponding equipment for Loops 2 and 3 (steam generators B and C) are located in Room 1509. The feedwater piping and associated isolation and nonreturn valves for Loops 1 and 4 are installed in Room 1411 while Room 1412 has the equipment for Loops 2 and 3. In addition, Room 1412 has two valves on the steam supply from Loops 2 and 3 main steam piping to the auxiliary feedwater pump turbine. The steam generator blowdown isolation valves are installed in Rooms 1412 and 1411.

The failure modes on loss of power (or air if air operated) for the various valves in this area are given below:

	Valve	Normal Mode	Failure Mode
a.	Main steam isolation	Open	Open
b.	Main steam isolation valve bypass	Closed	Closed

for design data. Safety-related feedwater piping materials are discussed in Section 10.3.6.

MAIN FEEDWATER PIPING - Feedwater is supplied to the four steam generators by four 14-inch carbon steel lines. Each of the lines is anchored at the containment wall and has sufficient flexibility to provide for relative movement of the steam generators due to thermal expansion. The main feedwater line and associated branch lines between the containment penetration and the torsional restraint upstream of the MFIV are designed to meet the "no break zone" criteria, as described in NRC BTP MEB-3-1 (refer to Section 3.6).

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MAIN FEEDWATER ISOLATION VALVES - One main feedwater isolation valve (MFIV) is installed in each of the four main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from more than one steam generator in the event of a feedwater pipe rupture in the turbine building. The main feedwater check valve provides backup isolation. The MFIVs isolate the ~~nonsafety-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. The valves are bi-directional, double disc, parallel slide gate valves. ~~Stored energy for closing is supplied by accumulators which contain a fixed mass of high pressure nitrogen and a variable mass of high pressure hydraulic fluid. For emergency closure, a solenoid is de-energized, which causes the high pressure hydraulic fluid to be admitted to the top of the valve stem driving piston and also causes the fluid stored below the piston to be dumped to the fluid reservoir. Two separate pneumatic/hydraulic power trains are provided for each MFIV. Electrical solenoids are energized from separate Class IE sources.~~

Insert K

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MAIN FEEDWATER CONTROL VALVES AND CONTROL BYPASS VALVES - The MF control valves are air-operated angle valves which automatically control feedwater between 20 percent and full power. The bypass control valves are air-operated globe valves, which are used during startup up to 25-percent power. The MF control valves and bypass control valves are located in the turbine building.

In the event of a secondary cycle pipe rupture inside the containment, the main feedwater control valve (and associated bypass valve) provide a diverse backup to the MFIV to limit the quantity of high energy fluid that enters the containment through the broken loop. For emergency closure, both solenoids, when de-energized, will result in valve closure. Electrical solenoids are energized from separate Class IE sources.

MAIN FEEDWATER CHECK VALVES - The main feedwater check valves are located inside the containment, downstream of the auxiliary feedwater connection. In the event of a secondary cycle pipe rupture, inside the containment, the main feedwater check valves provide a diverse backup to the MFIV to ensure the pressure boundary of any intact loop not receiving auxiliary feedwater.

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The MFIV actuators utilize two separate actuation trains, which are energized from separate Class 1E sources. Energy for closing an MFIV is provided by the process fluid (feedwater), which is admitted to the volume above the actuator piston (upper piston chamber) to close the valve. The MFIV actuators utilize six solenoid valves, three solenoids per actuation train, to perform their safety design functions. Process fluid 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 process fluid 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 actuator lower piston chamber, which are in a de-energized state (vented position). After a thirty-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.

hydrazine injection are controlled by conductivity and by hydrazine residual in the system, which are continuously monitored by the process sampling system.

~~Pressure transmitters are installed in each MFIV actuator accumulator for the continuous monitoring of the nitrogen pressure. Pressure switches are also installed which activate control room alarms upon low actuator accumulator pressure. The alarm indicates that the actuator train in question is not capable of closing the valve in the required time.~~

Instrumentation, including pressure indicators, flow indicators, and temperature indicators, required for monitoring the system is provided in the control room. FSAR CN 01-036

10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM

The steam generator blowdown system (SGBS) helps to maintain the steam generator secondary side water within the chemical specifications prescribed by the NSSS supplier. Heat is recovered from the blowdown and returned to the feedwater system. The blowdown is then treated to remove impurities before being returned to the condenser.

10.4.8.1 Design Bases

10.4.8.1.1 Safety Design Basis

Portions of the SGBS are safety related and are required to function following a DBA and to achieve and maintain the plant in a safe shutdown condition. The following safety design bases have been met:

SAFETY DESIGN BASIS ONE - The safety-related portion of the SGBS is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (GDC-2).

SAFETY DESIGN BASIS TWO - The safety-related portion of the SGBS remains functional after an SSE or performs its intended function following a postulated hazard, such as a fire, internal missile, or pipe break (GDC-3 and 4).

SAFETY DESIGN BASIS THREE - Safety functions can be performed, assuming a single active component failure coincident with the loss of offsite power (GDC-34).

SAFETY DESIGN BASIS FOUR - The active components of the SGBS are capable of being tested during plant operation. Provisions are made to permit inservice inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI.

SAFETY DESIGN BASIS FIVE - The SGBS is designed and fabricated to codes consistent with the quality group classification assigned by Regulatory Guide 1.26 and

10.4.9.2.2 Component Description

Codes and standards applicable to the AFS are listed in Tables 3.2-1 and 10.4-12. The AFS is designed and constructed in accordance with quality groups B and C and seismic Category I requirements.

MOTOR-DRIVEN PUMPS - Two auxiliary feedwater pumps are driven by ac-powered electric motors supplied with power from independent Class 1E switchgear busses. Each horizontal centrifugal pump takes suction from the nonsafety-related condensate storage tank, or alternatively, from the ESWS. Pump design capacity includes ~~continuous~~ minimum flow recirculation, which is controlled by ~~restriction orifices.~~ INSERT A

TURBINE-DRIVEN PUMP - A turbine-driven pump provides system redundancy of auxiliary feedwater supply and diversity of motive pumping power. The pump is a horizontal centrifugal unit. Pump bearings are cooled by the pumped fluid. Pump design capacity includes continuous minimum flow recirculation. AC powered valves required for operability of the turbine driven pump are aligned in accordance with Technical Specifications such that their positions are not required to change upon a loss of all ac power. Air operated valves, controls and instrumentation required for operation of the turbine driven pump are powered by the Class 1E dc system or dc backed vital ac system. Swapover to ESW supply is not postulated during a loss of all ac power as discussed in Section 8.3A. The turbine driven pump is diverse from the two ac motor driven pumps with ac motor operated valves powered from the diesel backed on-site power system.

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Steam supply piping to the turbine driver is taken from two of the four main steam lines between the containment penetrations and the main steam isolation valves. Each of the steam supply lines to the turbine is equipped with a locked-open gate valve, normally closed air-operated globe valve with air-operated globe bypass to keep the line warm, and two nonreturn valves. Air-operated globe valves are equipped with dc-powered solenoid valves. These steam supply lines join to form a header which leads to the turbine through a normally closed, dc motor-operated mechanical trip and throttle valve. The main steam system is described in Section 10.3.

The steam lines contain provisions to prevent the accumulation of condensate. The turbine driver is designed to operate with steam inlet pressures ranging from 92 to 1,290 psia. Exhaust steam from the turbine driver is vented to the atmosphere above the auxiliary building roof. Refer to Safety Evaluation Two for a discussion of the design provisions for the exhaust line.

PIPING AND VALVES - All piping in the AFS is seamless carbon steel. Welded joints are used throughout the system, except for flanged connections at the pumps. INSERT B

The piping from the ESWS to the suction of each of the auxiliary feedwater pumps is equipped with a motor-operated butterfly valve, an isolation valve, and a nonreturn valve. Each line from the condensate storage tank is equipped with a motor-operated gate

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INSERT A

automatic recirculation control check valves.

INSERT B

and the automatic recirculation control check valves.

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valve and a nonreturn valve. Each motor-driven pump discharges through a nonreturn valve and a locked-open isolation valve to feed two steam generators through individual sets of a locked open isolation valve, a normally open, motor-operated control valve, a check valve followed by a flow restriction orifice, and a locked-open globe valve. The turbine-driven pump discharges through a nonreturn valve, a locked-open gate valve to each of the four steam generators through individualsets of a locked-open isolation valve, a normally open air-operated control valve, followed by a nonreturn valve, a flow restriction orifice, and a locked-open globe valve.

The turbine-driven pump discharge control valves are air operated with dc-powered solenoid valves. At each connection to the four main feedwater lines, the auxiliary feedwater lines are equipped with check valves.

The system design precludes the occurrence of water hammer in the main feedwater inlet to the steam generators. For a description of prevention of water hammer, refer to Section 10.4.7.2.1.

10.4.9.2.3 System Operation

NORMAL PLANT OPERATION - The AFS is not required during normal power generation. The pumps are placed in the automatic mode, lined up with the nonsafety-related condensate storage tank, and are available if needed.

EMERGENCY OPERATION - In addition to remote manual-actuation capabilities, the AFS is aligned to be placed into service automatically in the event of an emergency. Anyone of the following conditions will cause automatic startup of both motor-driven pumps:

a. Two out of four low-low level signals in any one steam generator

b. Trip of both main feedwater pumps

c. Safeguards sequence signal (initiated by safety injection signal or loss-of-offsite power)

d. Class IE bus loss of voltage sequence signal (i.e. loss-of-offsite power)

e. AMSAC

The turbine-driven pump is actuated automatically on either of the following signals.

a. Two out of four low-low level signals in any two steam generators

b. Class IE bus loss of voltage sequence signal (i.e. loss-of-offsite power)

c. AMSAC

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INSERT C

automatic recirculation control check

TABLE 10.4-6 CONDENSATE AND FEEDWATER SYSTEM DESIGN DATA

Main Feedwater Piping (Safety-Related Portion)	
Design (VWO) flowrate, lb/hr	15,960,000
Number of lines	4
Nominal size, in.	14
Schedule	80
Design pressure, psig	1,185
Design temperature, F	450
Design code	ASME Section III, Class 2
Seismic design	Category I
Feedwater Isolation Valves	
Number per main feedwater line	1
Closing time, sec	5 15
Body design pressure, psig	1,950 FSAR CN 01-036
Design temperature, F	450
Design code	ASME Section III, Class 2
Seismic design	Category I
Feedwater Control Valves	
Number per main feedwater line	1
Closing time, sec	5
Design code	ASME Section III, Class 3
Seismic design	None

TABLE 10.4-7 FEEDWATER ISOLATION SINGLE FAILURE ANALYSIS

Component	Failure	Comments
Main feedwater control valve (MFCV) (1)	Valve fails to close upon receipt of automatic signal (FIS)	MFIV will close, providing adequate isolation to limit high energy fluid addition
	Loss of power from one power supply	Valve fails as is upon loss of either train of power; however, MFIV can be closed providing adequate isolation to limit high energy fluid addition.
Main feedwater bypass control valve. MFBCU(1)	Same as main feedwater control valve	Same as main feedwater control valve
Main feedwater isolation valve (MFIV)	Valve fails to close upon receipt of automatic signal (FIS)	MF control valve (1) and MF check valve close as required to isolate.
	INSERT L	The MF control valve (1) (and bypass control valve) serve to limit the addition of high energy fluid into the containment following a main feedwater line rupture inside the containment or a main steam line break.
	~~Loss of power from one power supply~~	~~Valve fails closed upon loss of either train of power~~
Main feedwater check valve	Valve fails to close	MFIV will close, providing adequate isolation
Chemical addition isolation valve	Valve fails to close upon receipt of automatic signal (FIS)	Associated check valve will close, providing adequate isolation
	Loss of power for valve operation	Valve fails closed
Chemical addition check valve	Valve fails to close	Chemical addition isolation valve will close, providing adequate isolation
Auxiliary feedwater check valve	Valve fails to open properly	Remaining two intact steam generators will provide adequate auxiliary feedwater
Steam generator narrow range level (Four per steam generator)	No signal generated for protection logic from one transmitter	2-out-of-4 logic reverts to 2-out-of-3 logic, and protection logic is generated by other channel devices
	Loss of one of four logic channels	2-out-of-4 logic reverts to 2-out-of-3 logic, and protection logic is generated by other channel devices

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(1) Valve provides backup isolation capability following pipe rupture of feedwater line inside containment or following a MSLB, but is not part of primary success path for accident mitigation.

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Loss of Power from one power supply	Valve fails closed upon loss of either train of power. Slight venting of feedwater will occur through the opposite train solenoid valves. This slight venting has been evaluated and found acceptable.
Loss of one solenoid valve	Valve can still be closed by the redundant actuation train. Slight venting of feedwater may occur through the failed solenoid valve. This potential venting has been evaluated and found acceptable.

TABLE 15.1-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)
Excessive feedwater flow at full power	One main feedwater control valve fails fully open	0.0
	High-high steam generator water level setpoint reached	57.7
	Feedwater isolation valves close automatically	64.7 **
	Minimum DNBR occurs	67.5
	Low-low steam generator water level setpoint reached	98.9
	Rods begin to drop	100.9
	Turbine trip occurs due to reactor trip	102.9
Reduction in feedwater temperature at full power	Delivery of cooler feedwater begins	0.0
	Overpower ΔT setpoint reached	21.6
	Rods begin to drop	23.6
	Minimum DNBR occurs	23.8
	Turbine trip occurs due to reactor trip	25.6
	High-high steam generator water level setpoint reached	65.7
	Feedwater isolation valves close automatically	72.7 **
Excessive increase in secondary steam flow		
1. Manual reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached*	100
2. Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached*	50
3. Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached*	150
4. Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0

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TABLE 15.1-1 (Sheet 2)

Accident	Event	Time (sec)
	Equilibrium conditions reached*	50
Inadvertent opening of a steam generator relief or safety valve		
	Inadvertent opening of one main steam safety or relief valve	0.0
	Pressurizer empties	154
	SI actuation	192
	Boron reaches core Criticality attained	211 353
Steam system piping failure		
1. Case 1 (offsite power available)	Steamline ruptures	0
	Low steamline pressure setpoint reached	0.8
	Pressurizer empties	13
	SI actuation	27.8
	Boron reaches core	44
	Criticality attained	45
2. Case 2 (concurrent loss of offsite power)	Steamline ruptures	0
	Low steamline pressure setpoint reached	0.8
	Pressurizer empties	15
	SI actuation	39.8
	Boron reaches core	61
	Criticality attained	62

* Approximate time only

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The Main Feed isolation time listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feedwater Isolation Valves.

TABLE 15.2-1 (Sheet 3)

Accident	Event	Time (sec)
	Second peak water level in pressurizer occurs	3,456
	Core decay heat generation is exceeded by auxiliary feedwater heat removal capacity	~3600
Feedwater system pipe break		
1. With offsite power available (licensing basis)	EAM enables harsh environment low-low level trip setpoint	≤ 10.0
	Feedwater control system malfunction occurs due to harsh environment	10.0
	Low-low steam generator level reactor trip setpoint reached in all steam generators	61.2
	Low-low level trip signal is generated	61.2
	Rods begin to drop and feedwater line rupture occurs	63.2
	Steam generator safety valve setpoint reached (first occurrence)	65.0
	Main feedwater isolation valves closed	68.2 **
	Low steam line pressure setpoint reached in ruptured steam generator	112.9
	Main steam line isolation valves closed	119.9
	Pressurizer water relief begins	409
	Cold auxiliary feedwater is delivered to intact steam generators	~460

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TABLE 15.2-1 (Sheet 4)

Accident	Event	Time (sec)
	Steam generator safety valve setpoint reached in intact steam generators (second occurrence)	462.0
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~1,600
2. Without offsite power available (licensing basis)	EAM enables harsh environment low-low level trip setpoint	≤ 10.0
	Feedwater control system malfunction occurs due to harsh environment	10.0
	Low-low steam generator level reactor trip setpoint reached in all steam generators	61.2
	Low-low level trip signal is generated	61.2
	Rods begin to drop; power lost to the reactor coolant pumps; feedwater line rupture occurs	63.2
	Steam generator safety valve setpoint reached (first occurrence)	65.0
	Main feedwater isolation valves closed	68.2 **
	Low steam line pressure setpoint reached in ruptured steam generator	105.2
	Main steam line isolation valves closed	112.2
	Steam generator safety valve setpoint reached in intact steam generators (second occurrence)	176
	Pressurizer water relief begins	447

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TABLE 15.2-1 (Sheet 5)

Accident	Event	Time (sec)
	Cold auxiliary feedwater is delivered to intact steam generators	~460
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~760
3. With offsite power available (no SI case)	EAM enables harsh environment low-low level trip setpoint	≤ 10.0
	Feedwater control system malfunction occurs due to harsh environment	10.0
	Low-low steam generator level reactor trip setpoint reached in all steam generators	61.2
	Low-low level trip signal is generated	61.2
	Rods begin to drop and feedwater line rupture occurs	63.2
	Steam generator safety valve setpoint reached (first occurrence)	65.0
	Main feedwater isolation valves closed	68.2 ** FSAR CN 01-036
	Low steam line pressure setpoint reached in ruptured steam generator	112
	Main steam line isolation valves closed	119
	Pressurizer water relief begins	397
	Cold auxiliary feedwater is delivered to intact steam generators	~460

TABLE 15.2-1 (Sheet 6)

Accident	Event	Time (sec)
	Steam generator safety valve setpoint reached in intact steam generators (second occurrence)	474
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~1700
4. Without offsite power available (no SI case)	EAM enables harsh environment low-low level trip setpoint	≤ 10.0
	Feedwater control system malfunction occurs due to harsh environment	10.0
	Low-low steam generator level reactor trip setpoint reached in all steam generators	61.2
	Low-low level trip signal is generated	61.2
	Rods begin to drop; power lost to the reactor coolant pumps; and feedwater line rupture occurs	63.2
	Steam generator safety valve setpoint reached (first occurrence)	65.0
	Main feedwater isolation valves closed	68.2 **
	Low steam line pressure setpoint reached in ruptured steam generator	105
	Main steam line isolation valves closed	112
	Steam generator safety valve setpoint reached in intact steam generators (second occurrence)	169
	Pressurizer water relief begins	426

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TABLE 15.2-1 (Sheet 7)

Accident	Event	Time (sec)
	Cold auxiliary feedwater is delivered to intact steam generators	~460
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~850

*DNBR does not decrease below its initial value.

** INSERT M

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INSERT M

The Main Feed isolation time listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feedwater Isolation Valves.

be expected due to the adsorption of some of the iodine is very small compared with the design capacity of these filters.

15.6.2.1.3.2 Dose to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The radiological consequences resulting from the occurrence of a postulated letdown line rupture have been conservatively analyzed, using assumptions and models described in previous sections.

The thyroid inhalation total-body immersion doses have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident at the low-population ~~zone outer~~ boundary. The ~~results are listed~~ in Table 15.6-3. The resultant doses are well within the guideline values of 10 CFR 100.

15.6.3 STEAM GENERATOR TUBE FAILURE

INSERT A

~~15.6.3.1~~ 15.6.3.1.1 Identification of Causes and Accident Description

The letters listed under Reference 3 discuss the reanalysis of the SGTR accident. The licensing basis SGTR accident discussed in this section represents an update to the original SNUPPS generic analysis that assumes the failure of a steam generator atmospheric steam dump valve in the open position in order to maximize offsite doses. This update reflects current plant design, operation, and analysis parameters (e.g., VANTAGE 5/VANTAGE+ fuel, uprated power, 15% equivalent steam generator tube plugging).

~~The accident examined is the complete severance of a single steam generator tube. This event is considered an ANS Condition IV event, a limiting fault (see Section 15.0.1). The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in the contamination of the secondary system due to the leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power or failure of the steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power-operated atmospheric steam dump valves.~~

~~In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during plant operation.~~

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INSERT A

The accident examined is the complete severance of a single steam generator tube. This event is considered an ANS Condition IV event, a limiting fault (see Section 15.0.1). The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in the contamination of the secondary system due to the leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power or failure of the steam dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power-operated atmospheric steam dump valves.

In view of the fact that the steam generator tube material is Inconel-600 and is a highly ductile material, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance, and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during plant operation.

In order to select the reference worst case, a spectrum of steam generator tube rupture (SGTR) events was analyzed. The letters of Reference 3 provide a detailed description of the selection process. Based on the selection process, two major SGTR accident scenarios are identified as the major concerns for radioactive releases to the environment.

The events of major concern, associated with a SGTR, are: (1) failure of an ASD in the open position leading to continued release of steam generator fluid and contained radioactivity and (2) the potential for overfill of the ruptured steam generator with water entering the main steam line resulting in water relief through an atmospheric steam dump (ASD) valve and/or a main steam safety valve (MSSV). To examine these concerns, the SGTR Scoping Code (discussed in Appendix B of the SNUPPS report attached to SLNRC 86-01, see Reference 3), in conjunction with other analyses, was used to evaluate the sensitivity of offsite power; location of tube failures; availability of offsite power; location of the rupture; operator action times; power level; and iodine spiking.

For example, the operator is expected to determine that a SGTR has occurred and to identify and isolate the affected steam generator on a restricted time scale to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the affected generator. Sufficient indications, controls, alarms and procedures are provided to enable the operator to carry out these functions satisfactorily. Operator actions in response to an SGTR are assumed to follow plant-specific emergency procedures, which are based on procedure E-3 (SGTR response) and related procedures of the generic emergency response guidelines (ERGs) for Westinghouse plants.

INSERT A continued

As discussed above, two SGTR accident scenarios are identified as resulting in limiting radioactive releases to the environment. For the case of SGTR with postulated stuck-open Atmospheric Steam Dump (ASD) valve on the ruptured steam generator, the radioactive releases are maximized by assuming the ruptured steam generator ASD is stuck-open for 20 minutes (Reference 3). For the case of SGTR with postulated failure of the ruptured steam generator Auxiliary Feedwater (AFW) flow control valve, auxiliary feedwater flow is maximized in order to increase the probability for ruptured steam generator overfill and to maximize subsequent liquid relief from its safety valve. The radioactive releases are maximized by assuming that the safety valve is stuck-open following liquid relief with an effective flow area equal to 5% of the total safety valve flow area (Reference 5). Detailed analyses are presented for these two scenarios in Sections 15.6.3.1 and 15.6.3.2 respectively.

15.6.3.1 STEAM GENERATOR TUBE RUPTURE WITH POSTULATED STUCK-OPEN ATMOSPHERIC STEAM DUMP VALVE

~~In order to select the reference worst case, a spectrum of SGTR events was analyzed. The letters of Reference 3 provide a detailed description of the selection process.~~

~~Major concerns associated with a steam generator tube rupture (SGTR) are: (1) the potential for overfill of the faulted steam generator with water entering the main steam line resulting in water relief through an atmospheric steam dump (ASD) valve and (2) failure of an ASD in the open position leading to continued release of steam generator fluid and contained radioactivity. To examine these concerns, the SGTR Scoping Code (discussed in Appendix B of the SNUPPS report attached to SLNRC 86-01 - see Reference 3), in conjunction with other analyses, was used to evaluate the sensitivity of SGTR events to a number of parameters. Parameters investigated were: single active failures; availability of offsite power; location of tube rupture; operator action times; power level; and iodine spiking. The analysis presented in this section was chosen, based upon those investigations, as providing the worst case dose.~~

INSERT B

The recovery sequence for a SGTR is discussed in Section ~~15.6.3.2.~~ 15.6.3.1.2.

INSERT C

The operator is expected to determine that a SGTR has occurred and to identify and isolate the affected steam generator on a restricted time scale to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the affected generator. ~~The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam line. Sufficient indications, controls, alarms, and procedures are provided to enable the operator to carry out these functions satisfactorily.~~

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Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed such that pressure equalization between the primary and secondary can eventually be achieved and break flow terminated within 67.3 minutes of accident initiation.

~~Operator actions in response to an SGTR are assumed to follow plant-specific emergency procedures, which are based on procedure E-3 (SGTR response) and related procedures of the generic emergency response guidelines (ERGs) for Westinghouse plants.~~

The timing of operator actions utilized in the tube rupture analysis presented in this section has been estimated using data from the following sources: (1) plant simulator exercises; (2) SGTR events at the Ginna, Prairie Island, and North Anna plants; (3) draft standard ANS 58.8, Revision 2; and (4) Callaway operating experience in closing an atmospheric steam dump manual block valve. Heaviest weight has been placed on the simulator and experience data because it reflects what plant operators have done using plant-specific procedures. The letters of Reference 3 provide more detail on the timing of operator actions.

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INSERT B

As discussed in Reference 3, the analysis presented in this section was originally selected as the "worst case dose" scenario for the SGTR accident.

INSERT C

with a stuck-open ASD

g. The atmospheric steam dump (ASD) valve for the ruptured steam generator (SG) is assumed to fail open and steam release continues for 20 minutes until the ASD block valve is manually closed (Figure 15.6-3g). During this time, pressure falls in all SGs and RCS temperature drops in response to the steam release.

h. Once the ASD is isolated, controlled cooldown is initiated and continues until RCS temperature is reduced to 50°F less than the ruptured SG saturation temperature.

i. Primary depressurization is then performed until primary and secondary pressures equalize. This terminates break flow (Figure 15.6-3i). Safety injection flow is terminated 3 minutes after RCS depressurization. However, in the interim, safety injection flow repressurizes the primary side.

j. Five minutes after safety injection termination, it is assumed that the operators minimize the primary-secondary pressure difference by opening a pressurizer PORV (Figures 15.6-3a and 15.6-3m). Any primary side pressure rise after this second depressurization is moderate and a function of decay heat.

15.6.3.1.2

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15.6.3.2 Analysis of Effects and Consequences

Method of Analysis

Mass and energy balance calculations are performed using RETRAN (Section 15.0.11.8) to determine primary-to-secondary mass release and to determine the amount of steam vented from each of the steam generators from the occurrence of the tube rupture until after the second primary-secondary pressure equalization. RETRAN provides time-dependent values of RCS mass, break flow, flashed fraction, steam generator liquid mass, and steam generator atmospheric steam dump valve flow for the calculation of radiological consequences. Conservatively high values of break flow rate and flashed fraction are assumed for the first hour of the transient to maximize radiological consequences. Supplementary mass and energy balance calculations, with conservative assumptions, are performed for the period from pressure equalization until 8 hours after the accident, beyond the time of RHR initiation.

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

a. Reactor trip and safety injection occur coincidentally as a result of low pressurizer pressure. Overtemperature ΔT trip is not considered. This allows more break flow. Loss of offsite power occurs at reactor trip.

i. The ruptured steam generators's ASD ~~is set at 1184.7 psia.~~ This is 4% higher than the nominal setpoint which delays the release of pressure from the ruptured steam generator, resulting in increased valve discharge flow and integrated break flow. The ASD on the ruptured steam generator fails open for 20 minutes, beginning on initial demand, shortly after reactor trip. ~~This is the single failure that maximizes offsite doses.~~ FSAR CN 03-023

j. The initial steam generator pressure is 939 psia, the minimum expected pressure associated with 15% tube plugging. This increases the leaked reactor coolant.

k. The decay heat multiplier is 1.2 (based on 3636 MWt). ~~This maximizes the heat to be transferred which increases break flow.~~

l. The narrow range level in all steam generators must be greater than 10% and the ruptured steam generator pressure must be greater than 615 psig prior to initiating RCS cooldown.

m. RCS depressurization is assumed to begin 3 minutes after completion of cooldown. When the ruptured steam generator pressure is higher than the RCS pressure, the pressurizer PORVs are closed.

n. Safety injection is terminated 3 minutes after completion of RCS depressurization.

Other initial conditions, given in Table 15.0-2, are chosen to maximize RCS temperatures, decay heat, flashed fraction of RCS leakage, and break flow, thereby maximizing radioactivity transfer to the secondary and, consequently, offsite doses.

The above assumptions, suitably conservative for this case ~~the licensing basis tube rupture~~, are made to maximize offsite doses.

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Prior to reactor trip, steam is dumped to the condenser from both the ruptured and intact steam generators. After the condenser is lost, following assumed loss of offsite power at reactor trip, steam from all steam generators is released to the atmosphere.

Following isolation of the ruptured steam generator, it is assumed that atmospheric steam dump from the intact steam generators is ~~used to reduce the RCS temperature~~ to 50°F below the ruptured steam generator saturation temperature. From 2 to 5 hours, steam is assumed to be relieved from the intact steam generators to reduce the RCS temperature and pressure to RHRS conditions. The ruptured steam generator is depressurized to the RHRS cut-in pressure using the emergency recovery procedures. After 5 hours, further plant cooldown is carried out with the RHRS. The 0 to 2 hour and 2 to 8 hour steam releases from the intact steam generators required to remove decay heat, metal heat, reactor coolant pump heat, and stored fluid

Number	Title
15.6-3d	Steam Flow Rate (Intact Generators) Transient for Steam Generator Tube Rupture Event
15.6-3e	Steam Flow Rate (Faulted Generator) Transient for Steam Generator Tube Rupture Event
15.6-3f	Steam Generator Temperature (Faulted and Intact Generators) Transients for Steam Generator Tube Rupture Event
15.6-3g	Steam Generator Atmospheric Relief Valve Flow Rate (Faulted Generator) Transient for Steam Generator Tube Rupture Event
15.6-3h	Steam Generator Atmospheric Relief Valve Flow Rate (Intact Generators) Transient for Steam Generator Tube Rupture Event
15.6-3i	Faulted Steam Generator Break Flow Rate Transient for Steam Generator Tube Rupture Event
15.6-3j	Auxiliary Feedwater Flow Rate and Narrow Range Level (Faulted Generator) Transients for Steam Generator Tube Rupture Event
15.6-3k	Auxiliary Feedwater Flow Rate and Narrow Range Level (Intact Generators) Transients for Steam Generator Tube Rupture Event
15.6-3l	Steam Generator Liquid Volume (Faulted Generator) Transient for Steam Generator Tube Rupture Event
15.6-3m	Pressurizer PORV Flow Rate Transient for Steam Generator Tube Rupture Event
15.6-3n	Pressurizer Liquid Volume Transient for Steam Generator Tube Rupture Event
15.6-3o	Feedwater Flow Rate (Faulted Generator) Transient for Steam Generator Tube Rupture Event
15.6-3p	Feedwater Flow Rate (Intact Generators) Transient for Steam Generator Tube Rupture Event

15.6.3.1.3

~~15.6.3.3~~ Radiological Consequences

~~15.6.3.3.1~~ Method of Analysis

~~15.6.3.3.1.1 Physical Model~~

The evaluation of the radiological consequences due to a postulated steam generator tube rupture (SGTR) with a stuck open atmospheric steam dump valve on the ruptured steam generator assumes a complete severance of a single steam generator 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.

Steam generator blowdown will automatically be terminated by the SGBSIS (AFAS) signal (refer to Section 10.4.8) which is initiated by the safety injection signal. The assumed coincident loss of offsite power will cause closure of the condenser steam dump valves to protect the condenser. The steam generator pressure will then increase rapidly, resulting in steam discharge as well as activity release through the steam generator atmospheric steam dump valves. Venting from the affected steam generator, i.e., the steam generator which experiences the tube rupture, will continue until the manual block valve is closed, isolating the stuck open atmospheric steam dump valve on the ruptured steam generator. At this time, the affected steam generator is effectively isolated. The remaining unaffected steam generators remove core decay heat by venting steam through the atmospheric steam dump valves until the controlled cooldown is terminated.

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 break flow and coolant leakage rates, the percentage of defective fuel in the core, flashed fraction of reactor coolant, and the mass of steam discharged to the environment. Conservative assumptions were made for all these parameters.

15.6.3.3.1.2 <u>Assumptions and Conditions</u>

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.6-4 and 15A-1 and are summarized below.

The assumptions used to determine the concentrations of isotopes in the reactor coolant and secondary systems prior to the accident are as follows:

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a. The assumed reactor coolant iodine activity is determined for the following two cases:

Case 1 - The initial reactor coolant iodine activity corresponds to an isotope mixture that bounds Technical Specification allowable conditions for both tight and open fuel defects. The initial isotopic mix is based on the relative concentrations from Table 11.1-5. The concentrations are then changed to achieve a Does Equivalent I-131 (DEI) of 1.0 μCi/gm, while maintaining the isotopic ratios from Table 11.1-5. This provides conservative values for the longer lived iodines which contribute the majority of the calculated thyroid dose. The initial inventories of the shorter lived iodine isotopes, which can provide a significant contribution to the calculated whole body does, are then increased to conservatively bound isotopic mixes which may occur in the presence of open fuel defects. Case 1 then includes an accident initiated, spiked release rate that increases by a factor of 500 during the accident sequence.

Case 2 - The initial reator coolant iodine activity corresponds to an assumed pre-accident iodine spike which results in concentrations that are a factor of 60 higher than those used in Case 1.

b. The noble gas activity in the reactor coolant is based on 1-percent failed fuel, as provided in Table 11.1-5.

c. The initial secondary side radio-iodine concentrations are assumed to be 10% of the initial Case 1 primary side concentrations.

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

a. Break flow to the ruptured steam generator is conservatively assigned values that bound calculated break flow rate values. The assumed values bound the break flow rates calculated by the RETRAN code. Break flow rate values are discussed in Table 15.6-4.

b. The fraction of reactor coolant that flashes to steam after reaching the secondary side, as assumed in the accident analysis, varies over time. Key events which trigger changes in the assumed flashed fraction are reactor trip and closure of the manual block valve to isolate the failed open SG atmospheric steam dump valve. Flashed fraction values assumed in the radiological analysis are described in Table 15.6-4.

a. A 1-gpm primary-to-secondary leak is assumed to occur to the unaffected steam generators.

b. All noble gas activity in the reactor coolant which is transported to the secondary system via the tube rupture and the primary-to-secondary leakage is assumed to be immediately released to the environment.

c. At 67.3 minutes after the accident, it is assumed that the RCS and steam generator pressures are equalized and below the steam generator atmospheric relief valve set pressure. Break flow to the faulted steam generator and primary-to-secondary leakage to the intact steam generators are conservatively assumed to continue until 8 hours after the tube rupture.

d. The iodine partition faction between the liquid and steam in the steam generator is assumed to be 0.01.

e. The steam releases from the steam generators to the atmosphere are given in Table 15.6-4.

f. Offsite power is lost.

g. Five hours after the accident, the RHR system is assumed to be in operation to cool down the plant. Thus, no additional steam release is assumed.

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

i. Short-term accident atmospheric dispersion factor, breathing rates, and dose conversion factors are provided in Tables 15A-2, 15A-1, and 15A-4, respectively.

15.6.3.3.1.3 Mathematical Models Used in the Analysis

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Mathematical models used in the analysis are described in the following sections:

a. The mathematical models used to analyze the activity released during the course of the accident are based on the assumptions listed above.

b. The atmospheric dispersion factors used in the analysis were calculated based on the onsite meteorological measurements program, as described in Section 2.3 of the Site Addendum, and are provided in Table 15A-2.

c. The thyroid inhalation immersion doses to a receptor at the exclusion area boundary and outer boundary of the low-population zone were analyzed, using the models described in Appendix 15A.

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15.6.3.3.14 Identification of Leakage Pathways and Resultant Leakage Activity

For the purposes of evaluating the radiological consequences due to a postulated SGTR, the activity released from the affected steam generator, prior to isolation, is released directly to the environment by the atmospheric steam dump valve. The unaffected steam generators are assumed to continually discharge steam and entrained activity via the atmospheric steam dump valves up to the time initiation of the RHR system can be accomplished. Since the activity is released directly to the environment with no credit for plateout or retention, the results of the analysis are based on the most direct leakage pathway available. Therefore, the resultant radiological consequences represent the most conservative estimate of the potential integrated dose due to the postulated SGTR.

15.6.3.3.2 Identification of Uncertainties and Conservatisms in the Analysis

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a. Reactor coolant activities based on extreme iodine spiking effects are orders of magnitude greater than that assumed for normal operating conditions.

b. A 1-gpm steam generator primary-to-secondary leakage, with a conservatively high density, is assumed which is significantly greater than that anticipated during normal operation. This leakage continues for 8 hours, even though RHR operation is assumed to begin at 5 hours.

c. Tube rupture of the steam generator is assumed to be a double-ended severance of a single steam generator tube. This is a conservative assumption, since the steam generator tubes are constructed of highly ductile materials. The more probable mode of tube failure is one or more minor leaks of undetermined origin. Activity in the secondary steam system is subject to continual surveillance, and the accumulation of activity from minor leaks that exceeds the limits established in the technical specifications would lead to reactor shutdown. Therefore, it is highly unlikely that the total amount of activity considered available for release in this analysis would ever be realized.

d. The coincident loss of offsite power with the occurrence of an SGTR is a highly conservative assumption. In the event of the availability of offsite power, the condenser dump valves will open, permitting steam dump to the condenser. This will reduce the amount of steam and entrained activity discharged directly to the environment from the unaffected steam generators.

e. The meteorological conditions which may be present at the site during the course of the accident are uncertain. However, it is highly unlikely that the meteorological conditions assumed will be present during the course of the the accident for any extended period of time. Therefore, the radiological consequences evaluated, based on the meteorological conditions assumed, are conservative.

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f. The radiological consequences have been based on a worst case single failure in the open position of the ruptured steam generator atmospheric steam dump (ASD) valve which isn't isolated until 30.4 minutes after tube rupture.

g. The flashed fraction of the break flow to the ruptured steam generator varies over time. The values assumed in the radiological consequence calculation conservatively bound the calculated values. Key events which trigger changes in the assumed flashed fraction include reactor trip and closure of the manual block valve to isolate the failed open SG atmospheric steam dump valve. Specific values for the flashed fraction are listed in Table 15.6-4.

h. The flashed fraction of the primary-to-secondary leakage to the intact steam generators is conservatively assumed to be the same as in the ruptured steam generator.

i. Break flow to the ruptured steam generator is conservatively assigned values that bound calculated break flow rate values. Break flow rate values are discussed in Table 15.6-4.

j. There are two steam release pathways from the ruptured steam generator that are addressed by the radiological consequence calculation. These pathways are the steaming and flash pathways. The steam pathway accounts for the boiling of the secondary side water inventory of the ruptured steam generator. The flash pathway accounts for the fraction of the leaked primary fluids which immediately flashes to steam after arriving in the secondary side. Release via the steaming pathway is terminated by the SG atmospheric steam dump block valve closure at 30.4 minutes. Release via the flash pathway is conservatively continued following block valve closure. Release via this pathway is continued until the RETRAN results indicate that no further flashing will occur.

k. The steam release from the intact steam generator ASDs during the 5 hour ~~cooldown to RHR cut-in conditions is conservatively assumed to occur in its entirety during the 0-2 hour period of the transient.~~

l. Whole body doses from the intact steam generator ASDs during the cooldown to RHR cut-in conditions are calculated using conservative primary side activities.

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INSERT D

Table 15.6-5 lists the offsite doses for the SGTR with a stuck-open ASD.

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15.6.3.2 STEAM GENERATOR TUBE RUPTURE WITH FAILURE OF FAULTED STEAM GENERATOR AFW CONTROL VALVE

add underline

15.6.3.2.1 Identification of Causes and Accident Description

As discussed in Reference 3, an SGTR case demonstrating the effects of steam generator overfill was performed. In this case the analysis assumes the failure of the AFW control valve on the discharge side of the motor-driven AFW pump feeding the ruptured steam generator. The ARV on the ruptured steam generator is not assumed to fail open. ~~It is assumed that this valve fails in the wide-open position to maximize the flow to the ruptured steam generator.~~ Failure of this valve coupled with the contribution from the turbine-driven AFW pump provides a greater potential for overfilling the ruptured steam generator. For this special overfill scenario, reactor trip and safety injection actuation were conservatively assumed at SGTR initiation (time zero) to maximize the AFW addition to the ruptured steam generator. Some of the assumptions which differ from the analysis described in Section 15.6.3.1.1 do so because the trip time sensitivity has been eliminated. The effect of these revised assumptions is an increase in break flow and ruptured steam generator AFW flow, which results in overfill and water relief.

ASD

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The analysis scenario is outlined below. This analysis is consistent with the overfill scenario presented in Reference 3, but has been updated to match the current plant configuration. This includes revised (longer) operator action times that reflect recent simulator studies of this SGTR scenario.

ASDs

MSSVs.

setpoints of the

An SGTR occurs while the plant is at 100% thermal power and while at steady state. Concurrent with the SGTR a reactor trip occurs and a safety injection signal is generated. A loss of offsite power (LOOP) is assumed coincident with the reactor trip. Following reactor trip, safety injection actuation, and the loss of offsite power, the feedwater flow stops and the Main Steam Isolation Valves close. The secondary pressure rises and approaches the secondary ARVs and SVs setpoints. In response to the reactor trip and LOOP, auxiliary feedwater is delivered to the secondary. It is assumed that the AFW control valve fails full open on the ruptured SG and delivers excessive AFW to the ruptured steam generator. The excessive AFW flow quickly rebounds the ruptured steam generator water level and drives the steam generator toward overfill.

ASD

In accordance with the emergency operating procedures (EOPs), the ruptured SG is isolated by ensuring that the MSIV, ARV, and blowdown isolation valves are closed on the ruptured loop. The final isolation step requires AFW termination to the SG. After isolation, the primary and ruptured secondary pressure rise in response to reduced heat removal. Following isolation of the ruptured steam generator, operators begin cooldown of the primary via the intact steam generators' ARVs. Eventually proper subcooling limits are obtained and primary depressurization is initiated using a primary power operated relief valve (PORV). Primary depressurization is performed until primary and secondary pressures equalize. This stops break flow momentarily. In accordance with EOP procedures, the safety injection flow is terminated fairly soon after the depressurization step. Unfortunately, safety injection flow, in the interim, has

ASDs.

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The ASD never opens and all liquid relief is considered through a main steam safety valve (MSSV). The AFW control valve is assumed failed in the wide-open position to maximize the flow to the ruptured steam generator.

re-pressurized the primary and a primary/secondary pressure difference still exists. After SI termination, it is assumed that the operators minimize the primary/secondary pressure difference by opening a PORV. ~~Any primary~~ rise after this step is moderate and a function of decay heat.

Primary and secondary equilibrium does not occur before the ruptured steam generator overfills and water fills the steamline up to the MSIV. When the steam generator and steamline go water solid a pressure spike (on the secondary) occurs as the primary side (driven by SI) drives the secondary pressure toward equilibrium. Thus a safety valve opens and contaminated water is dumped to the atmosphere. Water continues to be relieved from the ruptured SG SV until equilibrium is reached between ~~the primary~~ and secondary pressures, effectively terminating flow into the ruptured steam generator. To assure continued relief, an active failure of the SV is assumed to occur, i.e., after water relief the valve remains partially open (5%). Eventually, water relief depletes the secondary mass and creates a steam void. This steam void grows until water is no longer able to pass out the safety valve.

SG MSSV

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It is assumed that steam relief continues until RHR cut-in, since steam relief continues to shrink the ruptured SG mass via cooling and mass depletion. Following break flow termination it is assumed that the operators transition to the cooldown procedures and initiate cooldown via intact SG atmospheric steam dump. Cooldown to RHR cut-in conditions requires approximately 4 hours from initiation of intact SG atmospheric steam dump.

15.6.3.2.2 Analysis of Effects and Consequences

add underlines

Method of Analysis

Mass ~~and energy balance~~ calculations are performed using RETRAN to determine the plant response to the SGTR and calculate the break flow, break flow flashing, secondary releases, and system masses for the calculation of the radiological consequences. The Westinghouse RETRAN version based on RETRAN02 MOD005 (Reference 30) was used for this analysis in place of RETRAN02 MOD003 (Section 15.0.11.8) which was used for the original analyses documented in Reference 3. (Modeling of the plant using RETRAN volumes, junctions, control blocks, etc. is unchanged from Reference 3. The change to the Westinghouse RETRAN version was verified to have an insignificant impact on the predicted plant response for this scenario.)

In the calculation of the plant response for this scenario the following assumptions are made:

a. Single failure: The ruptured steam generator's auxiliary feedwater control valve fails in the full open position.

b. Additional active failure: The ruptured steam generator's safety valve fails partially open (5% effective area) after water relief.

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Page 3

steam dump (ASD) valve

INSERT E2

c. The atmospheric ~~relief valve (ARV)~~ on the ruptured SG is assumed inoperable.

d. The tube rupture is modeled as a double-ended-guillotine break of a single tube at the cold leg tube sheet. An additional 5% uncertainty is added to the flow predicted for resistance limited flow.

e. Initial conditions 3565

- Core power = ~~3565~~ MWt
- Pressurizer pressure = 2280 psia. This is the nominal pressure plus error allowance. The higher pressure maximizes the break flow.
- Pressurizer level = 65% (NRS) of narrow range span
- Vessel average temperature = 583.4°F - 5°F = 578.4°F. This is the minimum expected vessel average temperature. The lower temperature increases the density of the reactor coolant and thus increases the leakage.
- RCS flow = minimum measured flow = ~~382640~~ gpm 382,640
- Steam generator pressure = 908 psia. This is the lowest credible secondary pressure and produces the highest initial break flow. The lower initial pressure also reduces the steam releases after trip, maintaining inventory in the ruptured steam generator and bringing it closer to overfill. INSERT E5
- Feedwater temperature = 390°F.
- Steam generator level = 55% NRS. This is the nominal level plus uncertainty to maximize the initial inventory.
- Steam generator tube plugging: The initial conditions are based on 10% steam generator tube plugging. However, the SG heat transfer model in RETRAN is based on 15% tube plugging and this is conservatively retained. A conservatively high initial secondary mass is assumed to bound 0% tube plugging.

f. Reactor trip occurs at time zero.

g. Loss of offsite power (LOOP) occurs at reactor trip (i.e., at time zero)

h. MSIV isolation is modeled at reactor trip and the assumed loss of offsite power, although it could be significantly delayed based on the expected operator response. Early isolation of the MSIV allows the ruptured SG to depressurize due to the addition of the (maximum) AFW flow, while the intact SG pressure stays relatively high. This results in increased break flow to the ruptured SG, which is conservative. It also leads to higher AFW flow to the ruptured SG. If the MSIV would be left open, the SGs would tend to be at the same pressure, which would be closer to that of the intact SGs (which are lumped together in the RETRAN model). Also, with the MSIV open, overfilling the ruptured SG would not necessarily lead to water relief, since ~~the~~ water could go to the intact

ruptured and intact

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in the closed position for the duration of the accident sequence.

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More limiting results are obtained by replacing the nominal value with this minimal value.

SGs. The secondary pressure would not spike and the safety valve would not lift.

i. The MSIV closes in 1.5 seconds. As noted above, early isolation is considered to be more limiting.

j. The main feedwater isolation valve (MFIV) closure is modeled as a step function after a 17 second delay. The SI signal generated at reactor trip initiates the isolation.

k. Decay heat = 0.8 x 1979 ANS 2σ model

l. The following maximum AFW flow rates are modeled prior to partial/full isolation of AFW flow to the ruptured SG:

- The AFW flow to the ruptured SG before isolation of the turbine driven AFW pump flow to the ruptured steam generator, at the intact SG pressure of 1235.7 psia is used as a base. As the intact SG pressure drops the flow to the ruptured SG is reduced. This model is reflected in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)		Intact SG Pressure (psia)	Reduction in AFW to Ruptured SG (gpm)
414.7	1317.0		414.7	72.6
614.7	1214.0		614.7	55.4
814.7	1104.0		814.7	37.8
1014.7	982.0		1014.7	20.0
1139.7	895.0		1139.7	8.6
1235.7	823.0		1235.7	0.0

- The AFW flow to the intact SGs (total for the 3) before isolation of the turbine driven AFW pump flow to the ruptured steam generator is provided in the table below.

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1691.0
414.7	1576.0
614.7	1455.0
814.7	1326.0
1014.7	1186.0
1139.7	1091.0
1235.7	1013.0

m. The following maximum AFW flow rates are modeled after partial/full isolation of AFW flow to the ruptured SG:

- The AFW flow to the ruptured SG after isolation of the turbine driven AFW pump flow to the ruptured steam generator is provided in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)
414.7	770.
614.7	712.
814.7	651.
1014.7	586.
1139.7	537.
1235.7	498.

- The AFW flow to ~~the intact SGs (total for the 3)~~ after isolation of the turbine driven AFW pump flow to the ruptured steam generator, and after complete isolation of AFW to the ruptured SG, is provided in the table below:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1760.
414.7	1656.
614.7	1546.
814.7	1425.
1014.7	1295.
1139.7	1205.
1235.7	1129.

n. AFW flow is initiated 5 seconds after reactor trip, with a 30-second ramp up to full flow. INSERT E3

o. Safety Injection modeling: High and intermediate injection pumps assumed with maximum expected flow. Injection starts 15 seconds after the SI signal (which is generated at the start of the event).

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p. All three intact SG ~~ARVs are~~ credited in the cooldown.

q. Operator actions modeled:

- Isolation of turbine-driven AFW flow to the ruptured SG at 10 minutes from the start of the event.

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Quicker initiation of AFW flow provides more limiting results for this accident sequence.

- Isolation of all AFW flow to the ruptured SG at 20 minutes from the start of the event.
- Initiate cooldown by dumping steam from the lumped intact loop SG ~~ARV~~ ASD after 30 minutes from reactor trip (which is at the start of the event).
- The cooldown is terminated when the core outlet temperature reaches the target temperature specified in the EOPs as a function of the ruptured SG pressure.
- Initiate RCS depressurization ~~by dumping steam~~ using the pressurizer power - operated relief valve at a time such that the depressurization will be completed at 40 minutes from the start of the event. FSAR CN 03-023
- The depressurization is terminated when the pressurizer pressure and the faulted SG pressure are equal.
- SI flow is terminated 5 minutes after the depressurization is completed. This is approximately 45 minutes from the start of the event.
- Depressurize ~~by dumping steam~~ using pressurizer power - operated relief valve following SI termination to terminate break flow at 60 minutes from the start of the event.
- Cooldown to RHR cut-in is initiated after break flow is terminated. The RETRAN analysis does not include the complete cooldown to RHR conditions. The initial part of the cooldown is shown to demonstrate that once the cooldown is initiated the pressure differential (and break flow) is minimal.

r. The break flow flashing fraction is conservatively determined assuming all break flow is at the ruptured loop hot leg temperature.

s. Pressurizer heaters and sprays are not modeled~~.~~ INSERT E4

t. Ruptured steam generator secondary side volume modeling: The secondary side volume of a single SG is ~5825 ft^3. The steam line volume up to the MSIV is ~680 ft^3. The estimated volume in the horizontal section of the steam pipe up to the MSIV is 201 ft^3. Only 100 ft^3 of this volume (about half) is credited. Water relief is not started until the pressure spike, which occurs when the defined RETRAN volume becomes water solid, lifts the SV. Until that time water is filling the steamline up to the MSIV but does not force the SV open so no water is released. Once water release starts it continues until the water in the ruptured SG steamline drops below the 100 ft^3 of horizontal steamline.

Results add underline

The sequence of events is presented in Table 15.6-1. These events include the postulated operator action times.

The single tube rupture leads to a slow depressurization of the reactor coolant system. Reactor trip and safety injection (SI) actuation are assumed to occur coincident with the

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(heaters are not energized at this point in the accident sequence).

tube rupture and loss of offsite power is assumed to result from the trip. The SI signal initiates makeup flow and isolates main feedwater. AFW is initiated to provide cooling for decay heat removal. The secondary pressure rises, but no steam is released from the ruptured SG since the ARV is not credited and the pressure does not reach the SV setpoint. ASD MSSV

Within 10 minutes the operators are assumed to isolate flow from the turbine driven AFW pump to the ruptured steam generator in response to the increase in level in that steam generator. Due to the assumed failure of the valve from the motor driven AFW pump, AFW flow to the ruptured steam generator continues until 20 minutes when the operators are assumed to have isolated all AFW flow to the ruptured SG.

Due to the assumed high initial secondary side water inventory, maximum AFW and conservatively high break flow modeling, the ruptured steam generator overfills before it is completely isolated. When the steamline volume up to the MSIV fills with water, a pressure spike occurs in that steam line, and the safety valve lifts. Initial flow out ~~of the~~ safety valve is high, matching the flow into the steam generator (AFW plus break flow). MSSV After AFW isolation and as break flow is reduced, the flow out the SV drops. It is assumed that the valve sticks open at 5% effective area, leading to continued water relief at rates that exceed the flow into the steam generator. Eventually, water relief depletes the secondary mass and creates a steam void. This steam void grows until water is no longer able to pass out the safety valve. ASD

At 30 minutes the operators initiate RCS cooldown by opening the intact SG ARVs. This cooldown continues until the subcooling margin appropriate to allow the primary depressurization is reached. The cooldown is completed approximately 36.5 minutes into the event.

At approximately 38 minutes operators depressurize the primary using a pressurizer power operated relief valve (PORV) until primary-secondary pressure equilibrium is reached, at approximately 40 minutes. Safety injection flow is terminated 5 minutes later. A second RCS depressurization is initiated at approximately 60 minutes from the start of the event, leading to break flow termination. Cooldown to RHR conditions using ASD the intact SG ARVs is assumed to be initiated 60 ~~minutes from the start of the~~ event.
at approximately

Eventually, the steam void resulting from continued water relief from the assumed stuck MSSV open SV on the ruptured steam generator grows to the extent that the valve no longer passes water. This occurs at approximately 76 minutes from the start of the event.

The following is a list of figures of pertinent time dependent parameters:

Number	Title
15.6-3.2.a	Pressurizer and Steam Generator (Ruptured and Intact Generators) Pressure Transients for SGTR Event with Overfill
15.6-3.2.b	Reactor Coolant System Temperature (Ruptured Loop) Transient for SGTR Event with Overfill

Number	Title
15.6-3.2.c	Reactor Coolant System Temperature (Intact Loops) Transient for SGTR Event with Overfill
15.6-3.2.d	Reactor Coolant System and Steam Generator (Ruptured and Intact Generators) Water Mass Transient for SGTR Event with Overfill
15.6-3.2.e	Ruptured Steam Generator Break Flow Flashing Fraction Transient for SGTR Event with Overfill
15.6-3.2.f	Steam Generator Temperature (Ruptured and Intact Generators) Transient for SGTR Event with Overfill
15.6-3.2.g	Steam Generator Atmospheric Release Flow Rate (Ruptured Generator) Transient for SGTR Event with Overfill
15.6-3.2.h	Steam Generator Atmospheric Release Flow Rate (Intact Generators) Transient for SGTR Event with Overfill
15.6-3.2.i	Ruptured Steam Generator Break Flow Rate Transient for SGTR Event with Overfill
15.6-3.2.j	Auxiliary Feedwater Flow Rate and Narrow Range Level (Ruptured Generator) Transients for SGTR Event with Overfill
15.6-3.2.k	Auxiliary Feedwater Flow Rate and Narrow Range Level (Intact Generators) Transients for SGTR Event with Overfill
15.6-3.2.l	Ruptured Steam Generator Liquid Volume Transient for SGTR Event with Overfill
15.6-3.2.m	Pressurizer PORV Flow Rate Transient for SGTR Event with Overfill
15.6-3.2.n	Pressurizer Liquid Volume Transient for SGTR Event with Overfill

15.6.3.2.3 Radiological Consequences add underline

The analysis of the radiological consequences of the SGTR with overfill and water release is performed in a manner consistent with that presented in Section 15.6.3.1.3 for the SGTR with the postulated stuck open ARV. The assumptions are outlined below. Unless otherwise noted, these assumptions are consistent with the Section 15.6.3.1.3 analysis assumptions.

a. Short-term accident atmospheric dispersion factors and breathing rates are provided in Tables 15A-2 and 15A-1, respectively.

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b. Thyroid dose conversion factors (DCFs) from ICRP-30 ~~DCFs~~ (Reference 31) are used in place of the values in Table 15A-4. The whole body dose conversion factors used in the Section 15.6.3.1.3 analysis are replaced with those from Table III.1 of Federal Guidance Report 12 (Ref. ~~34~~), consistent with the change to ICRP-30 thyroid DCFs. This is consistent with ~~DG 1.113~~ (Ref. ~~33~~). ~~Ref. 34~~ does not list values for Kr-89 & Xe-137, so the Section 15.6.3.1.3 analysis values are retained. 32

Regulatory Guide 1.195 Reference 33

c. The initial reactor coolant system (RCS) iodine and noble gas concentrations are defined as in the Section 15.6.3.1.3 dose calculations, except that the ICRP-30 DCFs are used in determining the initial concentrations of the longer lived iodine isotopes.

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d. Spike modeling

- The accident-initiated iodine spike is modeled as in the Section 15.6.3.1.3 dose calculations, except that the spike factor of 335 allowed by ~~Regulatory Guide 1.183 (Reference 32) and Draft~~ Regulatory Guide ~~1.113~~ 1.195 (Reference ~~33~~ 32) is applied. This replaces the spike factor of 500 modeled in previous analysis, including the analysis presented in Section 15.6.3.1.3.
- The pre-accident iodine spike case spike is modeled as in the Section 15.6.3.1.3 dose calculations.

e. Initial secondary activity is 10% of the primary side activity modeled for the accident-initiated iodine spike.

f. Water/Steam Iodine Partitioning: Fluid released from the steam generators as steam retains a portion of the activity present in the fluid. The partition factor is 0.01. All activity contained in break flow that flashes to steam upon entering the SG is released without partitioning.

g. Activity released with water from ruptured SG = 50%. Activity contained in water released from the ruptured SG after overfill is not subject to partitioning. However, only 50% of the activity contained in the water is assumed to become airborne. No additional activity release due to evaporation is modeled. These assumptions were made in the analysis approved in Reference 3.

h. Break flow rate for iodine doses:

- The Section 15.6.3.1.3 dose analysis conservatively modeled a constant break flow rate. For the analysis of doses for the overfill case the transient break flow rate from the RETRAN analysis presented in Figure 15.6-3.2.i is used, up until the time when water relief stops. This is consistent with the analysis approved in Reference 3.
- The calculation of iodine doses until RHR conditions are reached conservatively assumes a break flow of 4 lbm/sec until 5 hours after break flow termination. This is consistent with the analysis approved in Reference 3. This portion of the analysis assumes that RHR conditions are achieved within 5 hours of break flow termination, even though the intact SG releases and the noble gas releases assumed 8 hours.

i. The Noble gas doses are calculated in Section 15.6.3.1.3 assuming a constant break flow rate of 55 lbm/sec for the first hour of the transient and 10 lbm/sec thereafter for 8 hours. For the analysis of the SGTR with overfill the duration of the 55 lbm/sec break flow is extended until 2 hours.

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j. Break flow flashing fraction

- The Section 15.6.3.1.3 dose analysis modeled conservative bounding values for the flashing fraction. For the analysis of doses for the overfill case the transient flashing fraction from the RETRAN analysis presented in Figure 15.6-3.2.e is used. This is consistent with the analysis approved in Reference 3. This analysis models the release of all the activity contained in the flashed break flow. This conservative assumption is consistent with Section 15.6.3.1.3 which modeled the direct release of activity in flashed break flow even after the ruptured SG's failed open atmospheric relief valve (ARV) was isolated. steam dump (ASD) valve
- Primary to secondary leakage flashing is not modeled. This is consistent with the analysis approved in Reference 3. The leak is small and it is assumed that any steam bubbles formed by flashing leakage would collapse before reaching the top of the water level.

k. The ruptured SG releases are modeled using the RETRAN analysis results presented in Figure 15.6-3.2.g. In the calculation of doses for the cooldown to RHR conditions it is assumed that a steam flow rate of 8 lbm/sec is maintained due to the failed open safety valve. Thus, 144,000 lbm of steam is released from the ruptured SG in the 5 hours from break flow termination until RHR conditions are reached.

l. The intact SG releases are modeled using the RETRAN analysis results presented in Figure 15.6-3.2.h. In the calculation of doses ~~for the cooldown~~ to RHR conditions it is assumed that 4000 ~~lbm of~~ steam is released from the intact SGs from the time of break flow termination. This value was conservatively calculated

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for the case with the failed open ARV presented in Section 15.6.3.1.3 and remains conservative when applied to the analysis of the SGTR with overfill.

m. The calculation of the secondary activity in the analysis presented in the Section 15.6.3.1.3 only credits reduction by decay. I.e., The release of activity from the secondary side (steam and water) is not credited to reduce the secondary activity. This is an overly conservative assumption ~~when significant~~ water release occurs. The analysis for the SGTR with overfill and water relief accounts for the activity that leaves the SG in the water.

n. The reactor coolant system, ruptured steam generator and intact steam generators' masses are modeled using the RETRAN analysis results from Figure 15.6-3.2.d. This is consistent with the analysis approved in Reference 3. The analysis presented in the Section 15.6.3.1.3 modeled conservative bounding values for the RCS and secondary masses.

Table 15.6-5a lists the offsite doses for the SGTR with overfill and water release.

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15.6.3.3

~~15.6.3.3~~ Conclusions

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15.6.3.3.1 ~~15.6.3.3.1~~ Filter Loadings

The only ESF filtration system considered in the analysis which limits the consequences of the steam generator tube rupture is the control room filtration system. Activity loadings on the control room charcoal filter are based on flow rate through the filter, concentration of activity at the filter inlet, and filter efficiency.

Activity in the control room filter as a function of time has been evaluated for the LOCA, Section 15.6.5. Since the control room filters are capable of accommodating the potential design-basis LOCA fission product iodine loadings, more than adequate design margin is available with respect to postulated SGTR accident releases.

~~15.6.3.3.2~~ Doses to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

15.6.3.3.2

The potential radiological consequences resulting from the occurrence of a postulated SGTR have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2 hour period at the exclusion area boundary and for a time period effectively greater than the duration of the accident (0 to 8 hours) at the low-population zone outer boundary. ~~The results are listed in Table 15.6-5. The resultant doses are within the guideline values of 10 CFR 100.~~

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15.6.3.4 Conclusions

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 simultaneous loss of offsite power.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to the Callaway Plant.

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total

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Two potentially limiting failure scenarios have been analyzed. Table 15.6-5 presents the offsite dose results for the case of an SGTR with a stuck-open ASD for the ruptured steam generator. Table 15.6-5a presents the offsite dose results for the case of an SGTR with the postulated failure of the ruptures steam generator AFW flow control valve. For both scenarios, the doses considering a pre-accident iodine spike are within the guideline values of 10 CFR 100. For both scenarios, the doses considering an accident-initiated iodine spike are within the 10% of the guideline values of 10 CFR 100.

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30. Huegel, D. S., et. al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Reactor Non-LOCA Safety Analyses", WCAP-14882-P-A (Proprietary), April 1999.

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~~32. Regulatory Guide RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.~~

32. ~~33.~~ ~~Draft~~ Regulatory Guide ~~DG-1.113~~ 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," ~~January 2002.~~ May 2003.

33. ~~34.~~ K.F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

TABLE 15.6-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	26.2
	~~Rods begin to~~ drop	28.2
	Minimum DNBR occurs	28.4
Steam generator tube rupture with stuck-open atmospheric steam dump (ASD) valve	Tube rupture occurs	0.0
	Reactor trip signal	613.3
	Safety injection signal	613.3
	Rod motion	615.3
	Feedwater terminated	617.6
	Ruptured steam generator atmospheric/ steam dump valve opens	625.6
	Safety injection begins	628.3
	Auxiliary feedwater injection	675.3
	Operator isolates faulted steam generator by closing manual block valve	1826
	Operator initiates RCS cooldown via intact steam generator atmospheric steam dump valves	2447
	Operator completes RCS cooldown	3288
	Operator initiates RCS depressurization via pressurizer PORVs	3469
	Operator completes RCS depressurization	3558
	Operator terminates safety injection	3739
	Operator equalizes primary-secondary pressure	4039
	RHR cut-in conditions reached	18000

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Steam generator tube rupture with overfill	Tube rupture occurs	0.
	Reactor trip signal and loss of offsite power	0.
	Safety injection signal	0.
	Auxiliary feedwater injection starts	5.
	Safety injection delivered	15.
	Feedwater terminated	17.
	Operator terminates auxiliary feedwater from TDAFW pump to ruptured steam generator	600.
	Ruptured steam generator water relief begins	1149.
	Operator terminates auxiliary feedwater from MDAFW pump to ruptured steam generator	1200.
	Operator initiates RCS cooldown via intact steam generator atmospheric steam dump valves	1800.
	Operator completes RCS cooldown	2188.
	Operator initiates RCS depressurization via pressurizer PORVs	2280.
	Operator completes RCS depressurization	2401.
	Operator terminates safety injection	2701.
	Operator equalizes primary-secondary pressure	3623.
	Cooldown to RHR cut-in begins	3600
	Ruptured SG safety valve begins to relieve steam	4560.
	RHR cut-in conditions reached	21600.

TABLE 15.6-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)
Inadvertent opening of a pressurizer safety valve	Safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	26.2
	Rods begin to drop	28.2
	Minimum DNBR occurs	28.4
Steam generator tube rupture	Tube rupture occurs	0.0
	Reactor trip signal	613.3
	Safety injection signal	613.3
	Rod motion	615.3
	Feedwater terminated	617.6
	Ruptured steam generator atmospheric/ steam dump valve opens	625.6
	Safety injection begins	628.3
	Auxiliary feedwater injection	675.3
	Operator isolates faulted steam generator by closing manual block valve	1826
	Operator initiates RCS cooldown via intact steam generator atmospheric steam dump valves	2447
	Operator completes RCS cooldown	3288
	Operator initiates RCS depressurization via pressurizer PORVs	3469
	Operator completes RCS depressurization	3558
	Operator terminates safety injection	3739
	Operator equalizes primary-secondary pressure	4039
	RHR cut-in conditions reached	18000

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The Main Feed isolation time listed in this table should be increased by 10 seconds to support the plant design change which installed system-medium actuators on the Main Feedwater Isolation Valves.

TABLE 15.6-5 RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

WITH STUCK-OPEN ATMOSPHERIC STEAM DUMP VALVE

	Doses (rem)
1. Case 1, accident initiated iodine spike	
Exclusion Area Boundary (0-2 hr)	
Thyroid	22.9
Whole body	0.643
Low Population Zone Outer Boundary (duration)	
Thyroid	2.29
Whole body	0.0685
2. Case 2, pre-accident iodine spike	
Exclusion Area Boundary (0-2 hr)	
Thyroid	34.3
Whole body	0.324
Low Population Zone Outer Boundary (duration)	
Thyroid	3.43
Whole body	0.0367

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Table 15.6-5a

RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE WITH OVERFILL

1. Accident initiated iodine spike	Doses (rem)
Exclusion Area Boundary (0-2 hr)	
Thyroid	13.4
Whole Body	0.396
Low Population Zone Outer Boundary	
Thyroid	1.43
Whole Body	0.0424
2. Pre-accident iodine spike	Doses (rem)
Exclusion Area Boundary (0-2 hr)	
Thyroid	46.2
Whole Body	0.362
Low Population Zone Outer Boundary	
Thyroid	4.71
Whole Body	0.0385

FIGURE 15.6-3P

STEAM GENERATOR FEEDWATER FLOW RATE TRANSIENT

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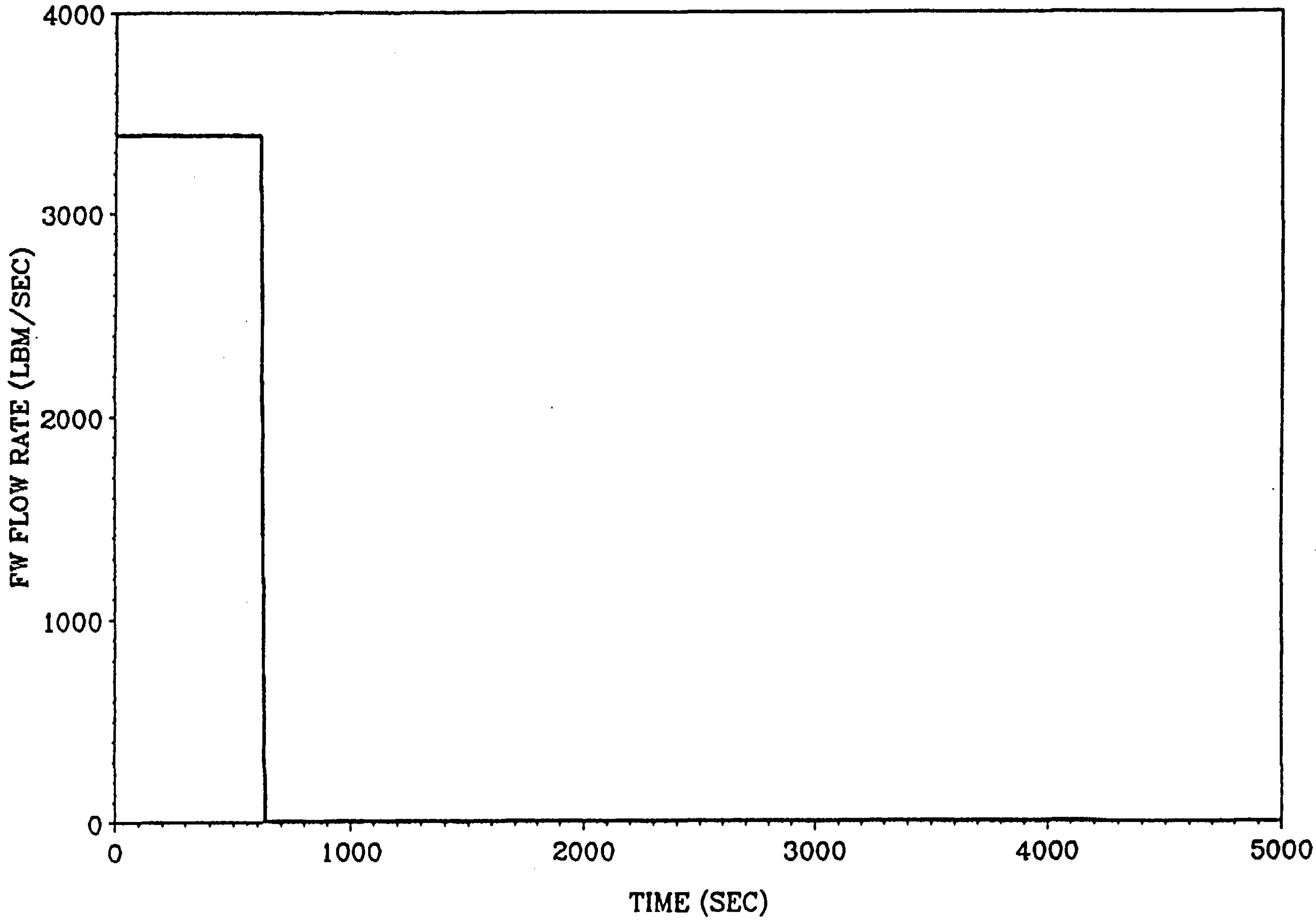

STEAM GENERATOR TUBE RUPTURE EVENT: —— INTACT GENERATORS

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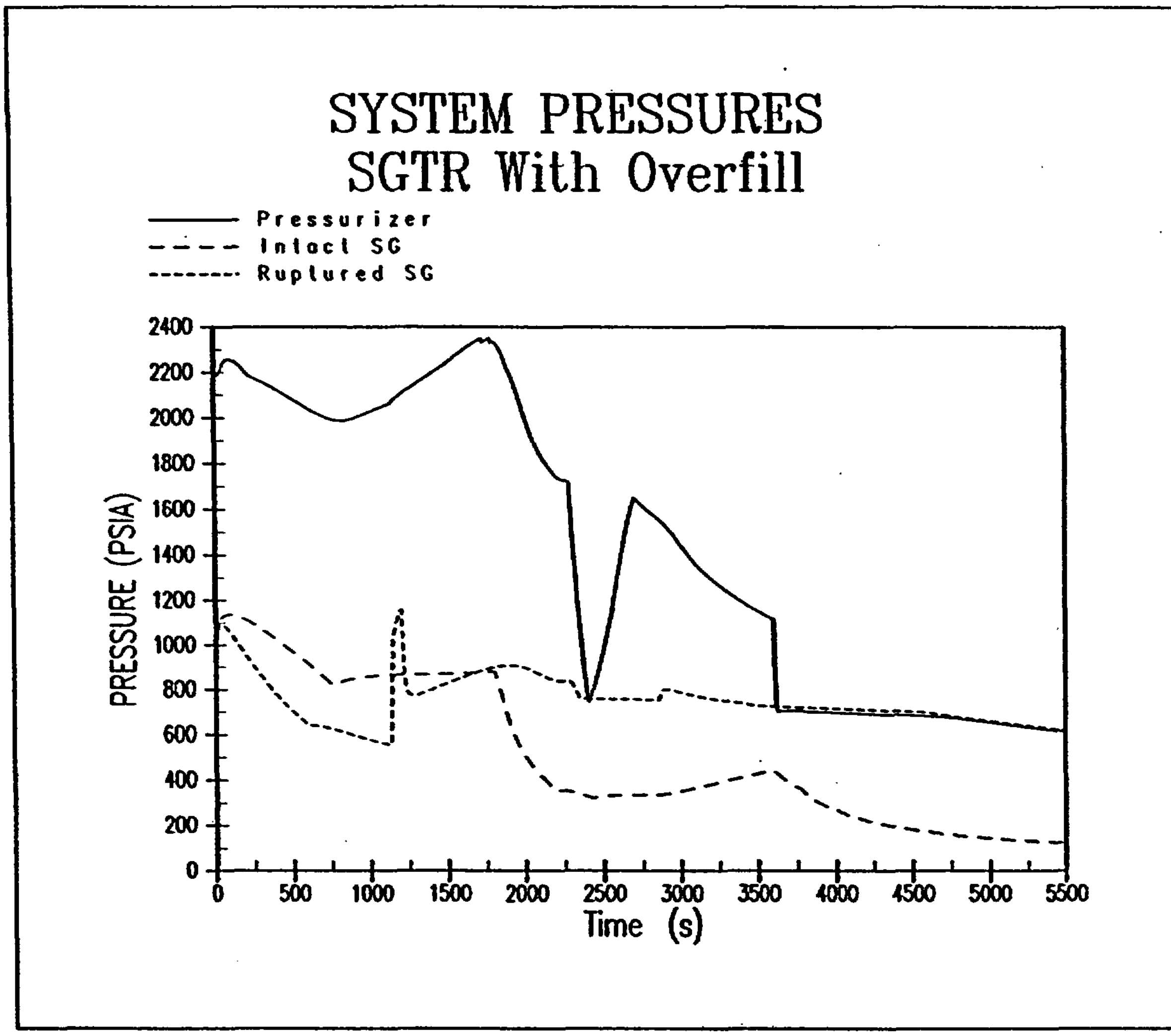

Figure 15.6-3.2.a: Pressurizer and Steam Generator (Ruptured and Intact Generators) Pressure Transients for SGTR Event with Overfill

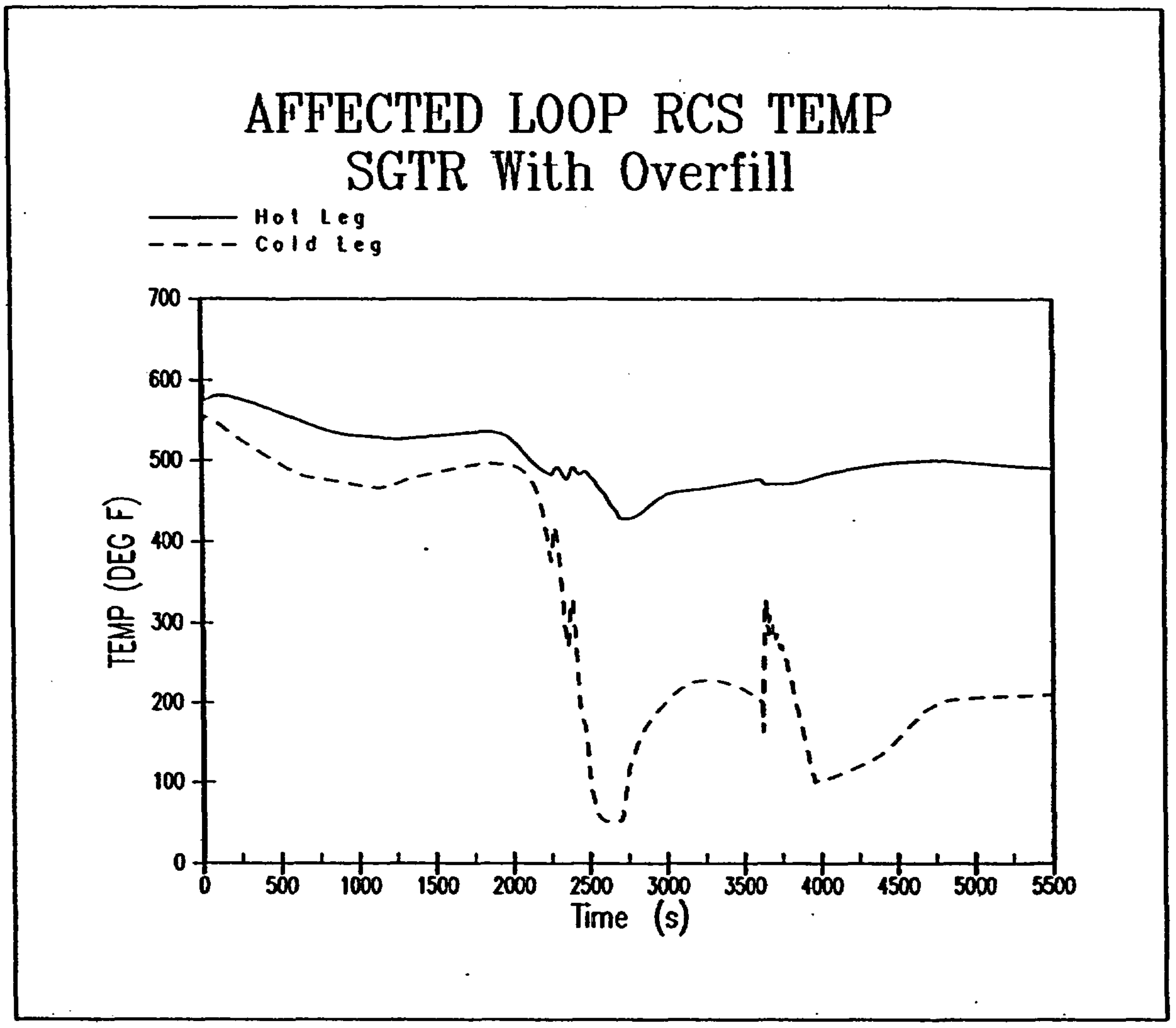

Figure 15.6-3.2.b: Reactor Coolant System Temperature (Ruptured Loop) Transient for SGTR Event with Overfill

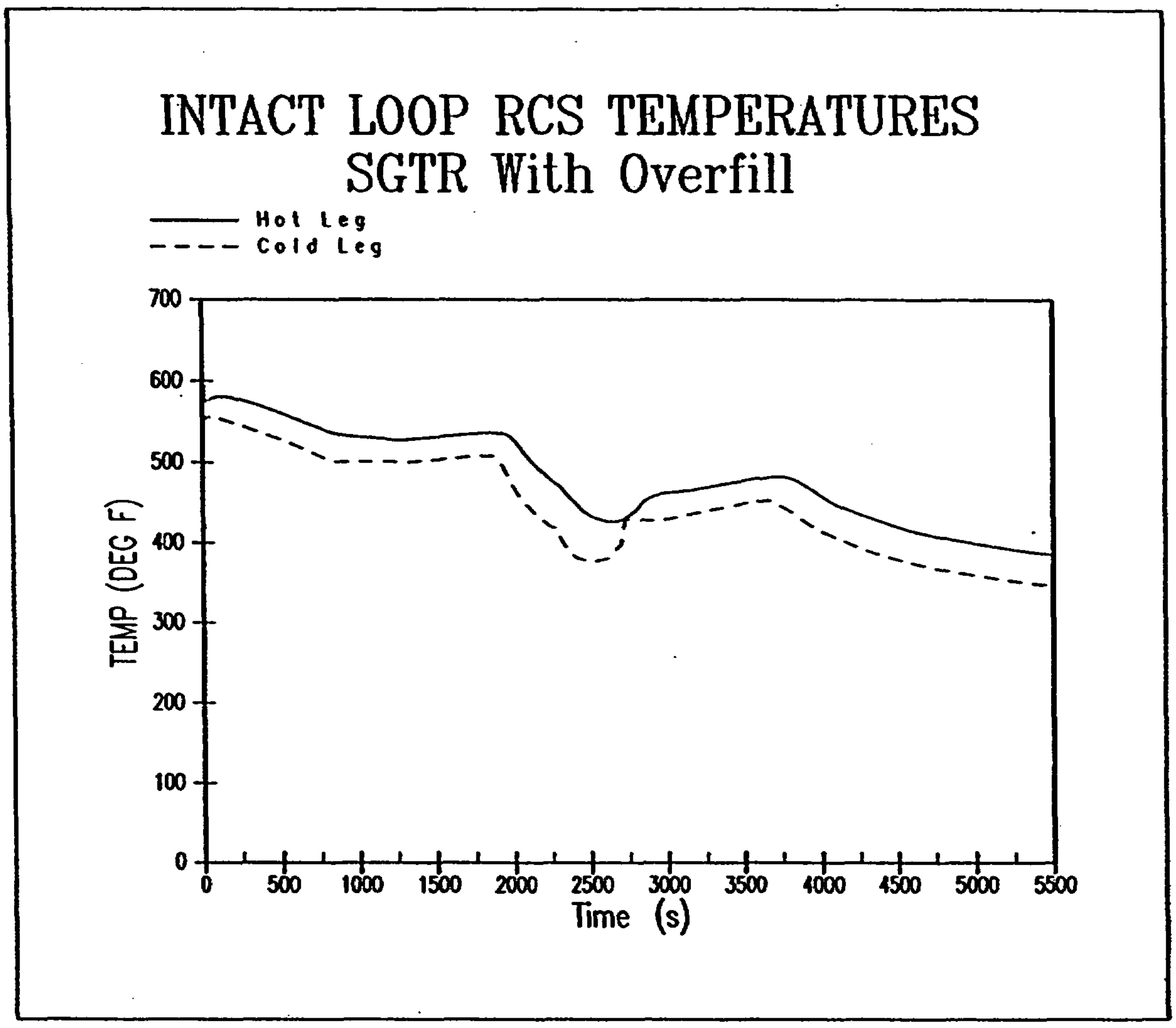

Figure 15.6-3.2.c: Reactor Coolant System Temperature (Intact Loops) Transient for SGTR Event with Overfill

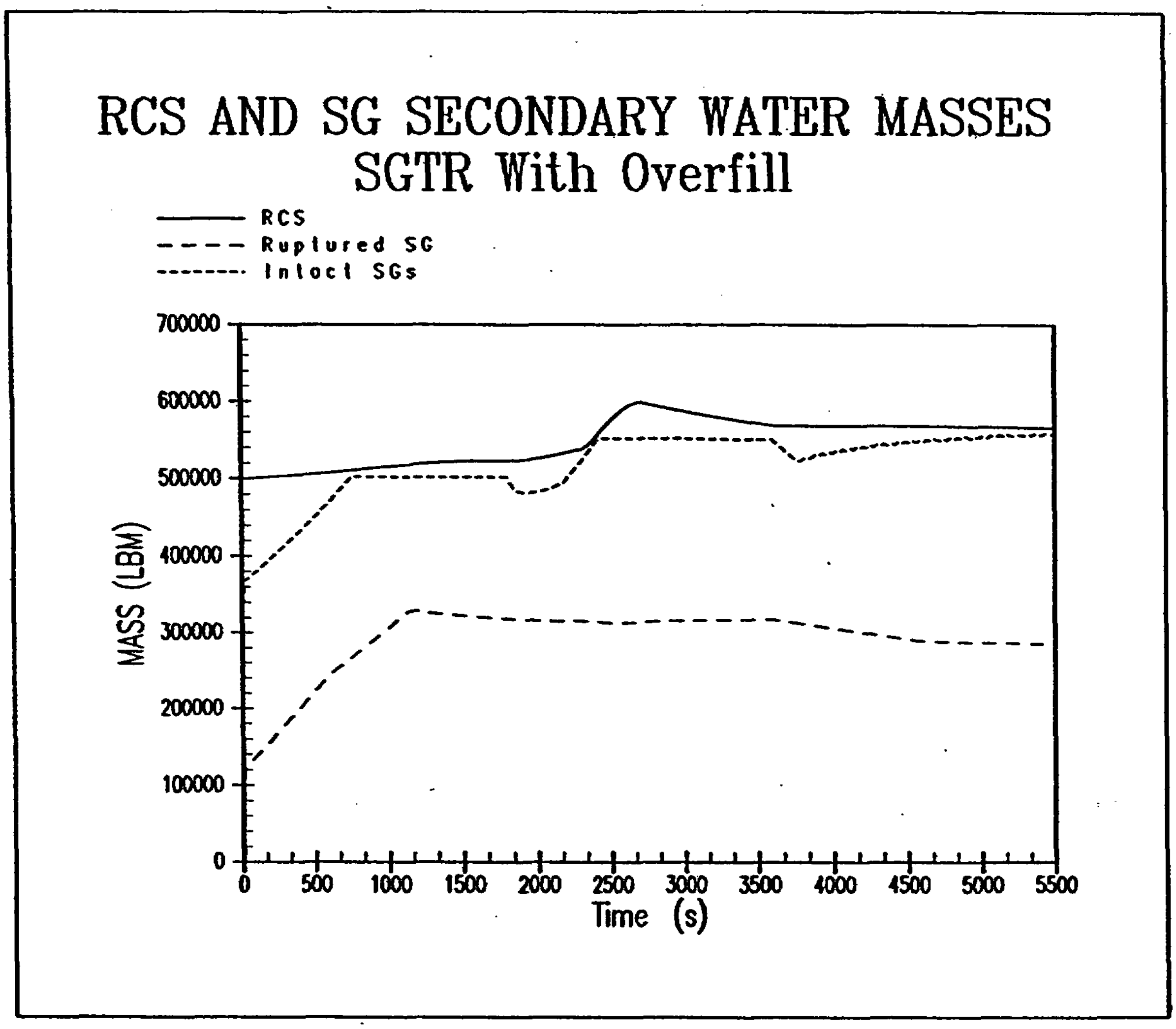

Figure 15.6-3.2.d: Reactor Coolant System and Steam Generator (Ruptured and Intact Generators) Water Mass Transient for SGTR Event with Overfill

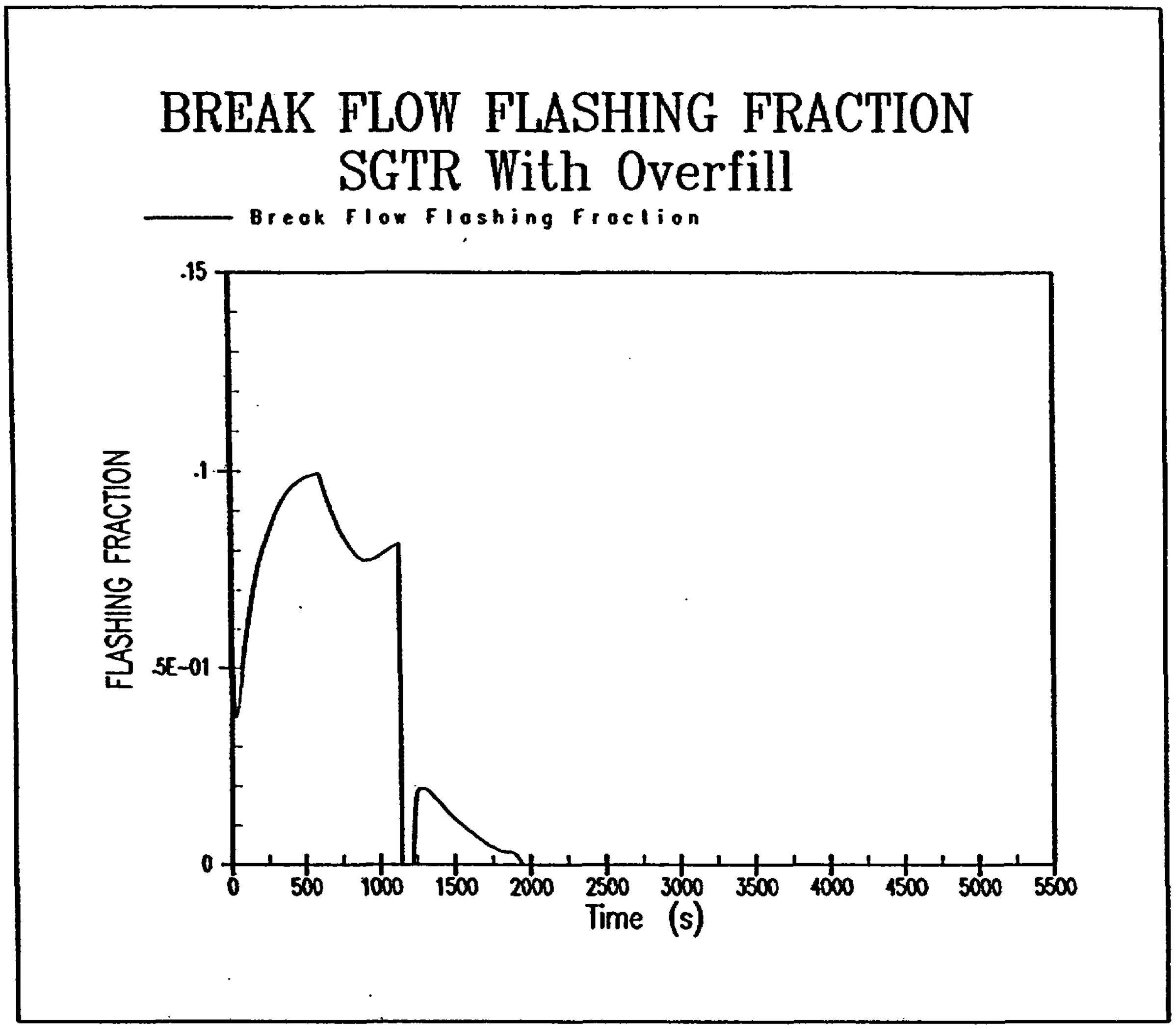

Figure 15.6-3.2.e: Ruptured Steam Generator Break Flow Flashing Fraction Transient for SGTR Event with Overfill

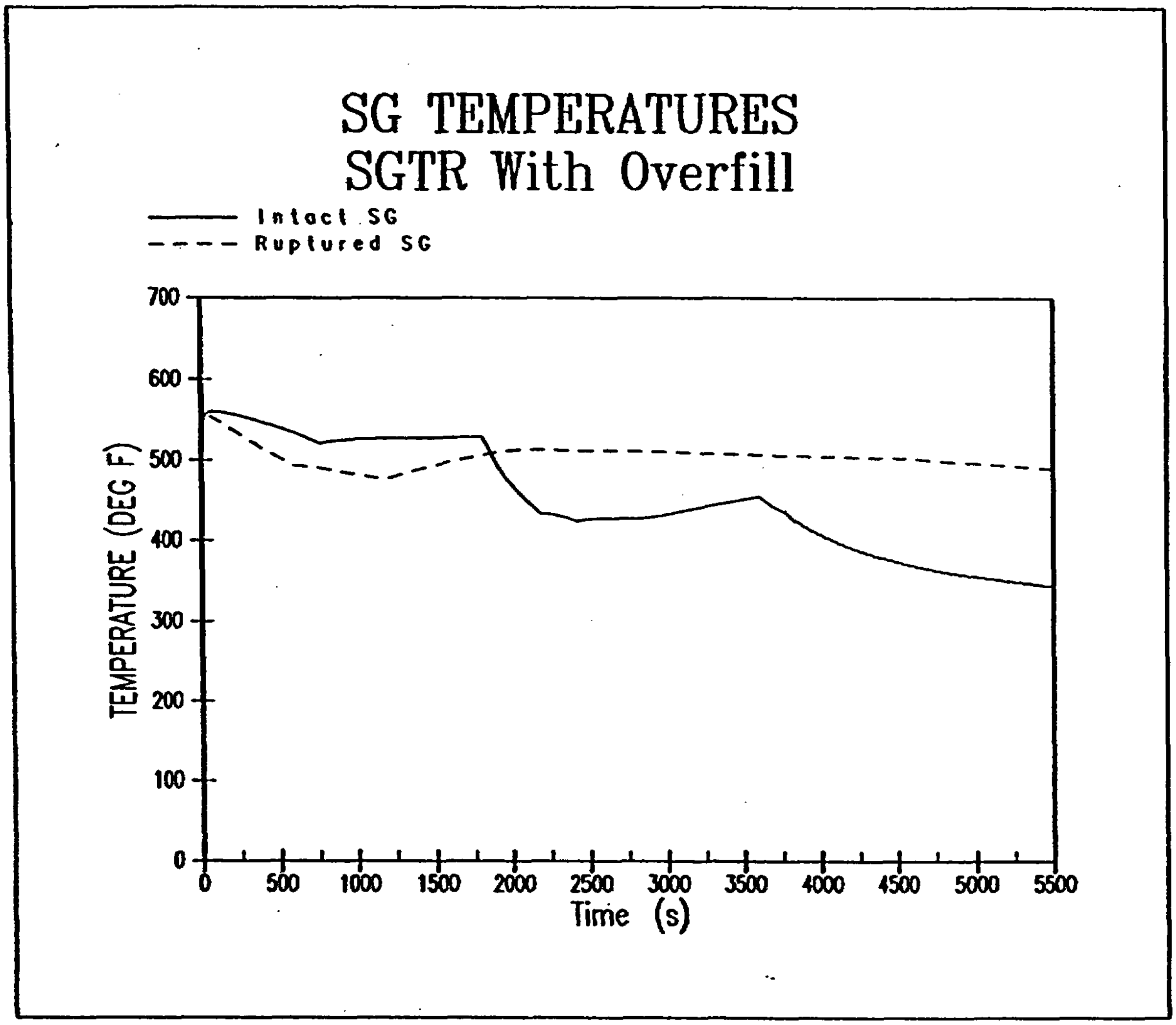

Figure 15.6-3.2.f: Steam Generator Temperature (Ruptured and Intact Generators) Transient for SGTR Event with Overfill

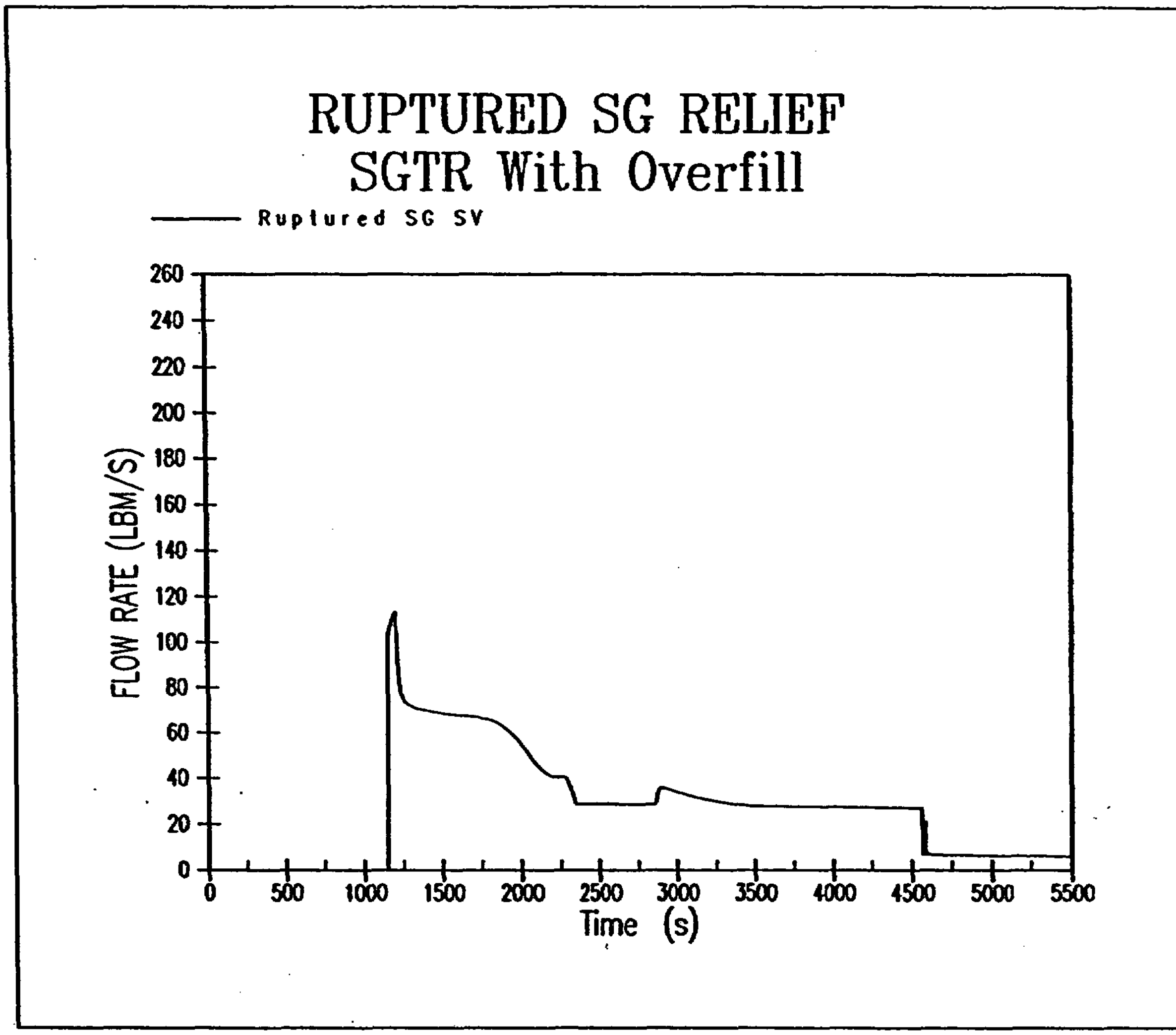

Figure 15.6-3.2.g: Steam Generator Atmospheric Release Flow Rate (Ruptured Generator) Transient for SGTR Event with Overfill

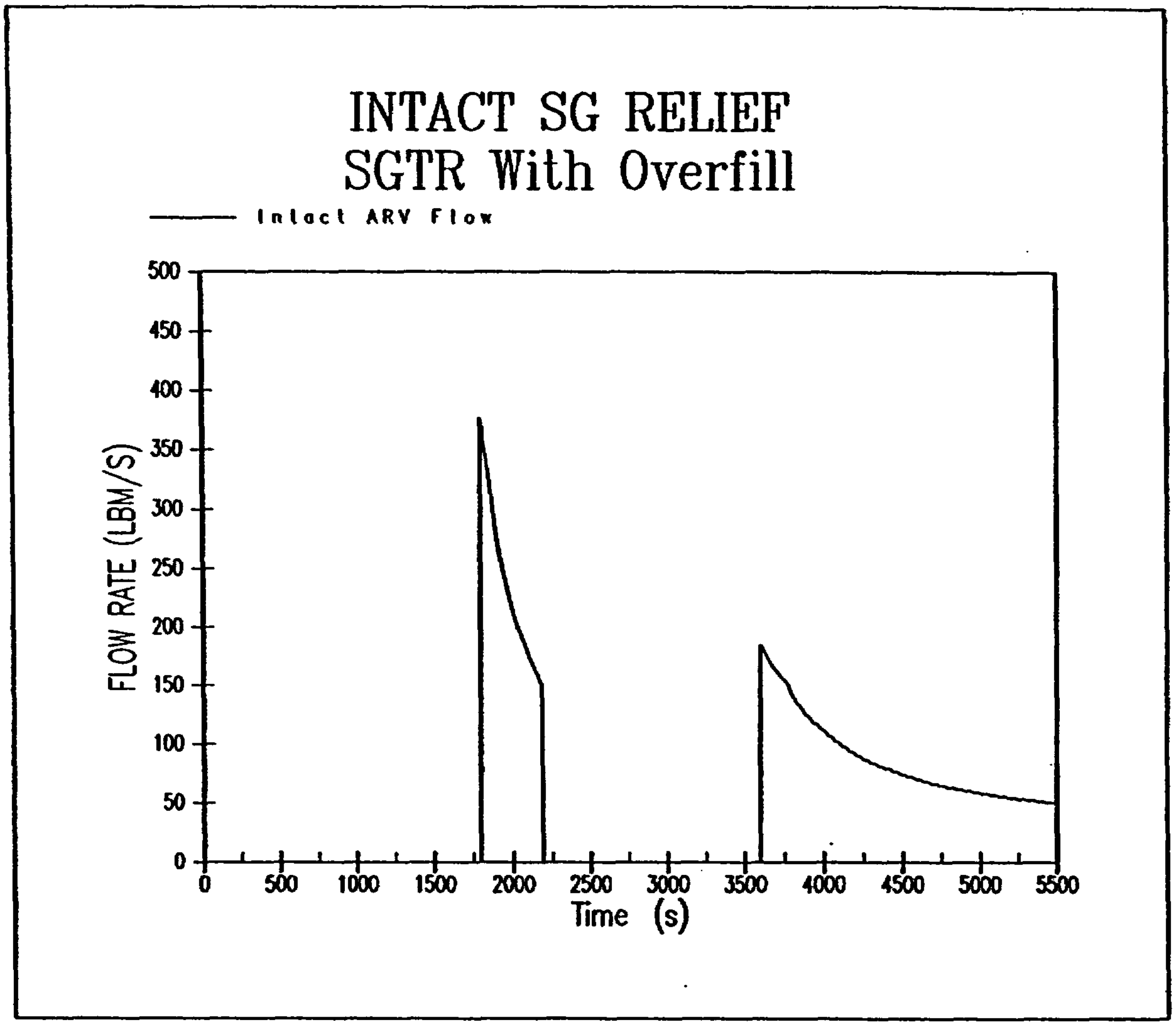

Figure 15.6-3.2.h: Steam Generator Atmospheric Release Flow Rate (Intact Generators) Transient for SGTR Event with Overfill

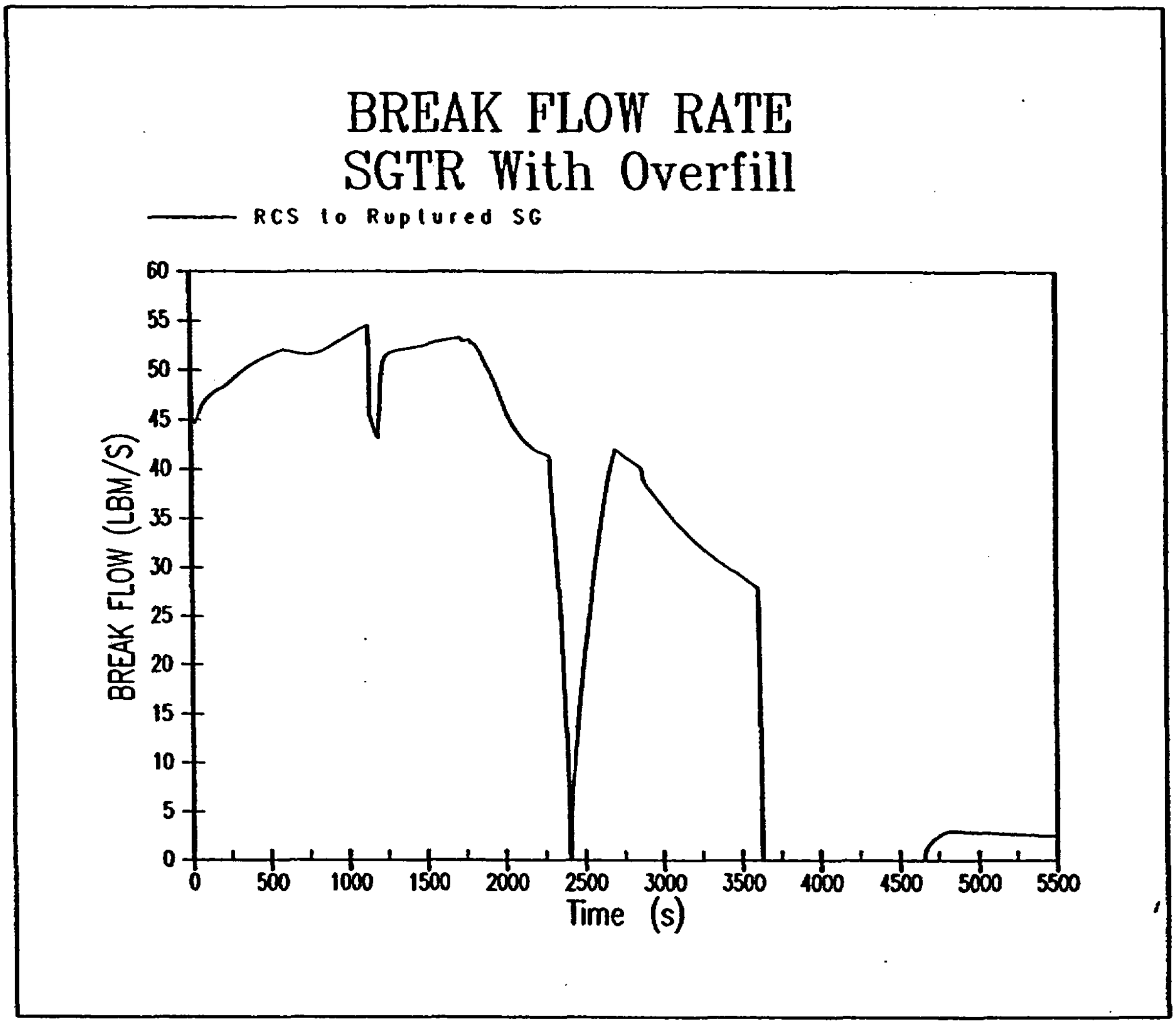

Figure 15.6-3.2.i: Ruptured Steam Generator Break Flow Rate Transient for SGTR Event with Overfill

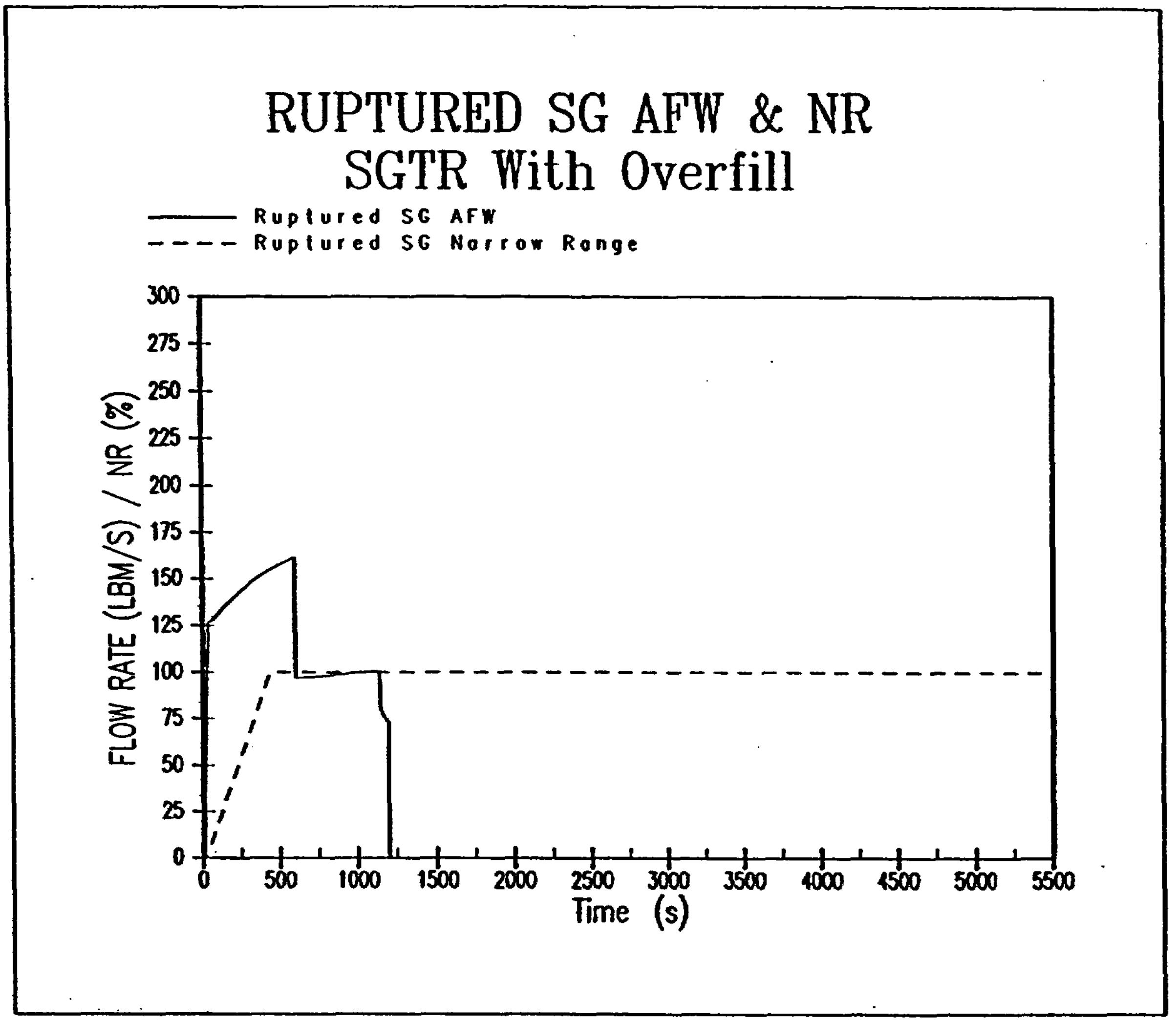

Figure 15.6-3.2.j: Auxiliary Feedwater Flow Rate and Narrow Range Level (Ruptured Generator) Transients for SGTR Event with Overfill

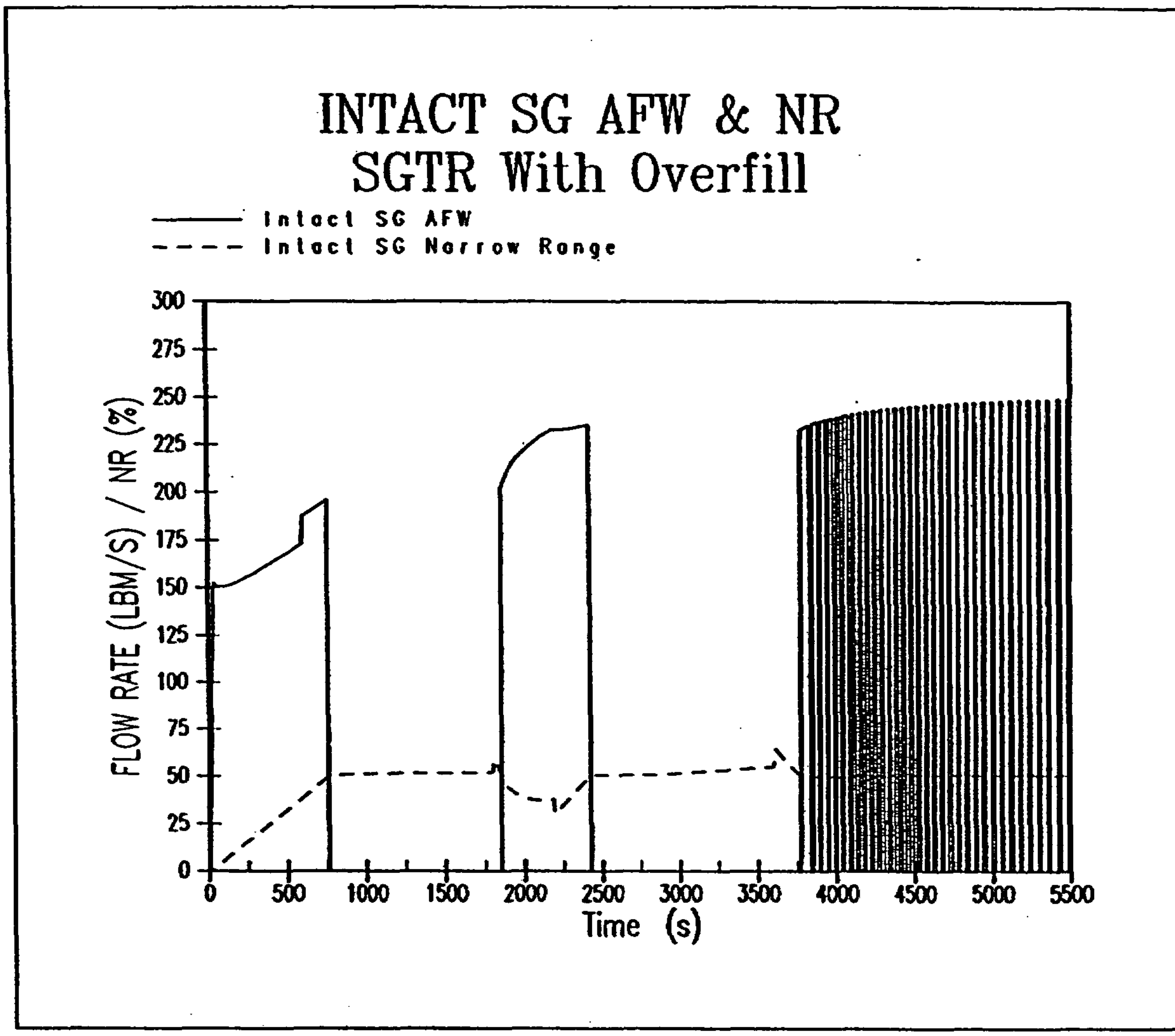

Figure 15.6-3.2.k: Auxiliary Feedwater Flow Rate and Narrow Range Level (Intact Generators) Transients for SGTR Event with Overfill

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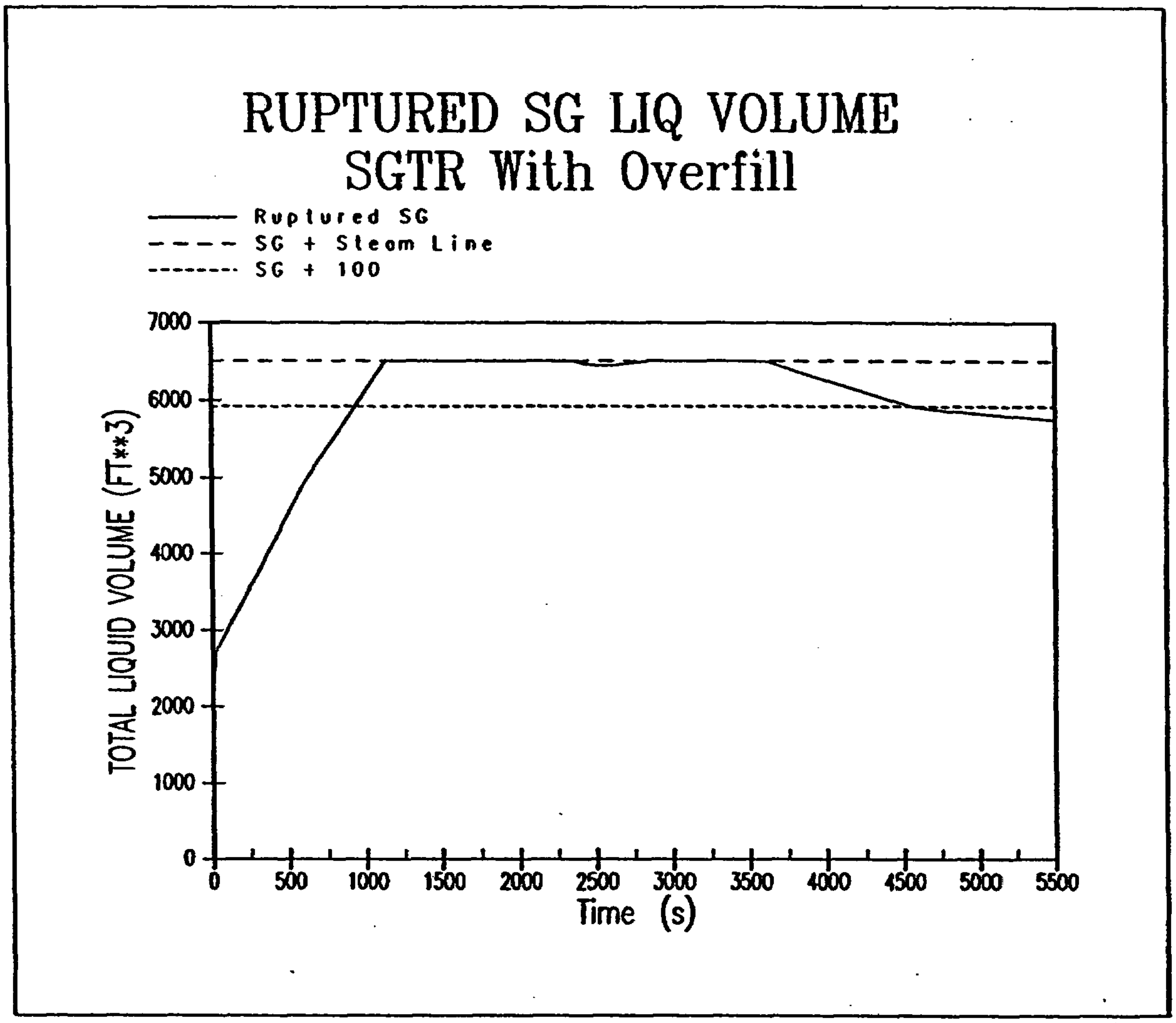

Figure 15.6-3.2.1: Ruptured Steam Generator Liquid Volume Transient for SGTR Event with Overfill

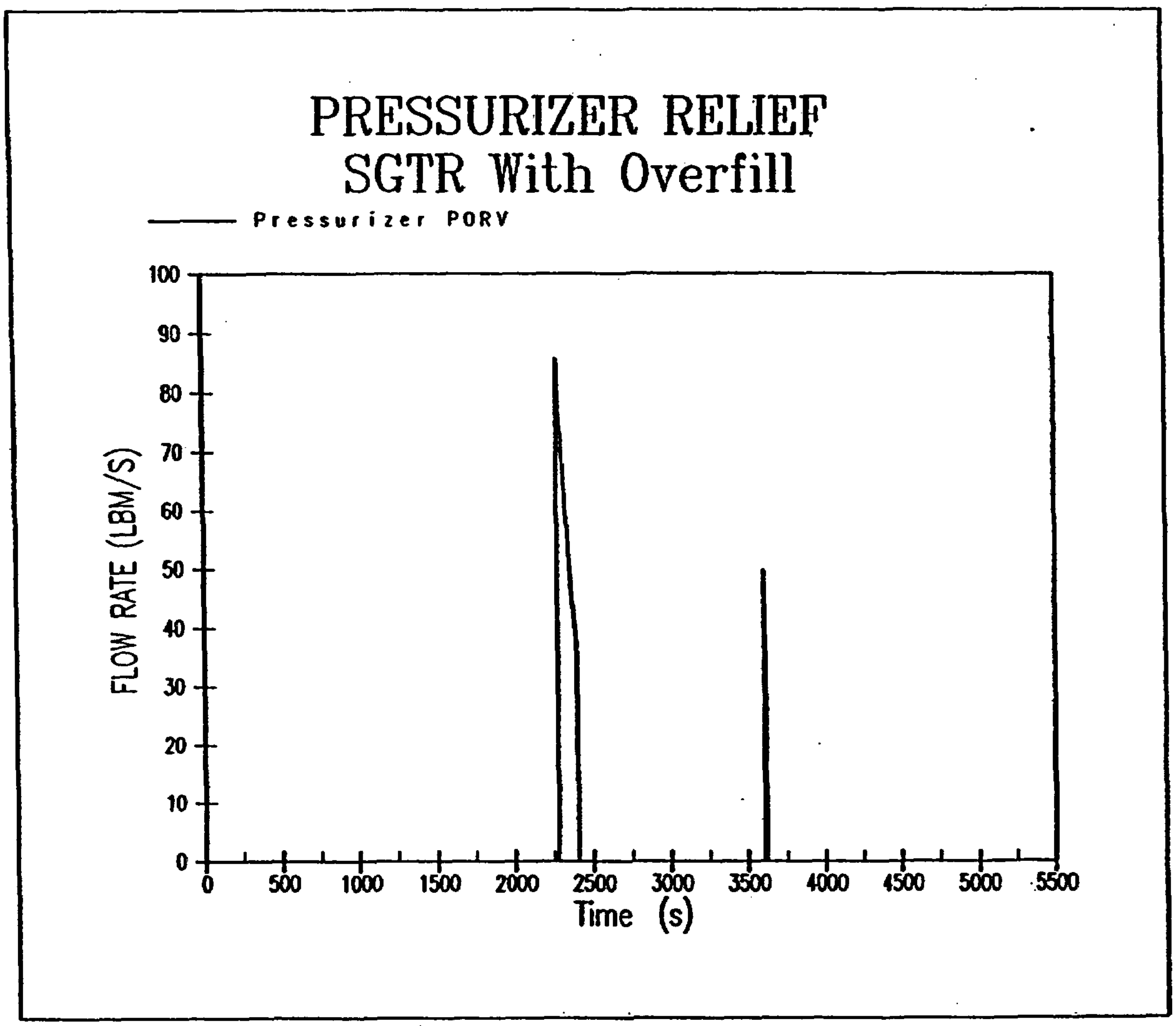

Figure 15.6-3.2.m: Pressurizer PORV Flow Rate Transient for SGTR Event with Overfill

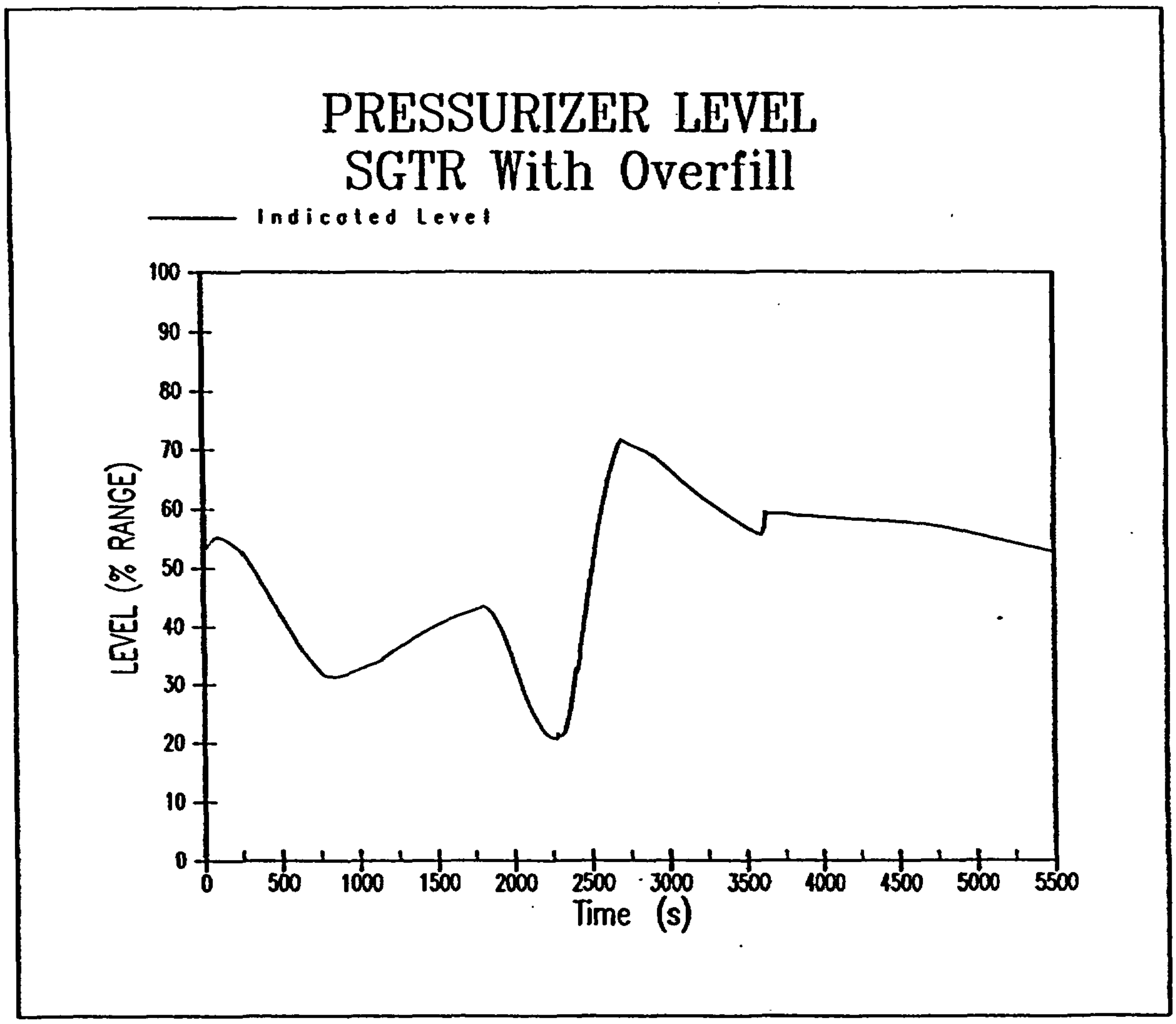

Figure 15.6-3.2.n: Pressurizer Liquid Volume Transient for SGTR Event with Overfill

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TABLE 15A-4 DOSE CONVERSION FACTORS USED IN ACCIDENT ANALYSIS

FSAR CN 03-023

Nuclide	Total Body Rem-meter^3 / Ci-sec	Beta Skin Rem-meter^3 / Ci-sec	Thyroid * Rem/Ci
I-131	8.72E-2	3.17E-2	1.49E+6
I-132	5.13E-1	1.32E-1	1.43E+4
I-133	1.55E-1	7.35E-2	2.69E+5
I-134	5.32E-1	9.23E-2	3.73E+3
I-135	4.21E-1	1.29E-1	5.60E+4
Kr-83m	2.40E-6	0	NA
Kr-85m	3.71E-2	4.63E-2	NA
Kr-85	5.11E-4	4.25E-2	NA
Kr-87	1.88E-1	3.09E-1	NA
Kr-88	4.67E-1	7.52E-2	NA
Kr-89	5.27E-1	3.20E-1	NA
Xe-131m	2.91E-3	1.51E-2	NA
Xe-133m	7.97E-3	3.15E-2	NA
Xe-133	9.33E-3	9.70E-3	NA
Xe-135m	9.91E-2	2.25E-2	NA
Xe-135	5.75E-2	5.90E-2	NA
Xe-137	4.51E-2	3.87E-1	NA
Xe-138	2.80E-1	1.31E-1	NA

* ← INSERT K

FSAR CN 03-023

FSAR 03-023

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The SGTR-Overfill radiological consequences have been re-analyzed using the following whole body and thyroid dose conversion factors from ICRP-30. These factors may be applied to other accident sequences as they are re-analyzed:

Nuclide	Total Body $\frac{\text{REM-meter}^3}{\text{Ci-sec}}$	Thyroid Rem/Ci
I-131	8.72E-02	1.07E+06
I-132	5.13E-01	6.29E+03
I-133	1.55E-01	1.81E+05
I-134	5.32E-01	1.07E+03
I-135	4.21E-01	3.14E+04
Kr-83m	2.40E-06	NA
Kr-85m	3.07E-02	NA
Kr-85	4.84E-04	NA
Kr-87	1.46E-01	NA
Kr-88	3.70E-01	NA
Kr-89	5.27E-01	NA
Xe-131m	1.52E-03	NA
Xe-133m	5.53E-03	NA
Xe-133	6.24E-03	NA
Xe-135m	7.75E-02	NA
Xe-135	4.82E-02	NA
Xe-137	1.98E-01	NA
Xe-138	2.80E-01	NA

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ATTACHMENT 7

SUMMARY OF REGULATORY COMMITMENTS

SUMMARY OF REGULATORY COMMITMENTS

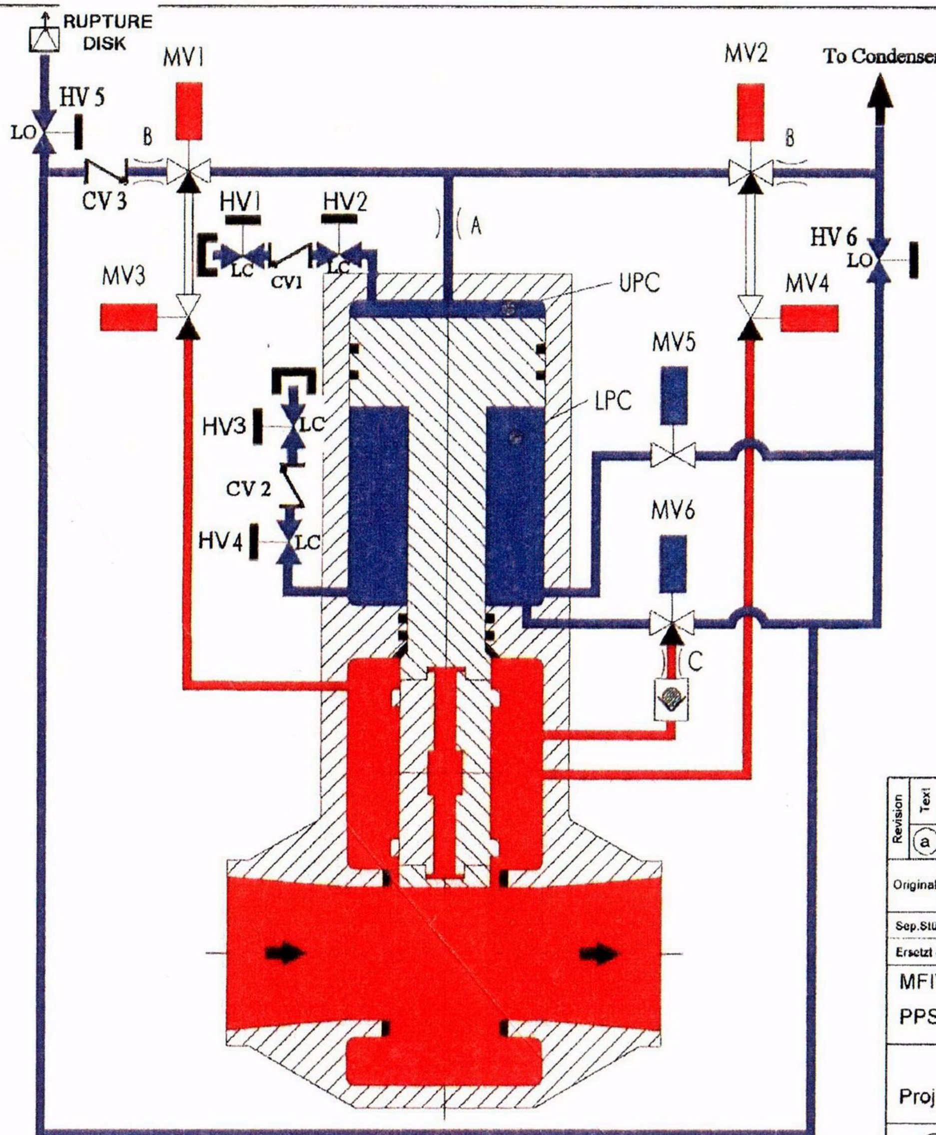
The following table identifies those actions committed to by AmerenUE, Callaway Plant in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Dave E. Shafer, Superintendent, Licensing at AmerenUE, (314) 554-3104.

COMMITMENT	Due Date/Event
The approved amendment will be implemented prior to restart of the unit following Refueling Outage 13. This means that all required procedure and document revisions shall be completed prior to restart of the unit following Refuel 13.	Prior to restart following Refueling 13
The associated FSAR and TS Bases revisions, as approved by plant review programs performed under 10 CFR 50.59, 10 CFR 50.71(e), and TS 5.5.5, will be incorporated into the next licensing document regulatory update.	6 months following the end of Refuel 13
The NRC approved reanalysis for the SGTR with overfill accident will be incorporated into the FSAR during the next licensing document regulatory update.	6 months following the end of Refuel 13

ULNRC- 04592

ATTACHMENT 8

DIAGRAM FOR MFIV SYSTEM-MEDIUM OPERATED ACTUATOR



system medium pressure

pressureless(drainage pressure)

Auxiliary medium

UPC ... upper piston chamber
LPC ... lower piston chamber
MV1-4 normally energized
MV5+MV6 normally de-energized
MV1+MV3+MV5 el. power supply train 1
MV2+MV4+MV6 el. power supply train 2

plug connection for aux. medium

de-energized

energized

Bearbeitungsvorschriften siehe Zeichnungsnormen.

Revision / Text								
a	SCHMID_R	6-8-2001						
Original	BRUNNE_R Gezeichnet:	26-7-2001 Datum	Geprüft/SBK:	Datum	Qualität/SBQ:	Datum	Auftragsleitung:	Datum

Sep.Stückverz. Ja | ähnl. Zchg.

Ersetzt durch: | Ersatz für: | Original Massstab: 1:1

MFIV / MSIV
PPS - Actuator

Proj. Callaway

System medium operated actuator
for quick closing valve
Layout diagram

SolidWorks NORMAL OPERATION (keep open)

CCi AG, CH-8404 Winterthur

D | A3 | 103.224.185 | 1 - 8 a

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