Omaha Public Power District

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July 1, 2005
LIC-05-0001

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

Reference: Docket No. 50-285

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event"

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby transmits an application for revision of the Fort Calhoun Station Unit 1 (FCS) Updated Safety Analysis Report (USAR) Section 14.10 for operator action during the Loss of Main Feedwater (LMFW) Event. The proposed change will amend the design and licensing basis by revising the USAR to describe an existing Emergency Operating Procedure (EOP) operator action to isolate steam generator blowdown within 15 minutes of reactor trip during a loss of main feedwater event. This operator action ensures that the auxiliary feedwater system performs its design function of maintaining adequate SG water level for decay heat removal once the auxiliary feedwater actuation signal (AFAS) is actuated. Based on an OPPD evaluation under 10 CFR 50.59, "Changes, Tests, and Experiments," a license amendment is required.

The proposed USAR change has been evaluated in accordance with 10 CFR 50.91(a) (1) using criteria in 10 CFR 50.92(c) to determine that the change does not involve significant hazard considerations. The bases for this determination, information supporting the change, a no significant hazards consideration, and an environmental consideration are included in Attachment 1. Attachments 2 and 3 provide the proposed USAR Section 14.10 in marked up and clean form. Attachment 4 provides the associated AFAS Setpoint Verification Analysis.

The operator action is addressed within an existing procedure, EOP-00, Standard Post Trip Actions, which has been in place since institution of the EOPs and routinely used for simulator training exercises. OPPD plans to add the 15-minute limitation.

OPPD requests approval of the proposed change by September 30, 2006 with a 90-day implementation period to support plant operations following the 2006 outage during which the steam generators, pressurizer, and reactor vessel head will be replaced. No commitments are made to the NRC in this letter.

A001
A053

I declare under penalty of perjury that the forgoing is true and correct. (Executed July 1, 2005)

If you have any questions or require additional information, please contact T.C. Matthews at 402-533-6938.

Sincerely,

S. K. Gambhir
Division Manager
Nuclear Projects

SKG/RLJ/rar

Attachments:
1. OPPD's Evaluation of the Proposed Change
2. Markup of Affected USAR Pages
3. Proposed Revised USAR Section (clean)
4. FCS RSG -Auxiliary Feedwater Actuation Signal (AFAS) Setpoint Verification

cc: Division Administrator - Public Health Assurance, State of Nebraska

ATTACHMENT 1

OPPD's Evaluation of the Proposed Change

1.0 DESCRIPTION

2.0 PROPOSED CHANGE

3.0 BACKGROUND

4.0 TECHNICAL ANALYSIS

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

5.2 Applicable Regulatory Requirements/Criteria

6.0 ENVIRONMENTAL CONSIDERATION

7.0 REFERENCES

1.0 DESCRIPTION

The proposed change will amend the design and licensing basis by revising the Updated Safety Analysis Report (USAR) to credit existing Emergency Operating Procedure (EOP) operator action to isolate steam generator (SG) blowdown within 15 minutes of reactor trip during a loss of main feedwater flow event. This operator action ensures that the auxiliary feedwater system performs its design function of maintaining adequate SG water level for decay heat removal once the auxiliary feedwater actuation signal (AFAS) is actuated.

2.0 PROPOSED CHANGE

The proposed change adds clarification to the USAR section 14.10.1 description of the Loss of Feedwater Flow (LFWF) event by crediting manual isolation of steam generator blowdown within 15 minutes of reactor trip. The Loss of Feedwater Flow event described in the USAR and Loss of Main Feedwater (LMFW) event analyzed by AREVA for the post 2006 outage fuel cycles are the same event, i.e., the event titles are used interchangeably. The title of event as discussed in this submittal will be the Loss of Main Feedwater, however, to be consistent with existing FCS analysis documentation, the title, Loss of Feedwater Flow event, will continue to be used in the USAR. These changes remain valid for subsequent fuel cycles.

The specific change is included with the text of USAR Section 14.10.1 in marked up and clean form in Attachments 2 & 3 of this letter.

3.0 BACKGROUND

OPPD plans to replace the Fort Calhoun Station (FCS) steam generators (SGs), reactor vessel head, and pressurizer in 2006. To support plant operation with the Replacement Steam Generators (RSGs), Replacement Reactor Vessel Head (RRVH), and Replacement Pressurizer (RPZR), the Loss of Main Feedwater Flow (LMFW), and the Feedwater Line Break (FWLB) events were re-analyzed to determine the adequacy of the auxiliary feedwater system to remove decay heat and to verify the adequacy of the auxiliary feedwater actuation signal (AFAS) setpoints. The FWLB analysis results met the acceptance criteria of the current analysis of record; however, the LMFW analysis concluded that to meet the acceptance criteria, SG blowdown flow must be isolated within 15 minutes following the reactor trip. SG blowdown isolation, although a longstanding practice and a plant-specific deviation to the standard Combustion Engineering Owner's Group EOPs, has not been documented on the FCS license docket or in the USAR. This action was institutionalized at FCS during the establishment of EOPs by inclusion in EOP-00, Standard Post-Trip Actions, but was not specifically approved by the Nuclear Regulatory Commission (NRC).

The current LMFW analysis of record was performed with a computer code, CESEC that was not sufficiently refined to model the SG blowdown valves. Using the AREVA codes, OPPD has determined that the acceptance criteria of that analysis cannot be met without isolation of the SG blowdown valves. In the FWLB event, the SG blowdown valves isolate as a result of a containment pressure high signal (CPHS) or a pressurizer pressure low signal (PPLS). In the LMFW event, however, the SG blowdown isolation valves must be closed by operator action. Documentation from the analysis of record does not discuss or address operator action to isolate the SG blowdown. This application addresses that omission by providing clarification of operator action to isolate SG blowdown which ensures that the auxiliary feedwater system will be able to perform its design function to maintain adequate SG water level for decay heat removal once the AFAS is actuated.

4.0 TECHNICAL ANALYSIS

The purpose of the AFAS setpoint is to start at least one of the two available safety-based AFW pumps during a design basis event (DBE). This pump's operation results in the satisfaction of acceptance criteria: 1) to ensure adequate removal of decay heat, and 2) to maintain the primary and secondary systems within overpressure limits.

The NRC previously approved the design and capabilities of the AFW system and AFAS setpoint in Reference 7.1. This approval was based on analyses that were performed and submitted to the NRC in Reference 7.2. Per Reference 7.1, the DBEs to validate the adequacy of the AFAS setpoint are a loss of LMFW and a FWLB. The computer code, CESEC, that was used to perform the analyses at that time was not sufficiently refined to model the SG blowdown valves. It is important to note that the SG blowdown valves at FCS are automatically isolated following a CPHS or a PPLS. Although, the PPLS and/or the CPHS would actuate following a FWLB, neither of these signals occur during a LMFW.

Reliance on operator action to close SG blowdown valves following a LMFW was identified during the performance of the LMFW analysis to support operation of FCS with new steam generators, pressurizer, and reactor vessel head. OPPD has determined that this reliance on operator action also applies with the current components.

OPPD is including the updated Cycle 24 analysis (Reference 7.4, Attachment 4 of this submittal) to the NRC for validating the adequacy of the AFAS setpoint that takes into account the RPZR and RSGs.

Manual isolation of SG blowdown is acceptable because the action is performed from within the control room and occurs soon after a reactor trip associated with LMFW. This action satisfies the criteria of Reference 7.6 assuming a Plant Condition 3, which

allows a Time Margin of 10 minutes and an Operator Action Time Delay of 3 + n x 1 minute, where n signifies the number of discrete manipulations. In this case, n equals one, and the Operator Action Time Delay is 4 minutes, which is less than the 15 minutes assumed by the revised NSSSRP LMFW event. Based on simulator training observations, FCS operators complete this action within 8 minutes following reactor trip which indicates that crediting 15 minutes for this operator action in the LMFW event is conservative. Therefore, OPPD is specifically requesting the NRC to allow the requested clarification to USAR Section 14.10.1 to credit the use of operator action to isolate the SG blowdown valves within 15 minutes following a reactor trip during a LMFW.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Omaha Public Power District has evaluated whether or not a significant hazards consideration is involved with the proposed USAR change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No.

   The proposed change to the USAR clarifies reliance on operator action which has been utilized since implementation of the EOPs. It does not affect an accident initiator previously evaluated in the USAR or Technical Specifications and will not prevent safety systems from performing their accident mitigating function as discussed in the USAR or Technical Specifications.

   Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

   Response: No.

   The proposed change provides clarification to the existing USAR accident analysis of record. The change does not modify or install any safety related equipment. It does not alter any design or licensing basis assumptions and does not alter any operating procedures other than the

explicit specification the time constraint of the 15 minutes. Presently the action is included in EOP-00 without a time constraint.

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

3. **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

The proposed change provides clarification to the USAR section 14.10.1 and has no effect on safety margins.

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

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

5.2 Applicable Regulatory Requirements/Criteria

The proposed change provides clarification to the USAR section 14.10.1 and does not affect commitments to FCS design criteria presented in the USAR Appendix G, accident analysis, approved methodologies, Regulatory Guides, or NUREGs. The technical information associated with this change does not meet the criteria of 10 CFR 50.36 for inclusion in the TS.

Because the USAR does not address automatic isolation of blowdown flow during a LMFW, the proposed change enhances and does not alter, degrade, or prevent actions described or assumed in any accident analysis. The proposed change will not change any assumptions previously made in evaluating radiological consequences or affect any fission product barriers, nor does it increase any challenges to safety systems. Therefore, the proposed change does not increase or have any impact on the consequences of events described in Section 14 of the FCS USAR.

In accordance with the methods presented in Example 4 on page 46 of NEI-96-07 – Revision 1 "Guidelines for 10 CFR 50.59 Implementation" (dated November, 2000), this change presents only a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. The crediting of operator action to maintain heat removal capability involves a

"new" operator action to support a design function credited in safety analyses is acceptable because:

1. The action is reflected in plant procedures, Standard Post Trip Actions, Emergency Operating Procedure, EOP-00, and operator training programs.
2. OPPD has demonstrated that this action conforms to ANSI-58.8 (Reference 7.6) through simulator training observations, where the longest time required by operators to complete this action was 8 minutes following reactor trip of a LMFW event. The action can be completed in the 15 minutes allowed by the analysis considering the aggregate effects, such as workload or environmental conditions, expected to exist when the action is required.
3. Further, time is available for recovery from the most likely credible performance error. As demonstrated by the simulator training observations, even using the longest observed time of 8 minutes, an additional 7 minutes is available to complete this action. This is ample time to recognize and perform the manual action to recover from the delayed initiation of SG blowdown isolation.
4. Crediting this operator action has been evaluated by OPPD. This evaluation determined that because this action is specific to the LMFW event, it has no adverse effects on plant systems.

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

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to 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. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 Safety Evaluation by the office of Nuclear Reactor Regulation supporting Amendment No. 65 to Facility Operating License No. DPR-40, dated June 18, 1982

7.2 Letter from OPPD (W. C. Jones) to NRC (Robert A. Clark), LIC-82-117, dated March 22, 1982

7.3 EOP-00 "Standard Post Trip Actions", (CEOG Emergency Procedure Guidelines, CEN-152)

7.4 AREVA Document 86-5056804-00, "FCS RSG – Auxiliary Feedwater Actuation Set-point (AFAS) Verification (See Attachment 4)

7.5 Safety Evaluation by the office of Nuclear Reactor Regulation supporting Amendment No. 203 to Facility Operating License No. DPR-40, dated March 4, 2002

7.6 ANSI/ANS-58.8-1984, Time response design Criteria for Nuclear Safety Related Operator Actions

Attachment 2

Markup of Affected USAR Pages
Section 14.10.1

During the first few seconds of the event, the secondary temperature and pressure rise. The steam generator water level drops since the turbine is continuing to demand full power in addition to shrinkage caused by the secondary pressure increase. The steam generator water level continues to decrease until a reactor trip on low water level occurs and subsequently initiates a turbine trip. During this time, the primary pressure increase is mitigated by the action of the pressurizer sprays. The primary pressure increase is not sufficient to lift either the pressurizer power operated relief or primary safety valves.

After the reactor trip occurs, the reactor core power rapidly decreases to the decay power levels. The amount of residual heat contained in the fuel and structural materials determines the rate at which the liquid inventory in the steam generators is depleted. The higher the fuel temperatures (i.e., low H_{gap}) and the higher the fission product inventory (i.e., higher power and burnup) the greater the rate of steam generator liquid mass loss. The turbine trip leads to a quick opening of the steam dump and bypass valves, normally in the automatic mode of operation. The reactor coolant system (RCS) and steam generator pressures and temperatures are regulated by this system to remove decay heat, which is extracted by forced coolant flow through the core.

The inventory remaining in the steam generators after trip will not be completely depleted until about 30 minutes (Reference 14.10-3) assuming no operator action other than closing the steam generator blowdown isolation valves within 15 minutes of reactor trip and no additional feedwater. During this time interval, automatic actuation of the safety grade auxiliary feedwater system on low S.G. level (32% wide range level) would occur to assure that a secondary heat sink is maintained. This will allow the cooldown of the plant to proceed in an orderly fashion using the power operated safety valves (MS-291 and MS-292), after which, shutdown cooling can be initiated.

Attachment 3

Proposed Revised USAR Section 14.10.1 (clean)

During the first few seconds of the event, the secondary temperature and pressure rise. The steam generator water level drops since the turbine is continuing to demand full power in addition to shrinkage caused by the secondary pressure increase. The steam generator water level continues to decrease until a reactor trip on low water level occurs and subsequently initiates a turbine trip. During this time, the primary pressure increase is mitigated by the action of the pressurizer sprays. The primary pressure increase is not sufficient to lift either the pressurizer power operated relief or primary safety valves.

After the reactor trip occurs, the reactor core power rapidly decreases to the decay power levels. The amount of residual heat contained in the fuel and structural materials determines the rate at which the liquid inventory in the steam generators is depleted. The higher the fuel temperatures (i.e., low H_{gap}) and the higher the fission product inventory (i.e., higher power and burnup) the greater the rate of steam generator liquid mass loss. The turbine trip leads to a quick opening of the steam dump and bypass valves, normally in the automatic mode of operation. The reactor coolant system (RCS) and steam generator pressures and temperatures are regulated by this system to remove decay heat, which is extracted by forced coolant flow through the core.

The inventory remaining in the steam generators after trip will not be completely depleted until about 30 minutes (Reference 14.10-3) assuming no operator action other than closing the steam generator blowdown isolation valves within 15 minutes of reactor trip, and no additional feedwater. During this time interval, automatic actuation of the safety grade auxiliary feedwater system on low S.G. level (32% wide range level) would occur to assure that a secondary heat sink is maintained. This will allow the cooldown of the plant to proceed in an orderly fashion using the power operated safety valves (MS-291 and MS-292), after which, shutdown cooling can be initiated.

Attachment 4

AREVA Document 86-5056804-00
Fort Calhoun Station Document FC-06967

FCS RSG -Auxiliary Feedwater Actuation Signal (AFAS) Setpoint Verification

AREVA

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 86 - 5056804 - 00

Title FCS RSG - AUXILIARY FEEDWATER ACTUATION SIGNAL (AFAS) SETPOINT VERIFICATION

PREPARED BY:				REVIEWED BY:			
				METHOD: ☒ DETAILED CHECK ☐ INDEPENDENT CALCULATION			
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COST CENTER	41619	REF. PAGE(S)	37	TM STATEMENT: REVIEWER INDEPENDENCE	TBC for Giavedoni		

PURPOSE AND SUMMARY OF RESULTS:

The purpose of this report is to document the results of a Loss of Feedwater Flow event analysis and a Feedwater Line Break event analysis to verify the acceptability of the auxiliary feedwater actuation signal (AFAS) setpoint, along with the acceptability of the Low Steam Generator Water Level trip setpoint, and the minimum auxiliary feedwater (AFW) flow rate.

The results of the analyses indicate that the AFAS setpoint, the Low Steam Generator Water Level trip setpoint, and the minimum AFW flow rate are satisfactory to remove decay heat and reactor coolant pump heat and preclude a significant heatup of the reactor coolant system following reactor trip. Therefore, the AFAS setpoint is verified. In addition, it was determined that the Loss of Feedwater Flow event was the more limiting event regarding verification of the AFAS setpoint since it requires an operator action to terminate steam generator blowdown flow at 15 minutes. The Loss of Feedwater Flow event also results in a higher reactor coolant system temperature following reactor trip.

(FCS Document Number – FC-06967)

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:		THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK
CODE/VERSION/REV	CODE/VERSION/REV	
		☐ YES ☒ NO

Framatome ANP, Inc., *an AREVA and Siemens company*

Contents

1.0 FCS RSG – AUXILIARY FEEDWATER ACTUATION SIGNAL (AFAS) SETPOINT VERIFICATION 4
1.1 Purpose of Report 4
1.2 Summary and Conclusions 4
1.3 Loss of Feedwater Flow Event Description 5
1.4 Feedwater Line Break Event Description 7
1.5 Acceptance Criteria 8
1.6 Loss of Feedwater Flow Event Analysis Results 8
1.7 Feedwater Line Break Event Analysis Results 21

2.0 REFERENCES 37

Tables

Table 1 Initial Operating Conditions and Other Key Parameters for the Loss of Feedwater Flow Event 11
Table 2 Sequence of Events for Loss of Feedwater Flow Event with Isolation of SG Blowdown Flow after 15 Minutes 12
Table 3 Initial Operating Conditions and Other Key Parameters for the FWLB Event 24
Table 4 FWLB Event - Break Spectrum Results 25
Table 5 Sequence of Events for the FWLB event – 80% Break Case 26

Figures

Figure 1 RCS Coolant Temperatures Versus Time for Loss of Feedwater Flow Event 13
Figure 2 Pressurizer Pressure Versus Time for Loss of Feedwater Flow Event 14
Figure 3 Steam Generator Narrow Range Level Versus Time for Loss of Feedwater Flow Event 15
Figure 4 Steam Generator Pressure Versus Time for Loss of Feedwater Flow Event 16
Figure 5 Pressurizer Level Versus Time for Loss of Feedwater Flow Event 17

Figure 6	Steam Generator Wide Range Level Versus Time for Loss of Feedwater Flow Event	18
Figure 7	Auxiliary Feedwater and Steam Generator Blowdown Flow Rates Versus Time for Loss of Feedwater Flow Event	19
Figure 8	Steam Generator Liquid Mass Inventory Versus Time for Loss of Feedwater Flow Event	20
Figure 9	Steam Generator Wide Range Level Versus Time for FWLB Event	27
Figure 10	MSIV Flow Rate Versus Time for FWLB Event	28
Figure 11	Steam Generator Narrow Range Level Versus Time for FWLB Event	29
Figure 12	Auxiliary Feedwater and Steam Generator Blowdown Flow Rates Versus Time for FWLB Event	30
Figure 13	Pressurizer Level Versus Time for FWLB Event	31
Figure 14	Pressurizer Pressure Versus Time for FWLB Event	32
Figure 15	Steam Generator Pressure Versus Time for FWLB Event	33
Figure 16	Steam Generator Liquid Mass Inventory Versus Time for FWLB Event	34
Figure 17	RCS Coolant Temperatures Versus Time for FWLB Event	35
Figure 18	Total MSSV Flow Rate Versus Time for FWLB Event	36

1.0 FCS RSG – AUXILIARY FEEDWATER ACTUATION SIGNAL (AFAS) SETPOINT VERIFICATION

1.1 *Purpose of Report*

The purpose of this report is to compare the results of the Loss of Feedwater Flow event analysis and the Feedwater Line Break (FWLB) event analysis to determine which event is the most limiting or most challenging for the auxiliary feedwater system. Both events challenge the auxiliary feedwater system in that both events lose main feedwater and steam generator inventory which challenges the ability to remove decay heat and reactor coolant pump (RCP) heat. In addition, the acceptability of the Low Steam Generator Water Level trip setpoint, the Auxiliary Feedwater Actuation Signal (AFAS) setpoint, and the minimum auxiliary feedwater (AFW) flow rate will be evaluated based in both event analyses.

The acceptance criteria for the Loss of Feedwater Flow event includes demonstrating that the Specified Acceptable Fuel Design Limits (SAFDLs) are satisfied. This is demonstrated by assuring that the steam generators provide a sufficient heat sink for decay heat and RCP heat, as evidenced by a limited heatup of the reactor coolant system (RCS). For the purposes of this comparison, the event that is most challenging relative to the acceptance criteria for the Loss of Feedwater Flow event will be considered the most challenging event for the auxiliary feedwater system.

Both the Loss of Feedwater Flow event analysis (Reference 1) and the Feedwater Line Break event analysis (Reference 2) were performed to support plant operation with the replacement steam generators (RSGs) and the replacement pressurizer. Both events were analyzed to demonstrate the adequacy of the auxiliary feedwater system including the Low Steam Generator Water Level trip setpoint, the AFAS setpoint, and the minimum AFW flow rate. The replacement pressurizer was modeled in the transient analyses utilizing data from Reference 3.

1.2 *Summary and Conclusions*

The results of the Loss of Feedwater Flow event analysis (Reference 1) demonstrate that the acceptance criteria for the Loss of Feedwater Flow event stated in Section 1.5 are satisfied, provided that steam generator blowdown flow is isolated within 15 minutes after reactor scram. The results of the Feedwater Line Break event analysis (Reference 2) also demonstrate that the acceptance criteria stated in Section 1.5 are satisfied.

The Loss of Feedwater Flow event analysis shows that both steam generators are dry by about 1100 sec, even with isolation of steam generator blowdown flow 15 minutes after reactor scram. The RCS coolant temperatures increase until decay heat eventually decreases to the point where the auxiliary feedwater flow is adequate to remove decay heat and RCP heat. The maximum post-scram RCS hot leg temperature for the Loss of Feedwater Flow event is 567°F.

The results of the Feedwater Line Break event analysis show that both steam generators dry out. The unaffected steam generator goes dry by about 1516 sec. The RCS coolant temperatures increase until decay heat eventually decreases to the point where the auxiliary feedwater flow is adequate to remove decay heat and RCP heat. The maximum post-scram RCS hot leg temperature for the FWLB event is 557°F.

Both the Loss of Feedwater Flow event (with a 15 minute steam generator blowdown flow isolation time) and Feedwater Line Break event analyses demonstrate that the AFWS is adequate to remove decay heat and RCP heat and that the current Low Steam Generator Water Level trip setpoint, AFAS setpoint, and minimum AFW flow rate are satisfactory. However, the Loss of Feedwater Flow event is the most challenging event since without manual isolation of steam generator blowdown flow 15 minutes after reactor scram, criteria would not be satisfied for that event. Even with isolation of steam generator blowdown flow 15 minutes after reactor scram, the Loss of Feedwater Flow event is still the most limiting event as far as RCS heatup following reactor scram. Therefore, the Loss of Feedwater Flow event is the most challenging event to evaluate the adequacy of the AFWS including the Low Steam Generator Water Level trip setpoint, the AFAS setpoint, and the minimum AFW flow rate.

The results of the transient analyses include the effects of both the RSGs and the replacement pressurizer.

1.3 *Loss of Feedwater Flow Event Description*

The Loss of Feedwater Flow event is defined as a reduction in main feedwater (MFW) flow to the steam generators when operating at power without a corresponding reduction in steam flow from the steam generators. The limiting case is a total loss of MFW, which most likely would result from:

a. Inadvertent closure of the MFW control or regulating valves or feedwater isolation valves due to a feedwater controller malfunction or manual positioning by the operator, or

b. Loss of all feedwater or condensate pumps.

Upon the loss of MFW flow to the steam generators and a continued steam demand by the turbine, water inventories in the steam generators begin decreasing as well as the heat removal rate from the RCS. This in turn causes the RCS temperatures to increase. The RCS coolant expands into the pressurizer. The resulting increase in pressure actuates the pressurizer spray system and may cause the pressurizer power operated relief valves (PORVs) to open. The pressurizer sprays and PORVs are assumed to function in this event to exacerbate the challenge to pressurizer overfill.

Steam generator liquid levels, which have been steadily dropping since the termination of MFW flow, soon reach the Low Steam Generator Water Level reactor protective system (RPS) trip setpoint. This initiates a reactor scram, which ends the short-term-heatup phase of the event.

Turbine trip at reactor scram and the continuing primary-to-secondary transfer of decay heat and RCP heat cause steam generator pressures to rapidly increase. When steam generator pressures and coolant temperatures have increased to the appropriate values, the steam dump and bypass system and/or the main steam safety valves (MSSVs) mitigate the increase in steam generator pressures.

Steam generator levels continue to drop and reach the AFAS setpoint. This initiates the starting sequence for the AFWS pumps. When the delivery of AFW begins, the rate of level decrease in the steam generators slows.

Eventually, a long-term-heatup phase of the event may begin if primary-to-secondary heat transfer degrades as a result of steam generator tube uncovery. As the decay heat level drops, liquid levels in the steam generators stabilize and then begin to rise. Also, RCS temperatures stabilize and then begin to decrease. At this point, the event is considered to be over.

The Fort Calhoun Station (FCS) has two safety grade AFW pumps. One is an electric motor driven pump and the other is a steam turbine driven pump. Typically, the pump with the highest flow rate is assumed inoperable due to a single failure assumption. Since both the motor driven pump and the steam turbine driven pump have the same minimum flow rate of 180 gpm, the

single failure is arbitrarily assumed to be the steam turbine driven pump. An alternate single failure of an AFW isolation valve, isolating one steam generator, would be less limiting than the loss of one safety grade AFW pump since the total AFW flow would be greater for the case of a failed closed AFW isolation valve.

1.4 *Feedwater Line Break Event Description*

The FWLB event is defined as a major break in a main feedwater line that is sufficiently large to prevent maintaining the steam generator (SG) secondary side water inventory in the affected SG. A spectrum of break sizes in the main feedwater line between the check valve and the SG are analyzed. The largest feasible break area analyzed is either the full crossectional area of the main feedwater line, the feedring, or combined feedring nozzles, whichever is smallest. The choke plane will occur at the smallest of these areas.

The event has three distinct phases. The first phase results in a mild RCS heatup due to the loss of feedwater to the steam generators. This heatup portion of the transient produces the so-called "first peak" RCS pressure response, which does not result in a challenge to the RCS pressure limit. A reactor trip will occur on Low Steam Generator Water Level in the affected SG. Following reactor trip, the second phase involves a cooldown of the RCS due to energy removal during the SG blowdown stage. The cooldown phase is bounded by the Main Steam Line Break (MSLB) event since the MSLB analysis assumes all steam flow out the break and the break area is larger for a MSLB event (1.1 ft^2 for the MSLB event versus 0.9 ft^2 for the FWLB event). The RCS pressure during the second phase of the event may decrease enough to cause the Safety Injection (SI) system to activate. Following the cooldown portion of the event, the third phase involves the eventual depletion of secondary-side inventory in the affected SG. In addition, lack of main feedwater to the unaffected SG results in a long term RCS heatup much like a Loss of Feedwater Flow event. AFW flow to the unaffected SG is actuated on the AFAS for the unaffected SG. The expansion of the RCS coolant and the potential SI flow will re-fill the pressurizer and re-pressurize the RCS. The RCS pressure during the third phase of the event results in the so-called "second peak", which is limited by the opening of the Primary Safety Valves (PSVs), if necessary. AFW will eventually restore the inventory in the unaffected SG and the decay heat and RCP heat will be removed via steam flow through the MSSVs. As the decay heat level drops, the liquid level in the unaffected SG stabilizes and then begins to rise. Also, RCS temperatures stabilize and then begin to decrease. When the unaffected SG level

begins to increase and the RCS temperatures begin to decrease, the FWLB transient is considered to be over.

The FCS has two safety grade AFW pumps. One is an electric motor driven pump and the other is a steam turbine driven pump. Typically, the pump with the highest flow rate is assumed inoperable due to a single failure assumption. Since both the motor driven pump and the steam turbine driven pump have the same minimum flow rate of 180 gpm, the single failure is arbitrarily assumed to be the steam turbine driven pump. An alternate single failure could be a failed closed AFW isolation valve. However, in this case AFW flow would be directed to the intact SG through the main feedwater line. Thus, the minimum AFW flow rate would be the same as for the single failure of an inoperable AFW pump.

1.5 *Acceptance Criteria*

The acceptance criteria for the Loss of Feedwater Flow event are:

1. The pressure in the RCS and main steam system must be less than 110% of design values. The design pressure of the RSGs is 1010.8 psig and the design pressure of the remainder of the steam system is 985.8 psig. Therefore, the peak pressure in the RSGs must be shown to be less than 1111.9 psig and the peak pressure in the remainder of the steam system must be shown to be less than 1084.4 psig. The peak pressure in the RCS must be less than 2734.4 psig.

2. Fuel cladding integrity must be maintained by ensuring that the departure-from-nucleate-boiling (DNB) and fuel centerline melt SAFDLs are not exceeded. This is further demonstrated by assuring that the steam generators provide a sufficient heat sink for decay and reactor coolant pump heat, as evidenced by the maintenance of the EOP-06 Core Heat Removal Safety Function (i.e., $T_H < 600°F$), consequently ensuring that the RCS Inventory Control Safety Function (including RCS subcooling $\geq 20°F$) is met.

3. The event must not generate a more serious plant condition without other faults occurring independently. This is further demonstrated by assuring that the pressurizer does not overfill such that liquid is expelled through the PORVs and/or PSVs.

For the purposes of this evaluation, the acceptance criteria for the FWLB event are assumed to be the same as for the Loss of Feedwater Flow event.

1.6 *Loss of Feedwater Flow Event Analysis Results*

Two cases were analyzed in Reference 1. Case 1 assumed an instantaneous loss of MFW with isolation of SG blowdown flow 15 minutes after reactor scram. Case 1 assumed that the steam dump and bypass system was not available. Case 2 also assumed an instantaneous loss of

MFW with isolation of SG blowdown flow 15 minutes after reactor scram. However, case 2 assumed that the steam dump and bypass system was available. Between Cases 1 and 2, Case 1 was determined to be the most limiting case regarding minimum SG liquid inventory and maximum post-scram RCS hot leg temperature. Therefore, only the results for Case 1 are presented herein.

The initial plant operating conditions and other key parameters for the Loss of Feedwater Flow event analysis are given in Table 1. The sequence of events for the limiting case is given in Table 2.

The results of the Loss of Feedwater Flow event analysis are depicted in Figures 1 through 8 for the limiting case.

The event is initiated by an instantaneous loss of all MFW. The water level in the steam generators begins to decrease as steam flow continues through the tubine control valves. The heat transfer between the primary and secondary systems begins to decrease and RCS temperatures begin to increase as shown in Figure 1, causing an insurge of reactor coolant into the pressurizer. A moderate increase in RCS pressure occurs prior to reactor scram as shown in Figure 2, but not significant enough to open the PORVs.

The SG water level continues to decrease (see Figure 3) until a reactor trip signal occurs on a narrow range Low SG water level at 25.05 sec (with delay). Reactor trip is followed by a turbine trip. A large increase in SG pressures occurs following turbine trip as shown in Figure 4. The first MSSVs open at 28 sec, mitigating the increase in SG pressures. The peak RCS pressurizer level of 77.6% span occurs at 28 sec as shown in Figure 5.

The SG water level continues to decrease as shown in Figure 6 until the AFAS setpoint is reached at 525.3 sec, actuating AFW flow (after a 50.9 sec delay time). The AFW and SG blowdown flow rates are shown in Figure 7. The SG blowdown flow is isolated 15 minutes after reactor scram. As shown in Figure 8, the steam generators effectively dry out by about 1100 sec. Once the steam generators dry out, the RCS temperatures and pressure begin to increase. As shown in Figure 2, the second RCS pressure peak after reactor scram is less than the first pressure peak prior to reactor scram. The peak post-scram RCS hot leg temperature was calculated to be 567°F at 3054 sec, at which time the AFW flow begins to provide sufficient

heat removal to remove decay heat and RCP heat. At this point the RCS temperatures and pressure begin to decrease.

A comparison of analysis results to criteria is as follows.

Acceptance Criterion 1:

The pressure in the RCS and main steam system must be less than 110% of design values.

The challenge to peak primary and secondary overpressure is less for the Loss of Feedwater Flow event (with reactor scram and secondary-system isolation nearly coincident) than for the Loss of Load to Both Steam Generators event (with secondary-system isolation at event initiation and continued operation at power for a considerable period of time). Thus, the primary pressure limit is satisfied as long as the pressurizer does not become liquid-filled and the pressurizer retains a steam "bubble" for pressure control. In this analysis, the pressurizer did not become liquid-filled. Therefore, the primary overpressure criterion is satisfied.

In addition, the peak secondary system pressure is bounded by the peak secondary system pressure in the Loss of Load to Both Steam Generators event since the primary to secondary heat transfer rate will be somewhat higher for the Loss of Load to Both Steam Generators event due to continued operation at power after the turbine valves close. Thus, the secondary system overpressure limit is satisfied for the Loss of Feedwater Flow event since this limit has been demonstrated to be satisfied in the Loss of Load to Both Steam Generators event analysis.

Acceptance Criterion 2:

Maintenance of fuel cladding integrity is demonstrated by assuring that the post-scram hot leg temperature does not exceed 600°F and the subcooling margin is greater than 20°F. The maximum post-scram hot leg temperature was calculated to be 567°F and the subcooling margin is significantly greater than 20°F as shown in Figure 1. Therefore, acceptance criterion 2 is satisfied.

Acceptance Criterion 3:

The maximum pressurizer level was calculated to be 77.6 % span. Therefore, the pressurizer does not fill and thus liquid is not expelled through the PORVs or PSVs.

Table 1 Initial Operating Conditions and Other Key Parameters for the Loss of Feedwater Flow Event

Parameter	Analysis Value	Comments
Initial Reactor Power (Including Measurement Uncertainty) (MWt)	1530	RTP is 1524 MWt with a 6 MWt measurement uncertainty.
Initial RCS Flow Rate (Including Measurement Uncertainty) (gpm)	195,210	The Technical Specification minimum RCS flow rate is 202,500 gpm. The measurement uncertainty is 3.6%.
Initial Core Inlet Temperature (°F)	543	
Initial Pressurizer Pressure (psia)	2100	
Initial Pressurizer Level (% span)	76.12	
Initial SG Pressure at 0% SGTP (psia)	836	
RPS trip on Low SG Level (% NR)	25.5	
AFAS Setpoint (% WR)	15	
Minimum AFW Flow Rate (gpm)	180	
AFW Actuation Delay Time (sec)	50.9	
SG Blowdown Flow Rate per SG (lbm/hr)	62,240	
SG Blowdown Flow Isolation Time (sec)	See Comment	Isolated 15 minutes after reactor trip by operator action.
MSSV Setpoints (psia)	1029.6 *1045.0* 1055.3 1070.8 1081.1	Note that the Technical Specifications allow one MSSV in each SG to be inoperable at full power. This is assumed to be a large MSSV with the lowest setpoint, 1045.0 psia.

Table 2 Sequence of Events for Loss of Feedwater Flow Event with Isolation of SG Blowdown Flow after 15 Minutes

Time (sec)	Event	Value
0.0	Event Initiation – Termination of main feedwater flow	---
24.15	SG water level reaches RPS trip setpoint	25.5% NR
25.05	RPS trip signal on SG low water level	0.9 sec delay
25.35	Turbine trip	0.3 sec delay on RPS trip signal
25.55	Scram rod insertion begins	0.5 sec delay on RPS trip signal
28	Peak pressurizer level	77.6%
~28	MSSVs begin to open	1029.56 psia
525.26	AFAS setpoint reached	15% WR
576.16	AFW delivery begins	50.9 sec delay
925.54	SG blowdown flow isolation	900 sec after scram
1112/1112	Minimum SG liquid inventory	158/169 lbm
1538 2720	Pressurizer spray on, sustained Pressurizer spray off	$P-P_{ref}>75$ psi $P-P_{ref}\leq75$ psi
3054	Peak post-scram RCS hot leg temperature	567°F

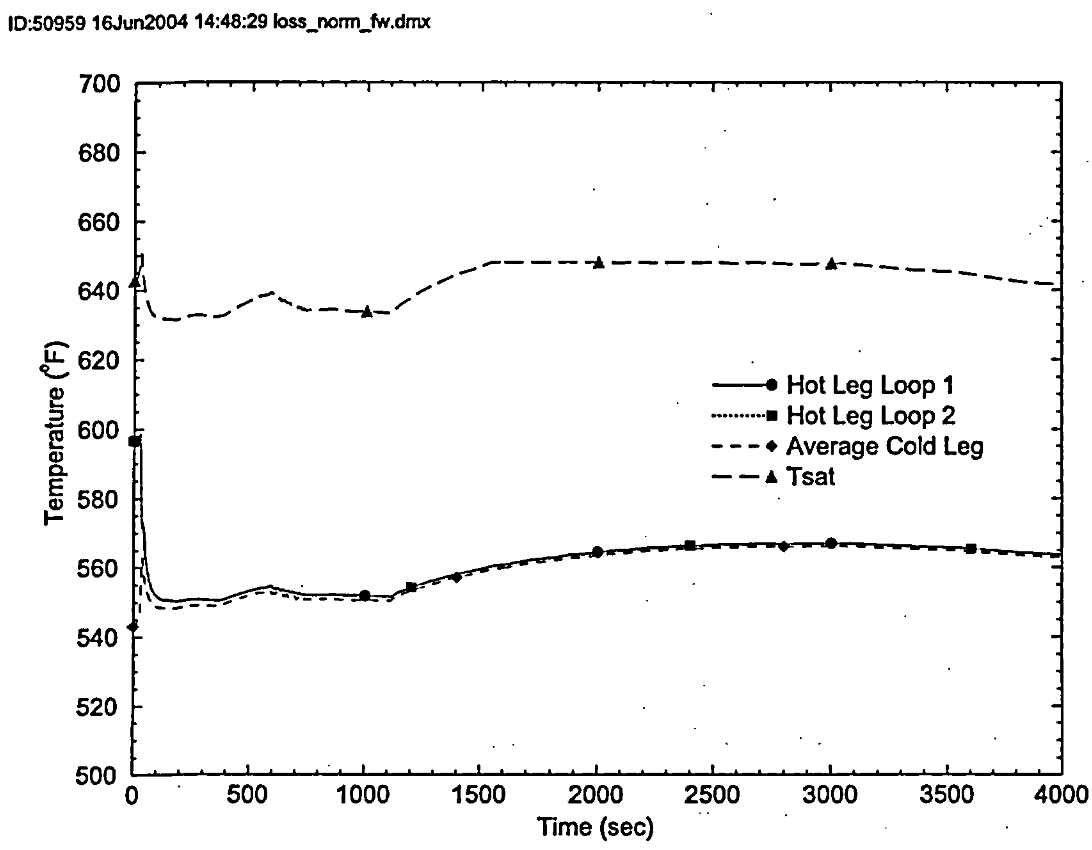

Figure 1 RCS Coolant Temperatures Versus Time for Loss of Feedwater Flow Event

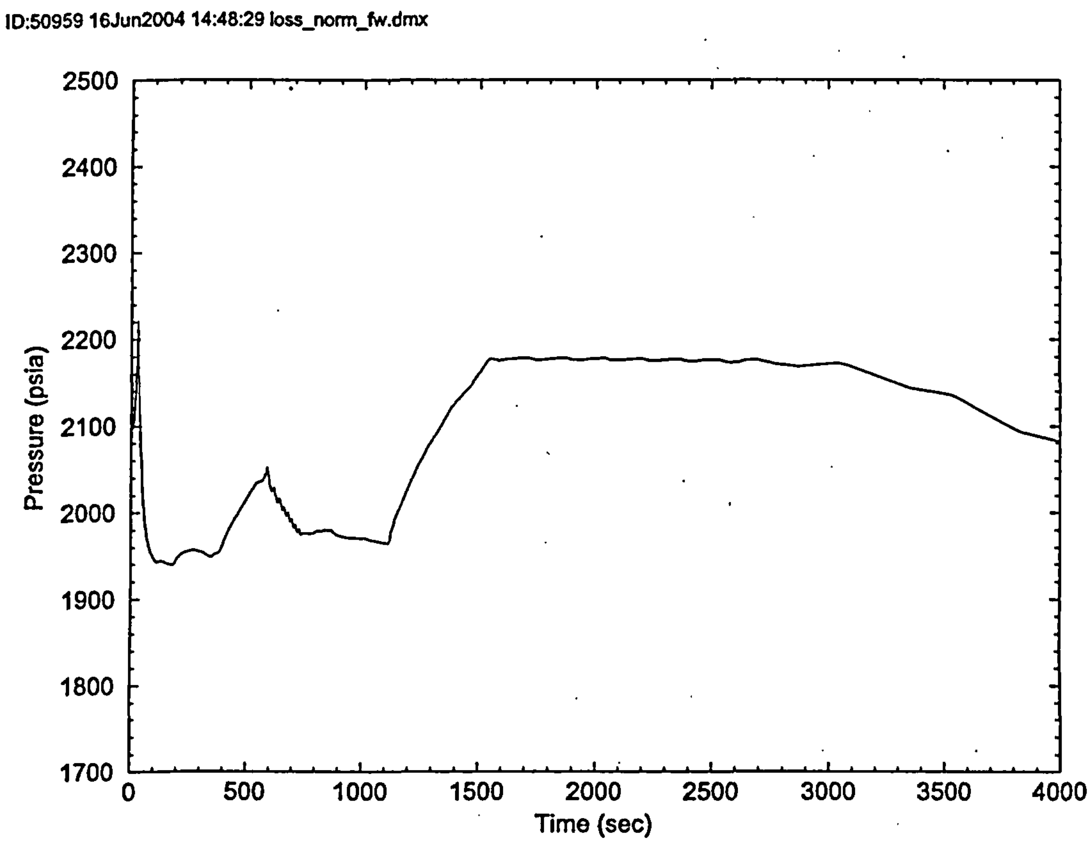

Figure 2 Pressurizer Pressure Versus Time for Loss of Feedwater Flow Event

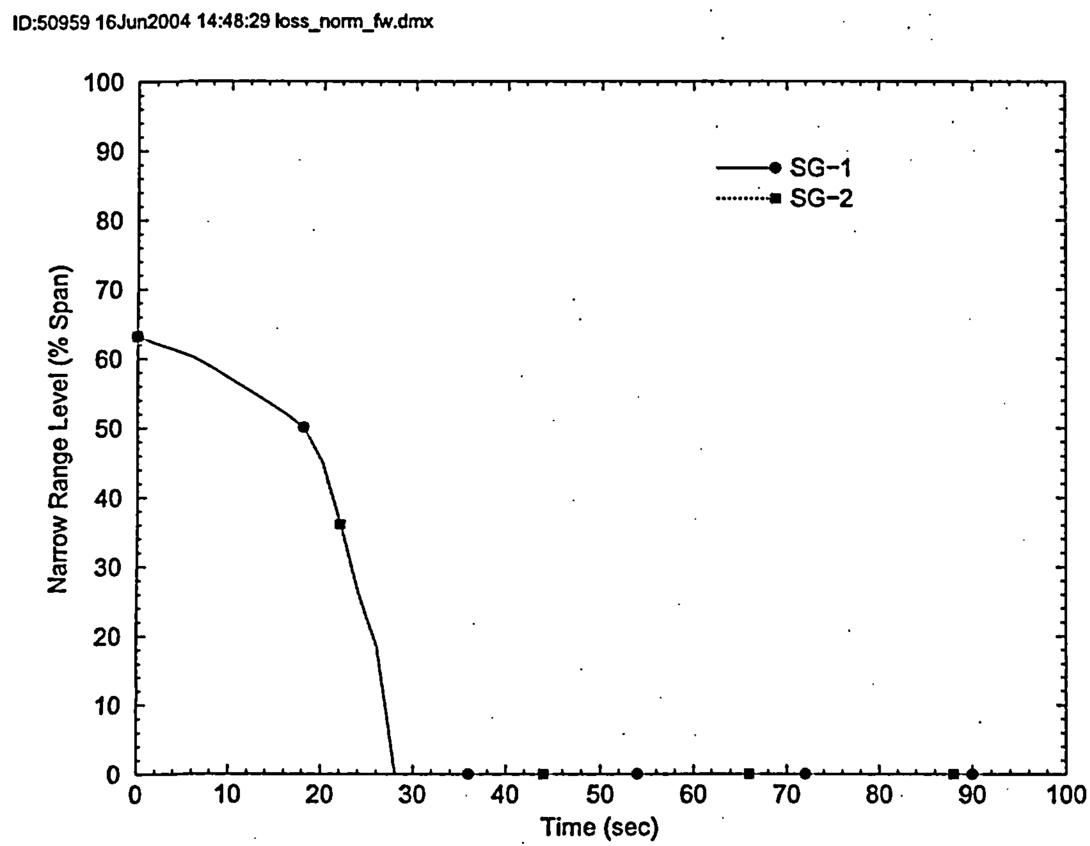

Figure 3 Steam Generator Narrow Range Level Versus Time for Loss of Feedwater Flow Event

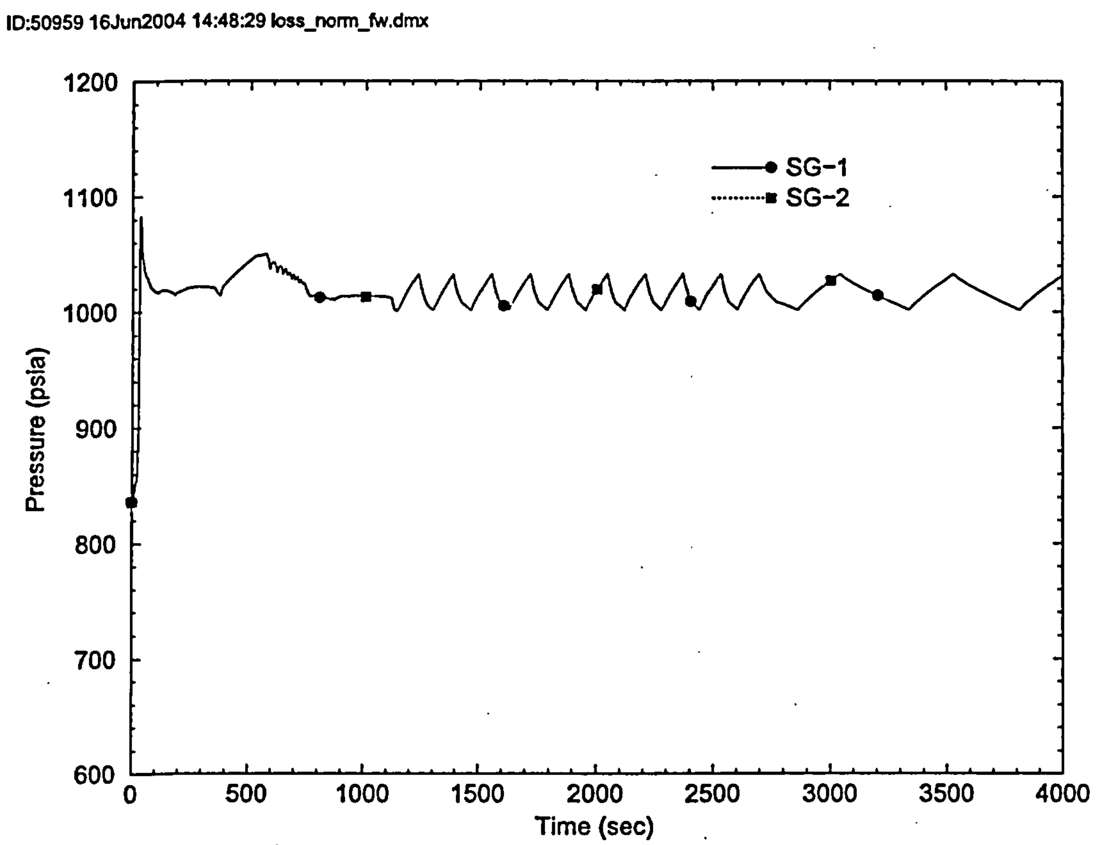

Figure 4 Steam Generator Pressure Versus Time for Loss of Feedwater Flow Event

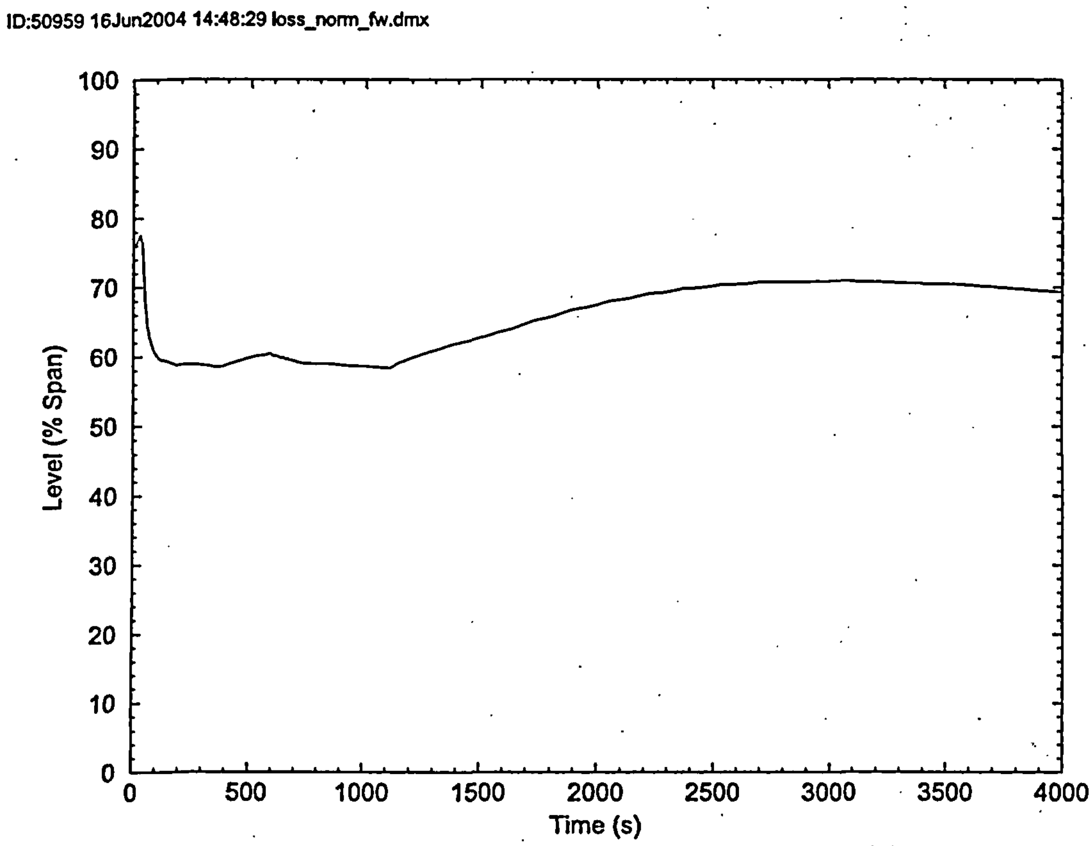

Figure 5 Pressurizer Level Versus Time for Loss of Feedwater Flow Event

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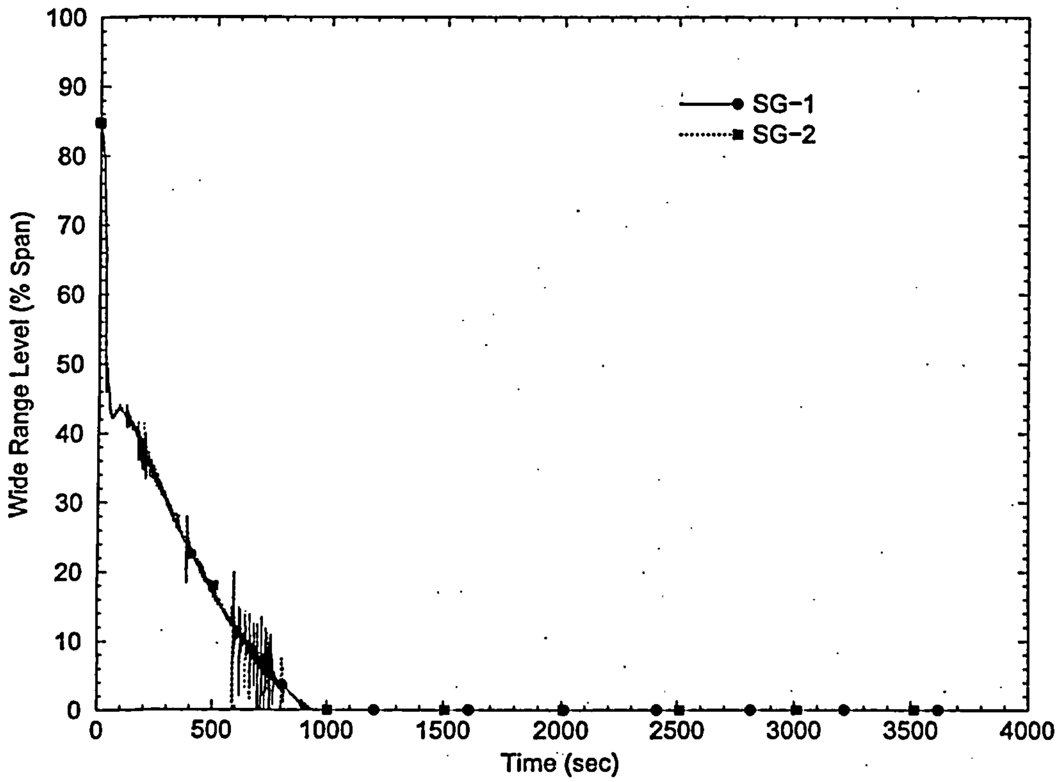

Figure 6 Steam Generator Wide Range Level Versus Time for Loss of Feedwater Flow Event

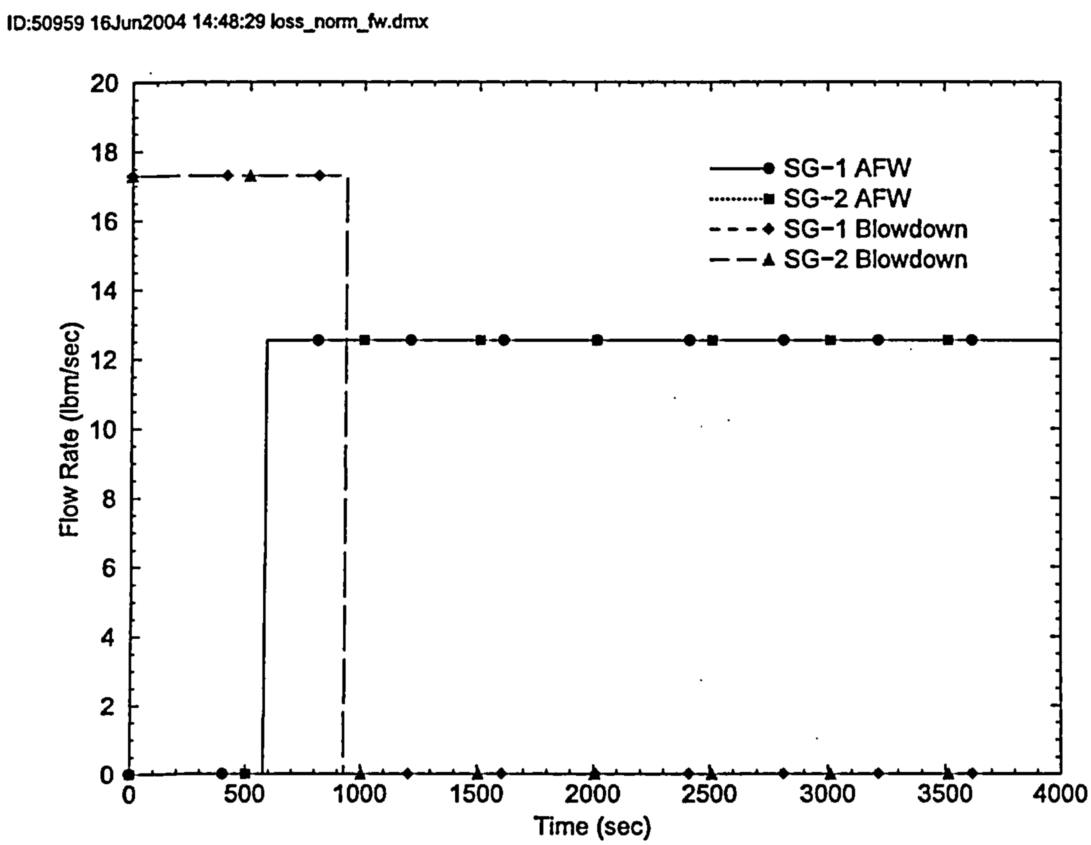

Figure 7 Auxiliary Feedwater and Steam Generator Blowdown Flow Rates Versus Time for Loss of Feedwater Flow Event

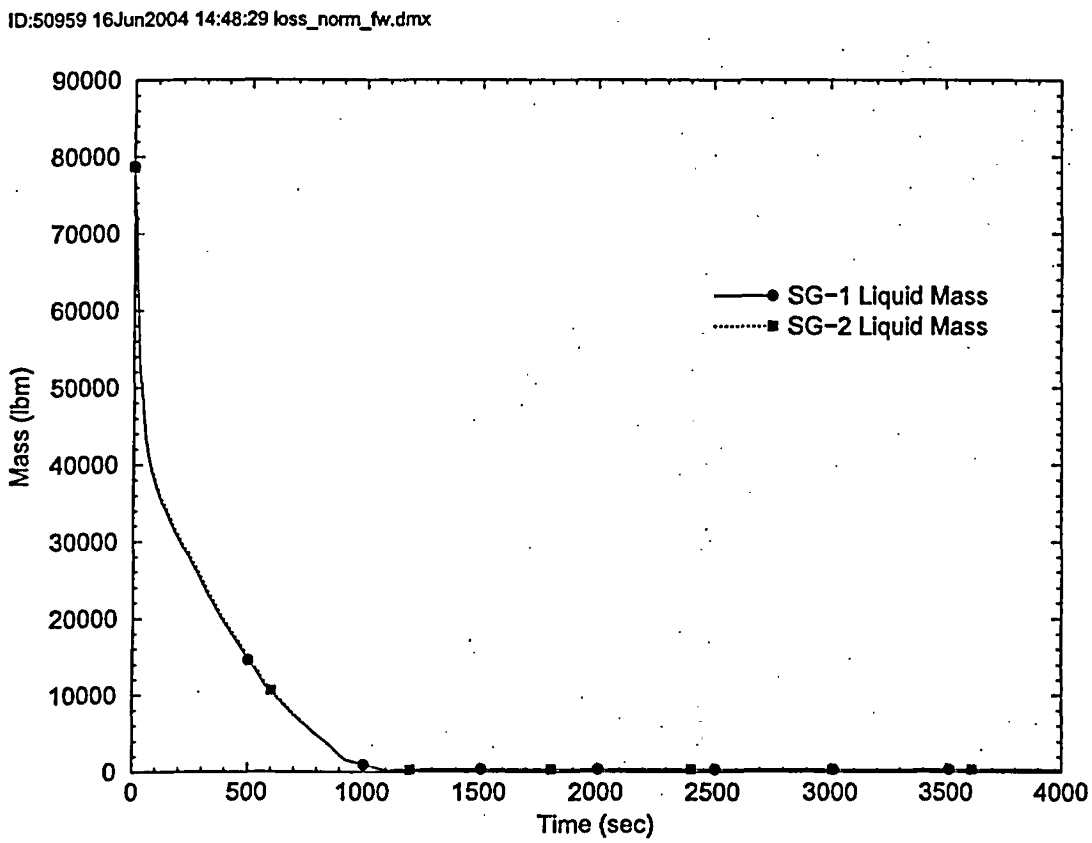

Figure 8 Steam Generator Liquid Mass Inventory Versus Time for Loss of Feedwater Flow Event

1.7 *Feedwater Line Break Event Analysis Results*

A series of break sizes were analyzed in Reference 2 for the FWLB event. Break sizes of 60%, 70%, 80%, 90%, and 100% of the largest feasible break area were analyzed. The largest feasible break area analyzed was 0.9 ft^2, which occurs at the inlet to the feedring.

The initial plant operating conditions and other key parameters for the FWLB event analysis are given in Table 3.

A summary of results for the break spectrum study is given in Table 4. It can be seen that the results for the various break sizes are not significantly different regarding peak RCS pressure, peak SG pressure, minimum SG mass, and peak post-scram hot leg temperature. Since both steam generators dry out for both the Loss of Feedwater Flow event and the FWLB event, the key parameter for comparison between the two events will be the peak post-scram hot leg temperature. The limiting break size regarding peak post-scram hot leg temperature was the 0.72 ft^2 (80%) break size. The peak post-scram hot leg temperature for this case was 557 °F.

Following is a description of the results for the 80% break area case. The sequence of events for the 80% break area case are given in Table 5 and the transient response is depicted in Figures 9 through 18.

The event is initiated by a break in the main feedwater line between the check valve and the SG. All MFW is assumed to be lost immediately at event initiation. Both SGs begin to lose inventory immediately as shown in Figures 9 and 16, but more quickly from the affected SG (SG-1) due to the break. Inventory is lost from the unaffected SG (SG-2) via flow through the main steam isolation valves (MSIVs) to the break as shown in Figure 10. Reactor trip is conservatively assumed to occur at a SG level of 0.0% NR (see Figure 11) in the affected SG. Reactor trip occurs at 6.36 sec, including signal delay time. Turbine trip occurs at 6.65 sec and the steam dump and bypass valves open at the time of turbine trip. Control Element Assembly (CEA) insertion begins at 6.85 sec, including a 0.5 sec delay time. The levels in the SGs continue to decrease and the AFAS setpoint of 15% WR is reached at 634.1 sec in the unaffected SG (see Figure 9). The AFW system logic at FCS allows AFW flow only to the unaffected SG for this event. AFW flow to the unaffected SG begins at 685 sec, including a 50.9 sec delay time after the AFAS setpoint is reached as shown in Figure 12. The MSIVs close on a Steam Generator Low Pressure Signal at 49.13 sec, precluding further loss of

inventory out the break from the unaffected SG as shown in Figure 10. SG blowdown flow is automatically and conservatively isolated at 49.03 sec as shown in Figure 12.

The pressurizer level begins to decrease following reactor scram as the RCS coolant temperatures decrease as shown in Figure 13. RCS overpressure is not challenged as shown in Figure 14.

Steam generator pressures are shown in Figure 15. Once the MSIVs close, the affected SG quickly depressurizes to atmospheric pressure while the unaffected SG pressure increases and eventually reaches the MSSV setpoints.

Steam generator liquid inventory is shown in Figure 16. The affected SG (SG-1) goes dry and decay heat and RCP heat are removed via the MSSVs on the unaffected SG. The unaffected SG (SG-2) goes dry at about 1516 sec.

The RCS coolant temperatures are shown in Figure 17. RCS temperatures decrease following reactor scram due to steam release from the affected SG until the affected SG goes dry. At that time, the RCS coolant temperatures begin to increase until about 350 sec when the MSSVs on the unaffected SG have opened (see Figure 18) to remove decay heat and RCP heat. At that time, the RCS coolant temperatures remain fairly constant until the unaffected SG goes dry at about 1516 sec. At that time, the RCS coolant temperatures begin to increase. The peak post-scram hot leg temperature was calculated to be 557°F at 2832 sec. After 2832 sec, the decay heat has decreased to the point where the AFW begins to refill the unaffected SG and the RCS temperatures begin to decrease. The peak post-scram hot leg temperature for the FWLB event is less than the peak post-scram hot leg temperature for the Loss of Feedwater Flow event.

A comparison of FWLB analysis results to the criteria for the Loss of Feedwater Flow event is as follows.

<u>Acceptance Criterion 1:</u>

The pressure in the RCS and main steam system must be less than 110% of design values.

The peak primary pressure was calculated to be 2291.3 psia (for the 80% break case), which occurred at 2442 sec. Therefore, the primary overpressure criterion is not challenged for the FWLB event.

The peak SG pressure was calculated to be 990.6 psia (for the 100% break case), which occurred at 196 sec. Therefore, the secondary overpressure criterion is not challenged for the FWLB event.

Acceptance Criterion 2:

Maintenance of fuel cladding integrity is demonstrated by assuring that the post-scram hot leg temperature does not exceed 600°F and the subcooling margin is greater than 20°F. The maximum post-scram hot leg temperature was calculated to be 557°F (for the 80% break case) and the subcooling margin is significantly greater than 20°F as shown in Figure 17. Therefore, acceptance criterion 2 is satisfied.

Acceptance Criterion 3:

The pressurizer level decreased from the initial value. Therefore, the pressurizer does not fill and thus liquid is not expelled through the PORVs or PSVs.

Table 3 Initial Operating Conditions and Other Key Parameters for the FWLB Event

Parameter	Analysis Value	Comments
Initial Reactor Power (Including Measurement Uncertainty) (MWt)	1530	RTP is 1524 MWt with a 6 MWt measurement uncertainty.
Initial RCS Flow Rate (Including Measurement Uncertainty) (gpm)	195,210	The Technical Specification minimum RCS flow rate is 202,500 gpm. The measurement uncertainty is 3.6%.
Initial Core Inlet Temperature (°F)	543	
Initial Pressurizer Pressure (psia)	2100	
Initial Pressurizer Level (% span)	76.12	
SG Pressure at 0% SGTP (psia)	836	
RPS trip on Low SG Level (% NR)	0.0	Conservative value relative to a setpoint of 25.5%.
AFAS Setpoint (% WR)	15	
Minimum AFW Flow Rate (gpm)	180	
AFW Actuation Delay Time (sec)	50.9	
SG Blowdown Flow Rate per SG (lbm/hr)	62,240	
Steam Generator Low Pressure Signal (psia)	478	
MSSV Opening Setpoints (psia)	970.4 985.0 994.7 1009.2 1018.9	These are nominal values minus a 3% uncertainty.

Table 4 FWLB Event - Break Spectrum Results

Cases	Minimum Intact SG Liquid Mass (lbm)	Peak SG pressure (psia)	Peak RCS Pressure (psia)	Peak Post-Scram Hot Leg Temperature (°F)
	Mass/Time	Pressure/Time	Pressure/Time	Temp./Time
100% Break Area (0.90 ft^2)	256/5048 s	990.6/196 s	2263.4/2702 s	556.6/2892 s
90% Break Area (0.81 ft^2)	269/2004 s	990.6/246 s	2237.3/2662 s	554.8/2884 s
80% Break Area (0.72 ft^2)	510/1516 s	989.3/494 s	2291.3/2442 s	556.8/2832 s
70% Break Area (0.63 ft^2)	263/1970 s	988.4/410 s	2258.4 /412 s	555.0/2788 s
60% Break Area (0.54 ft^2)	497/1848 s	988.0/492 s	2282.8 /500 s	554.3/2876 s

Table 5 Sequence of Events for the FWLB event – 80% Break Case

Time (sec)	Event	Value
0.0	Event Initiation – MFW line break and termination of main feedwater flow	---
5.48	Affected SG (SG-1) low water level reaches RPS trip setpoint	0.0% NR
6.36	RPS trip signal on SG low water level	0.9 sec delay
6.65	Turbine trip	0.3 sec delay on RPS trip signal
6.65	Steam dump and bypass valves open	$T_{avg} \geq (532°F+8°F)$ and turbine trip
6.85	Scram rod insertion begins	0.5 sec delay on RPS trip signal
48.14	Steam Generator Low Pressure Signal reached	< 478 psia
49.03	SG blowdown flow isolation	---
49.13	MSIVs close	---
308	First unaffected SG MSSV opens	---
494	Maximum SG pressure (unaffected SG)	989.3 psia
634.1	AFAS on unaffected SG (SG-2)	15% WR
685	AFW flow begins to unaffected SG	15% WR + 50.9 sec delay
1516	Minimum unaffected SG liquid mass	510 lbm
2442	Maximum RCS pressure	2291.3 psia
2832	Peak post-scram RCS hot leg temperature	556.8°F

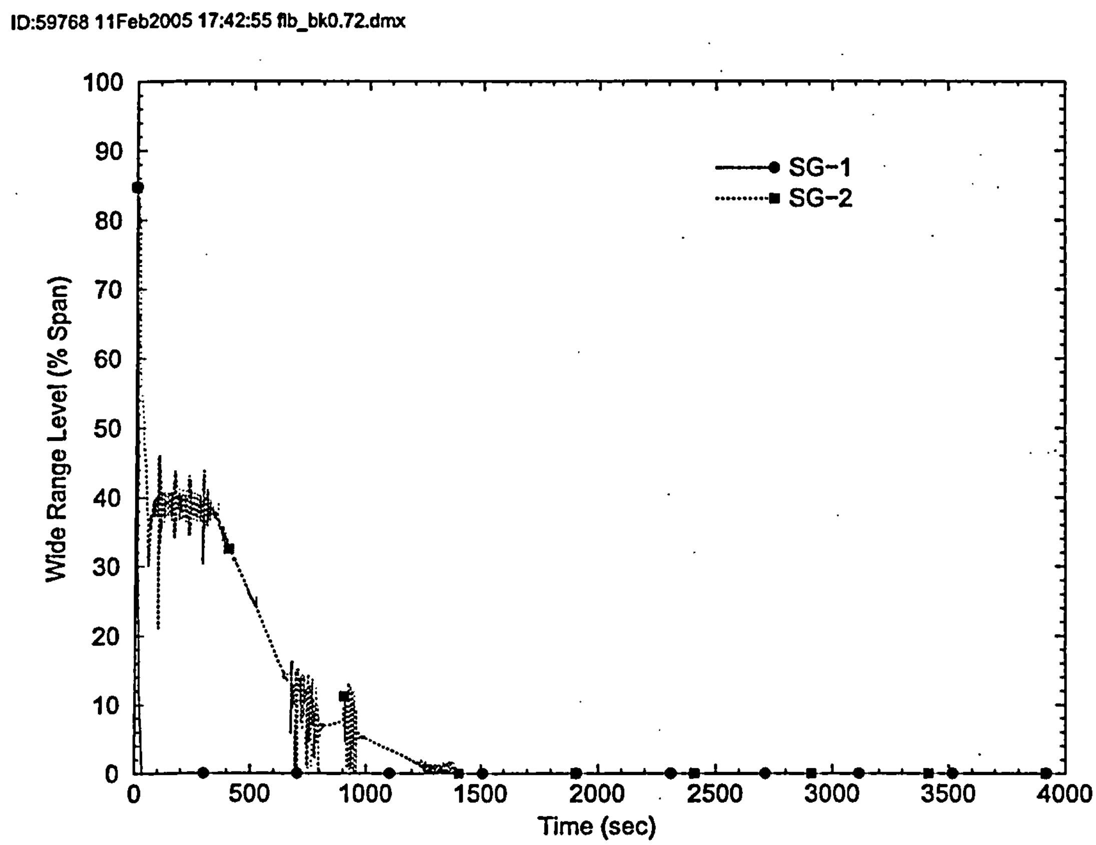

Figure 9 Steam Generator Wide Range Level Versus Time for FWLB Event

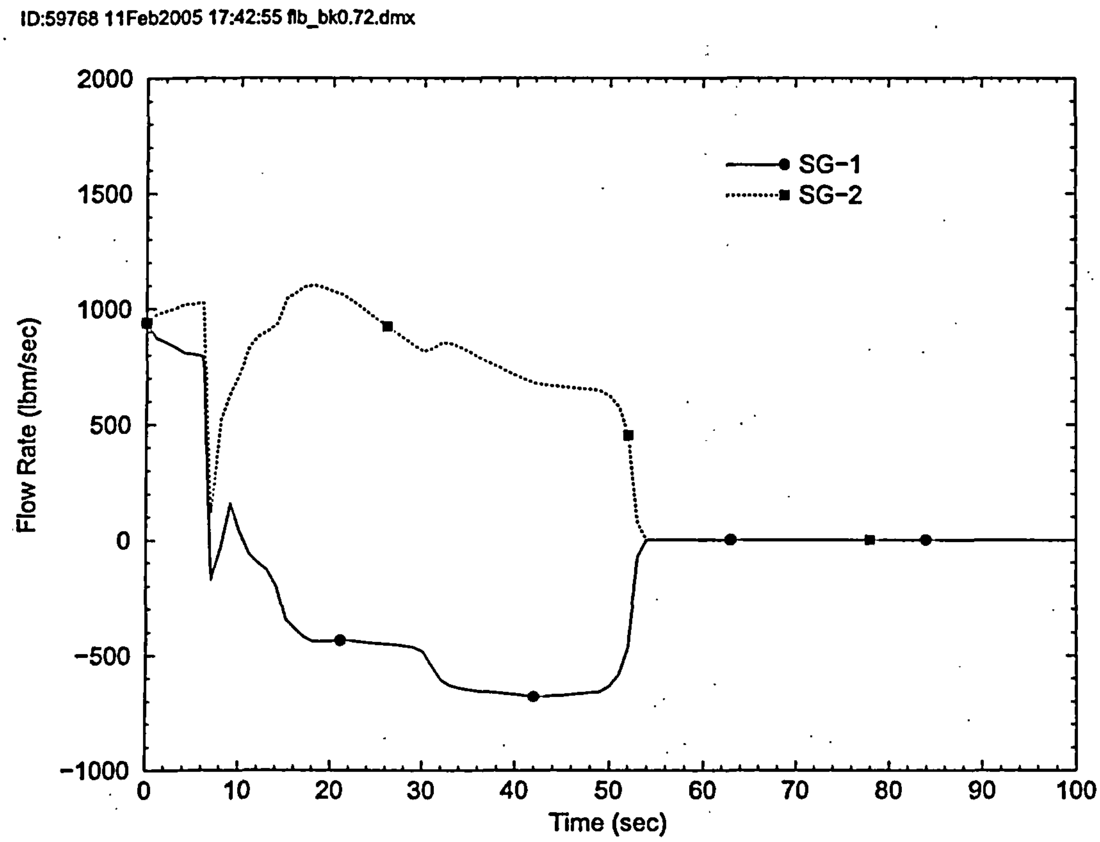

Figure 10 MSIV Flow Rate Versus Time for FWLB Event

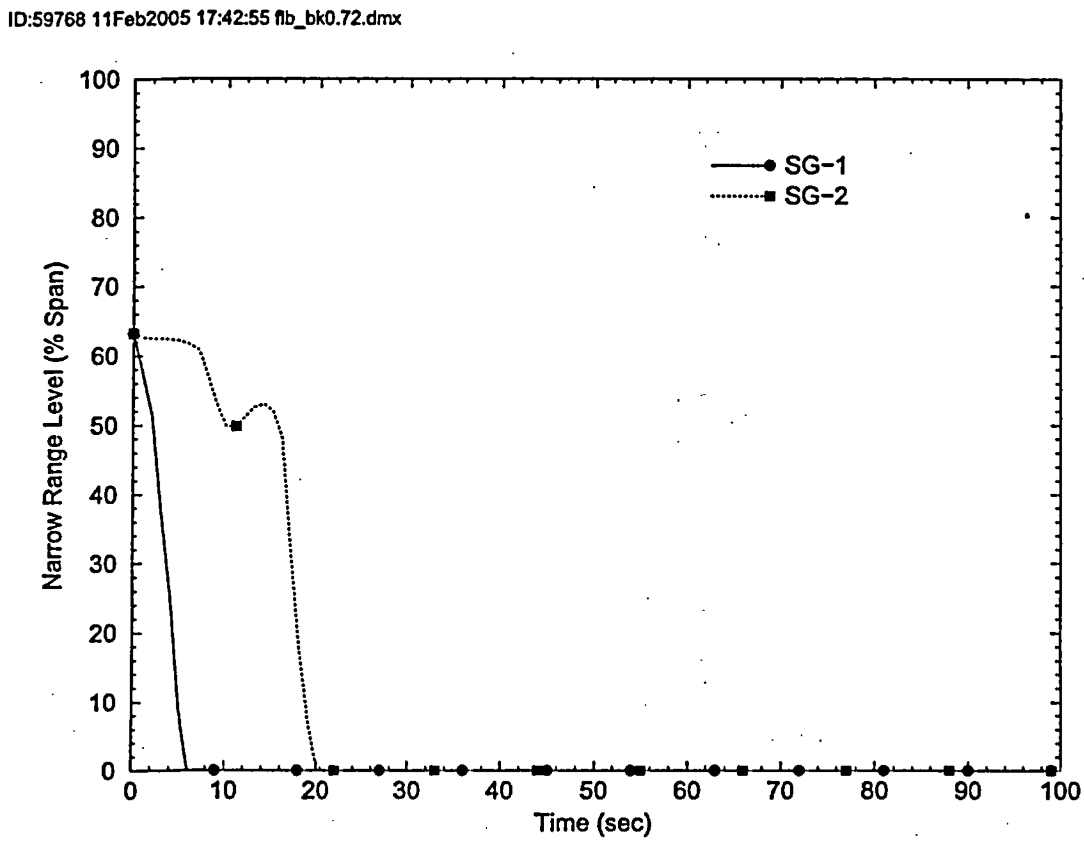

Figure 11 Steam Generator Narrow Range Level Versus Time for FWLB Event

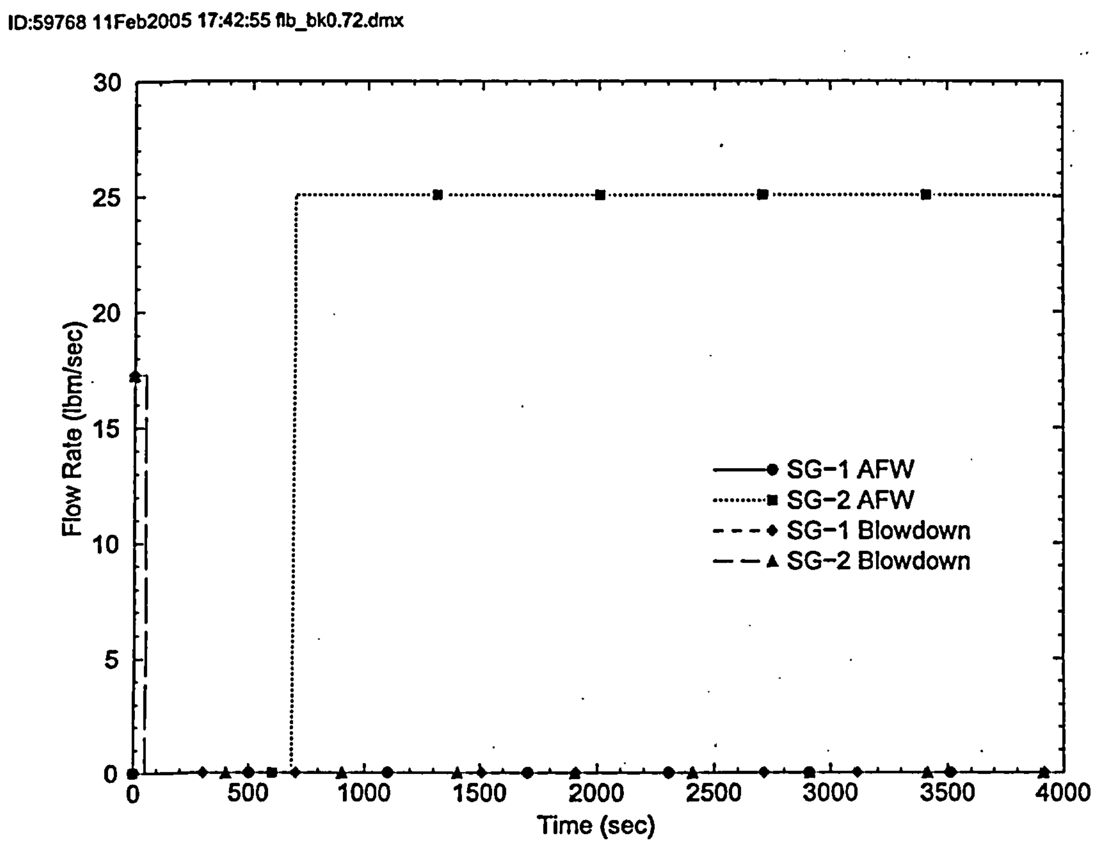

Figure 12 Auxiliary Feedwater and Steam Generator Blowdown Flow Rates Versus Time for FWLB Event

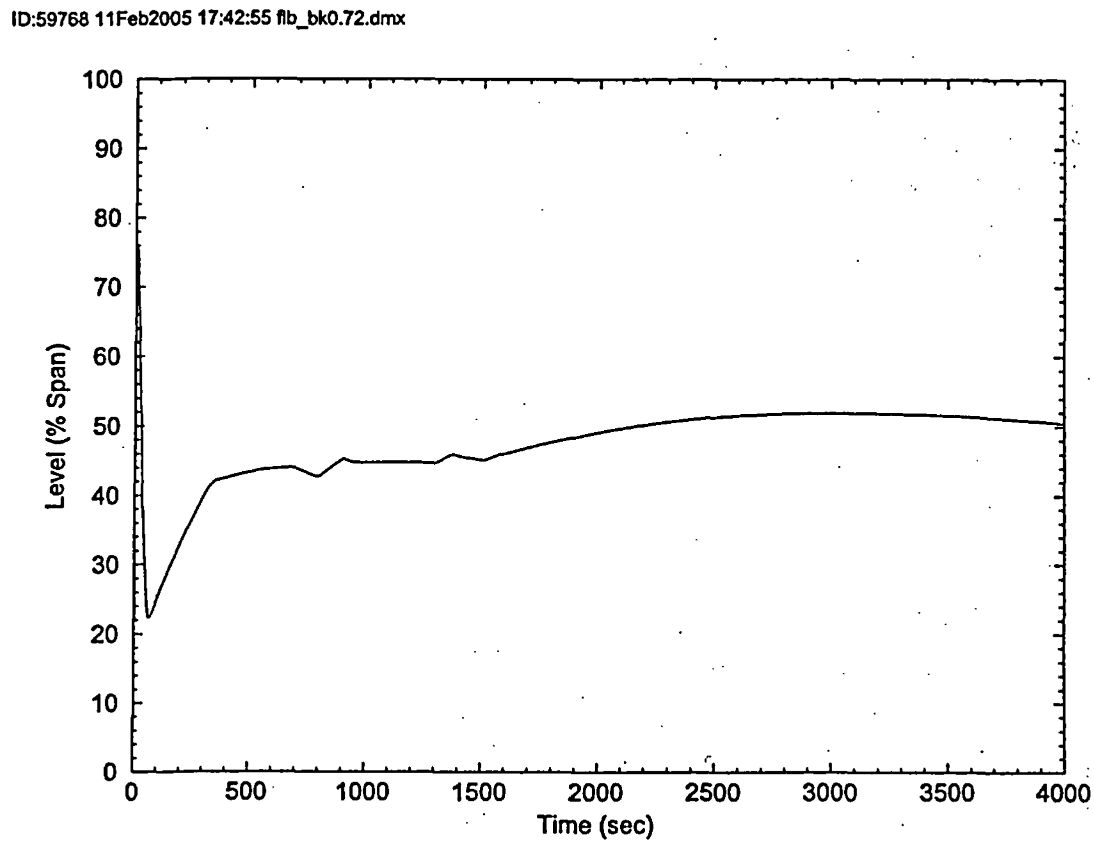

Figure 13 Pressurizer Level Versus Time for FWLB Event

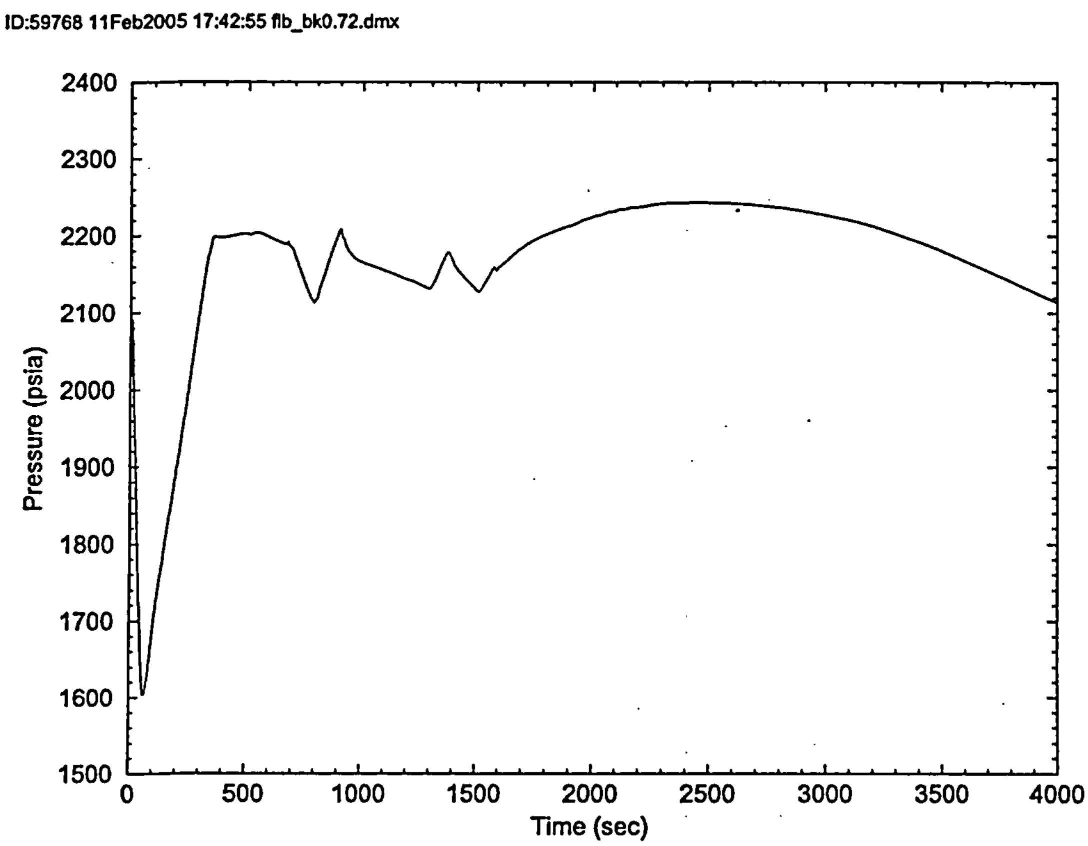

Figure 14 Pressurizer Pressure Versus Time for FWLB Event

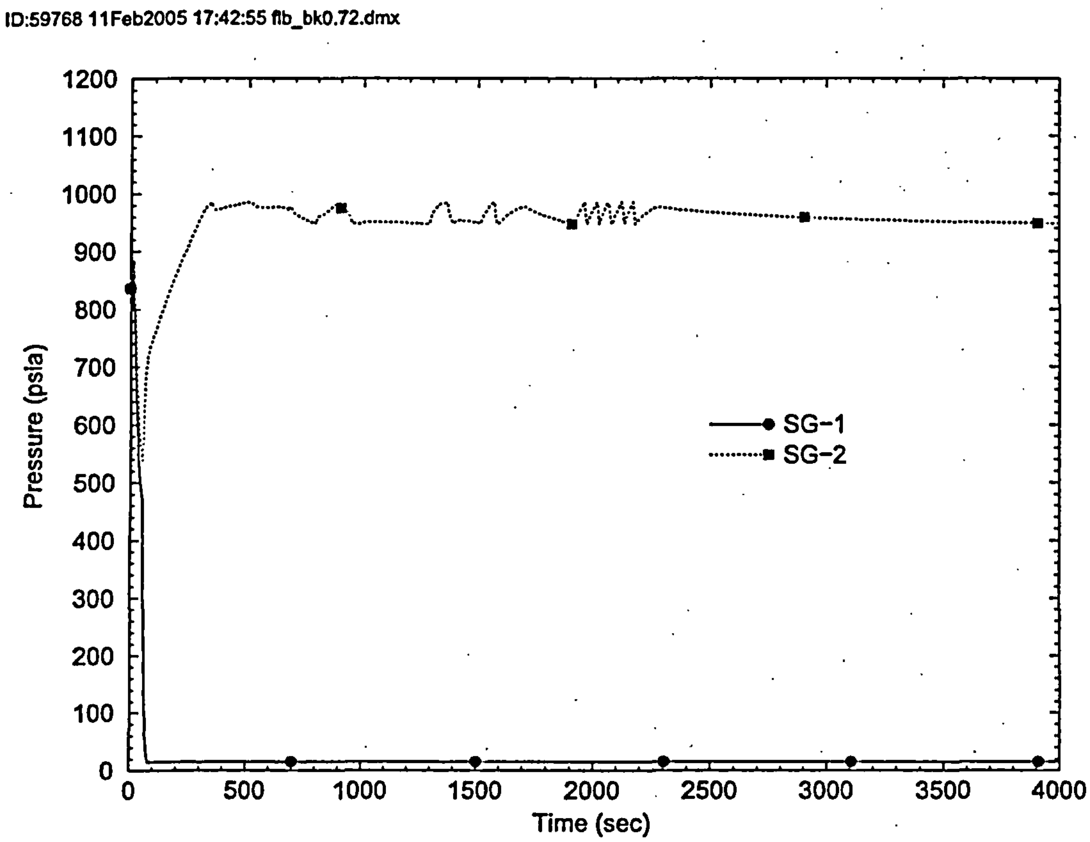

Figure 15 Steam Generator Pressure Versus Time for FWLB Event

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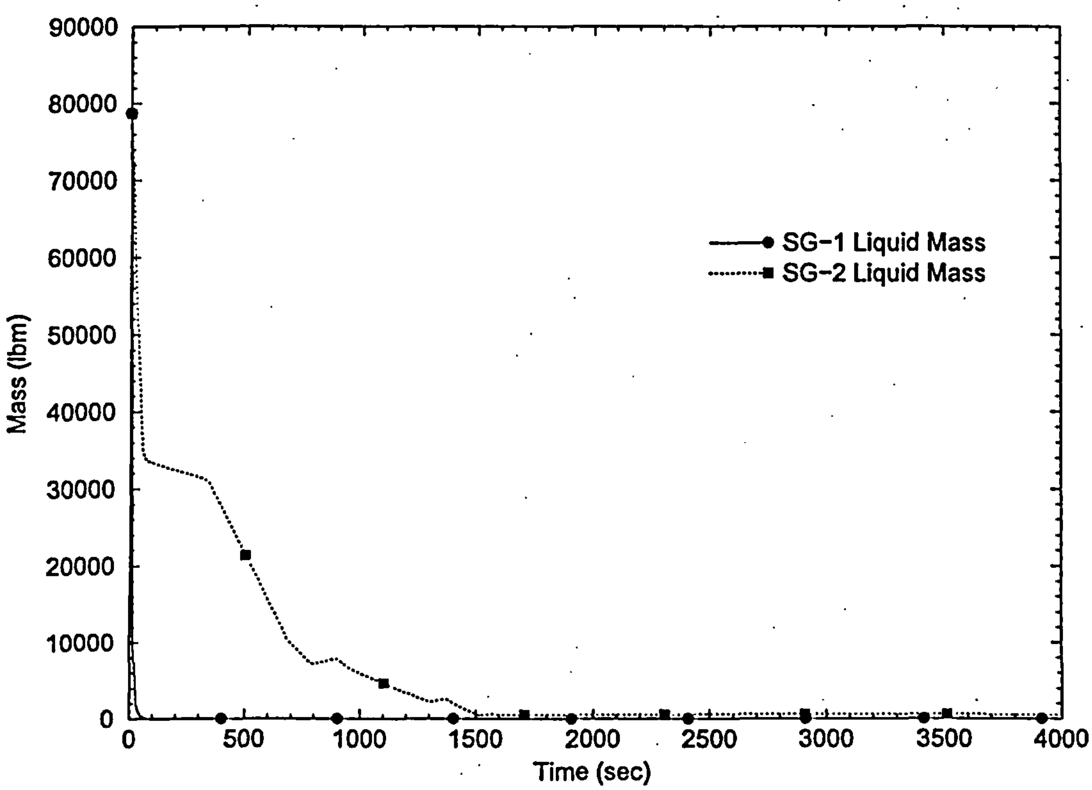

Figure 16 Steam Generator Liquid Mass Inventory Versus Time for FWLB Event

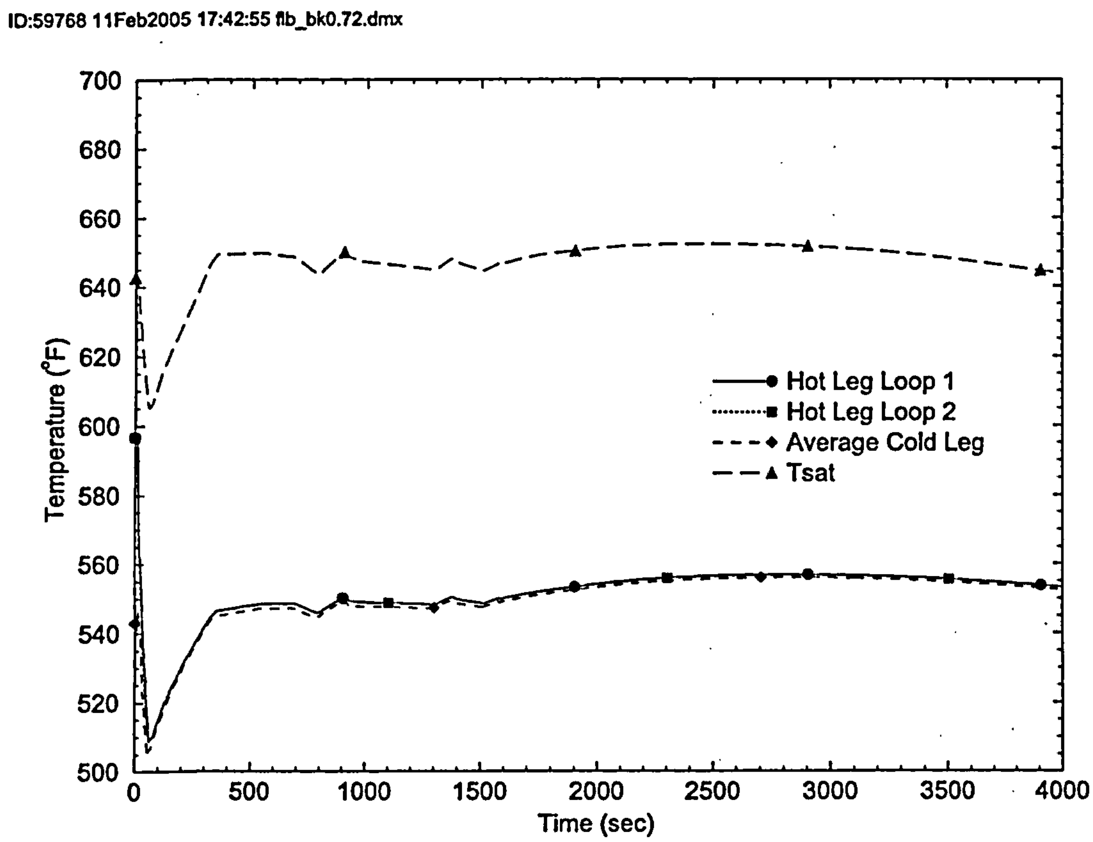

Figure 17 RCS Coolant Temperatures Versus Time for FWLB Event

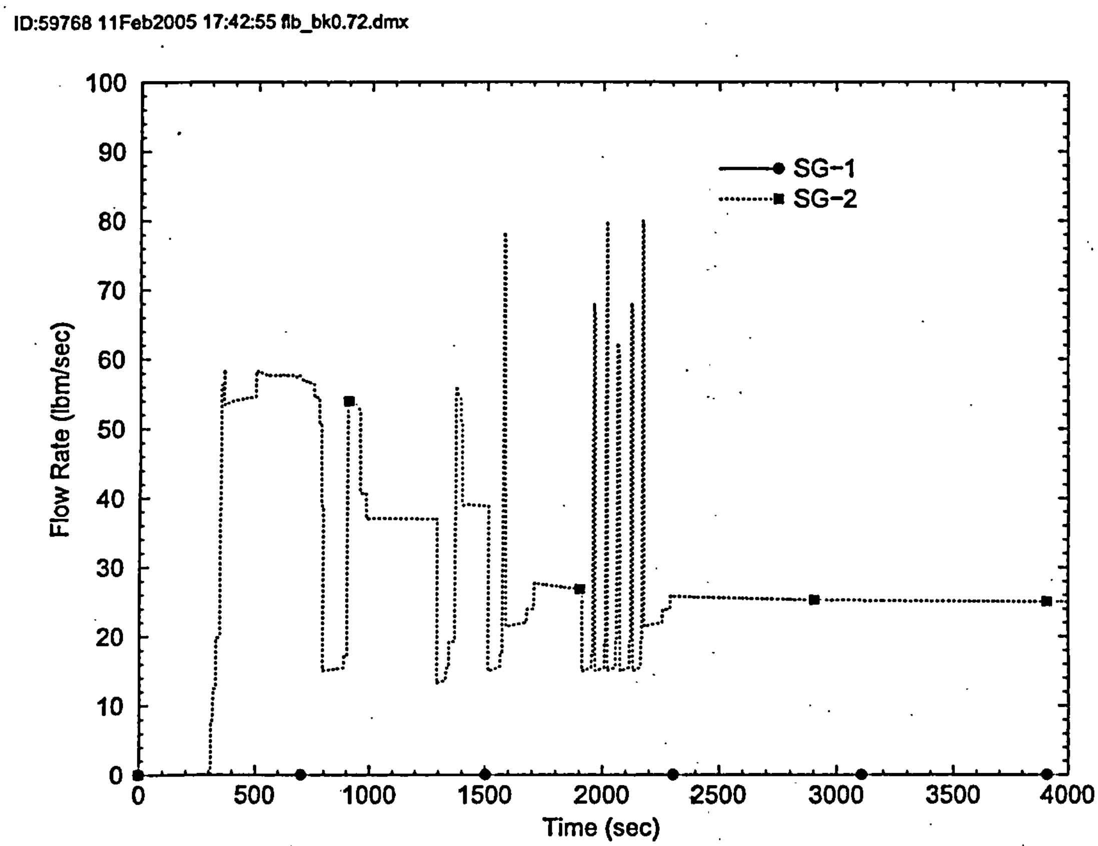

Figure 18 Total MSSV Flow Rate Versus Time for FWLB Event

2.0 REFERENCES

1. AREVA Document 32-5046049-00, *FCS RSG – Loss of Feedwater Flow Analysis,* October, 2004.

2. AREVA Document 32-5046821-01, *FCS RSG – Feedwater Line Break Analysis,* March 2005.

3. AREVA Document 38-5043229-00, *FCS RSG Licensing – Replacement Pressurizer inputs* (Westinghouse Letter LTR-NCE-04-8, dated January 19, 2004).

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DESIGN VERIFICATION CHECKLIST

Document Identifier 86 - 5056804 - 00

Title FCS RSG - AUXILIARY FEEDWATER ACTUATION SIGNAL (AFAS) SETPOINT VERIFICATION

1.	Were the inputs correctly selected and incorporated into design or analysis?	☒ Y	☐ N	☐ N/A
2.	Are assumptions necessary to perform the design or analysis activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent re-verifications when the detailed design activities are completed?	☒ Y	☐ N	☐ N/A
3.	Are the appropriate quality and quality assurance requirements specified? Or, for documents prepared per FANP procedures, have the procedural requirements been met?	☒ Y	☐ N	☐ N/A
4.	If the design or analysis cites or is required to cite requirements or criteria based upon applicable codes, standards, specific regulatory requirements, including issue and addenda, are these properly identified, and are the requirements/criteria for design or analysis met?	☒ Y	☐ N	☐ N/A
5.	Have applicable construction and operating experience been considered?	☒ Y	☐ N	☐ N/A
6.	Have the design interface requirements been satisfied?	☒ Y	☐ N	☐ N/A
7.	Was an appropriate design or analytical method used?	☒ Y	☐ N	☐ N/A
8.	Is the output reasonable compared to inputs?	☒ Y	☐ N	☐ N/A
9.	Are the specified parts, equipment and processes suitable for the required application?	☐ Y	☐ N	☒ N/A
10.	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?	☐ Y	☐ N	☒ N/A
11.	Have adequate maintenance features and requirements been specified?	☐ Y	☐ N	☒ N/A
12.	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?	☐ Y	☐ N	☒ N/A
13.	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?	☐ Y	☐ N	☒ N/A
14.	Has the design properly considered radiation exposure to the public and plant personnel?	☐ Y	☐ N	☒ N/A
15.	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?	☒ Y	☐ N	☐ N/A
16.	Have adequate pre-operational and subsequent periodic test requirements been appropriately specified?	☐ Y	☐ N	☒ N/A
17.	Are adequate handling, storage, cleaning and shipping requirements specified?	☐ Y	☐ N	☒ N/A
18.	Are adequate identification requirements specified?	☐ Y	☐ N	☒ N/A
19.	Is the document prepared and being released under the FANP Quality Assurance Program? If not, are requirements for record preparation review, approval, retention, etc., adequately specified?	☒ Y	☐ N	☐ N/A

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Comments:

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