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OPPD
Omaha Public Power District

*444 South 16th Street Mall*
*Omaha NE 68102-2247*

July 6, 2006
LIC-06-0038

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk) dated July 1, 2005, "Fort Calhoun Station Unit No. 1 License Amendment Request, "Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event" (LIC-05-0001)
3. Letter from NRC (A. B. Wang) to OPPD (R.T. Ridenoure) dated May 5, 2006, "Fort Calhoun Station, Unit 1 - Request for Additional Information Related to License Amendment Request for Updated Safety Analysis Report Clarification of Operator Actions during Loss of Main Feedwater Event (TAC No. MC7524)" (NRC-06-0052)

**SUBJECT: Response to Fort Calhoun, Unit 1 - Request for Additional Information Related to License Amendment Request for Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event (TAC No. MC7524)**

The Omaha Public Power District (OPPD) previously submitted an update to the Updated Safety Analysis Report in accordance with 10 CFR 50.90 to clarify existing operator actions during a Loss of Main Feedwater event (Reference 2). Attachment 1 of this letter provides the response to the NRC's additional questions presented in Reference 3 related to analyses supporting this USAR revision. Attachment 2 provides a summary of the Feedwater Line Break calculations which support the information contained in Attachment 1. Attachment 3 provides the applicable portions of Fort Calhoun Station's Emergency Operating Procedure – Standard Post Trip Actions (EOP-00) referred to in Attachment 1.

No commitments to the NRC are made in this letter. I declare under penalty of perjury that the foregoing is true and correct. (Executed July 5, 2006)

If you require additional information, please contact Thomas C. Matthews at (402) 533-6938.

Sincerely,

J. A. Reinhart
Site Director
Fort Calhoun Station

JAR/rlj

Attachments:

1. Response to Fort Calhoun, Unit 1 - Request for Additional Information Related to License Amendment Request for Updated Safety Analysis Report Clarification of Operator Action during Loss of Main Feedwater Event
2. FCS – AFAS Verification – FWLB – Inputs to Support Answers to NRC RAI, AREVA Document 51-9023091 - 000
3. Fort Calhoun Station – Emergency Operating Procedure – Standard Post Trip Actions (EOP-00), Revision 18 (pages 3, 16, 17, 18, 19, and 20, applicable to steam generator blowdown isolation)

cc: Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska

**Response to Fort Calhoun, Unit 1**
**Request for Additional Information Related to**
**License Amendment Request for Updated Safety Analysis Report**
**Clarification of Operator Action during Loss of Main Feedwater Event**

*The Nuclear Regulatory Commission (NRC) staff has performed a preliminary review of the human performance associated changes in the license amendment request. The licensee's responses to the following request for additional information with regard to containment analysis aspects of the license amendment are needed for the NRC staff to complete its review.*

*Page 22 of Attachment 4 to the licensee's letter of July 1, 2005, stated that for the feedwater line break (FLB) analysis that "SG blowdown flow is automatically and conservatively isolated at 49.03 sec ...." The NRC staff has noted that in FCS Technical Specification Table 2-1, a containment pressure high signal (CPHS) or a pressurizer pressure low signal (PPLS) would actuate a containment isolation actuation signal that isolates SG blowdown flow. As such:*

**NRC Question a:**

*Please indicate which signal (a CPHS or a PPLS) was credited to automatically isolate the SG in the analysis. Demonstrate that the FLB case with SG blowdown isolation occurring at 49.03 seconds is the limiting break case (considering full-break sizes of the FLB) with respect to the acceptance criterion of the peak post-trip hot-leg temperature of 600 degrees F.*

**OPPD Response to Question a:**

The original FLB analysis with S-RELAP5 in Attachment 4 of OPPD's License Amendment Request of July 1, 2005, conservatively assumed that the steam generator blowdown flow was isolated on a steam generator low pressure signal, rather than on a containment isolation actuation signal based on either a containment pressure high signal (CPHS) or a pressurizer pressure low signal (PPLS). Based on this analysis the SG blowdown isolation as a function of break size is shown in Table 1.

**Table 1**
**Break Size versus Time of SG Blowdown Isolation**

| Break Size (%) | Time of SG Blowdown Isolation (sec) |
|---|---|
| 100 | 30.14 |
| 90 | 36.26 |
| 80 | 49.03 |
| 70 | 60.53 |
| 60 | 75.25 |

This analysis demonstrated that the 80% break size associated with 49.03 sec for SG blowdown isolation is the most limiting break case with respect to peak post trip hot leg temperature of

600°F. A representative small break (5%) is evaluated and discussed in Attachment 2 of this letter, AREVA document 51-9023091-000 "FCS – AFAS Verification – FWLB – Inputs to support Answers to NRC RAI." This evaluation calculated the maximum post-trip hot leg temperature for the 5% break size case. The maximum post-trip hot leg temperature was 558.9°F (which is less than the criterion of 600°F but 2°F higher than the 80% break size case), assuming that the operator isolates steam generator blowdown flow within 15 minutes of reactor trip. It is important to note that in all instances, the FLB peak hot leg temperature is less than the peak hot leg temperature for a loss of main feedwater event (566.9°F).

An analysis was performed to confirm the conservative nature of the assumption on using the SG low pressure signal versus CPHS or PPLS to isolate SG blowdown. The time of CPHS was calculated with an S-RELAP5 model for feedwater line break sizes of 100% and 60% and the results are compared in Table 2 below with CPHS times calculated by a GOTHIC model of the Fort Calhoun containment and with the times of blowdown isolation calculated with S-RELAP5 based on SG low pressure signal. The results indicated that the S-RELAP5 model predicted conservatively longer times to CPHS than the GOTHIC model and that the SG blowdown isolation times based on a SG low pressure signal were conservative relative to either CPHS times with GOTHIC or S-RELAP5.

**Table 2**

**Comparison of Time of SG Blowdown Isolation**

| Break Size (%) | Time of CPHS Based on GOTHIC Model (sec) | CPHS – GOTHIC (sec)* | CPHS – S-RELAP5 (sec)* | SG Low Pressure S-RELAP5 (sec)** |
|---|---|---|---|---|
| 100 | 2.7 | 22.7 | 24.1 | 30.14 |
| 60 | 4.3 | 24.3 | 26.9 | 75.25 |

* Includes 20 sec SG blowdown valve stroke time

** The SG blowdown isolation valves were shut instantaneously when the signal was reached.

**NRC Question b:**

*If OPPD credited a CPHS to isolate SG blowdown flow, please discuss how the CPHS time was determined for FLB cases with break sizes ranging from the maximum break size to the smallest credible size.*

**OPPD Response to Question b:**

A simplified S-RELAP5 containment model was developed to evaluate times to CPHS for FLB break sizes of 100% and 60%. The simplified model demonstrated in both instances the times to CPHS with S-RELAP5 are conservative relative to the time to CPHS with GOTHIC.

Additionally, the time to SG blowdown isolation with S-RELAP5 based on SG low pressure signal is conservative relative to the time to CPHS with S-RELAP5 as shown in Table 2 above.

In addition to what was analyzed in Attachment 4 of OPPD's License Amendment Request of July 1, 2005, another FLB calculation was performed for a 10% break size with the S-RELAP5 containment model (see discussion on Page 7 and resulting values in Table 4 of Attachment 2 to this letter). The purpose of this analysis was to demonstrate that a CPHS would occur for break sizes down to 10% such that automatic isolation of steam generator blowdown flow would occur. This calculation showed that the isolation would occur at 132.5 sec (including a 20 sec steam generator blowdown valve stroke time) and the maximum post-scram hot leg temperature is less than the 600 °F criterion.

For FLB breaks less than 10%, it was assumed that CPHS will not occur. For such cases the operator would isolate steam generator blowdown flow within 15 minutes, as directed by EOP-00 step 13 following the reactor trip.

**NRC Question c:**

*Please provide a description of the containment model to show its adequacy for the CPHS time determination (with respect to prediction of a lower containment pressure resulting in a high CPHS time).*

**OPPD Response to Question c:**

The simplified S-RELAP5 containment model consists of a single volume representing containment. The volume of the containment is the same as that used in Appendix K Large Break Loss of Coolant Accident (LOCA) calculations. A heat structure is attached to that volume, representing containment structures. The left side of the heat structure is attached to the containment volume and the S-RELAP5 code is allowed to calculate the heat transfer coefficients for a convective boundary. The heat transfer area is based on the total heat transfer area from the Large Break LOCA ICECON containment model. The right side of the heat structure has heat transfer coefficients and convective boundary temperature conditions that produce a conservatively low containment pressure. The model was previously benchmarked against an Appendix K Large Break LOCA containment model (ICECON), which included pressure suppression systems such as sprays and fan coolers and which is biased to produce a conservatively low containment pressure. The S-RELAP5 containment model was shown to be equal to or conservative (low) relative to the ICECON model and is applicable for a steam generator blowdown period of up to 150 sec. The simplified S-RELAP5 containment model is the same as that developed for the Main Steam Line Break (MSLB) analyses of Reference 2 of Attachment 2 of this letter, which was approved by the NRC in Reference 3 of Attachment 2 to this letter.

**NRC Question d:**.

*For the small-size FLBs that would not result in a CPHS, please discuss the available Emergency Operating Procedures for the operator to manually isolate SG blowdown flow, and show that the proposed operation action (15 minutes after a reactor trip) for the Loss of Main Feedwater (LMFW) event is sufficient for the FLB event.*

**OPPD Response to Question d:**

For FLB breaks less than 10%, it was assumed that CPHS will not occur. For such cases the operator would isolate steam generator blowdown flow within 15 minutes as directed by Fort Calhoun Station – Emergency Operating Procedure – Standard Post Trip Actions (EOP-00), step 13 after a reactor trip (see Attachment 3 of this letter). EOP-00 step 13 directs the operator to isolate steam generator blowdown flow following a reactor trip if main feedwater is not restoring level in at least one steam generator to 35 – 85% Narrow Range (73 – 94% Wide Range). With the assumption of loss of offsite power at the time of the reactor trip following a FLB, main feedwater is lost at the time of reactor trip and the level in both steam generators will be decreasing. A calculation was performed for a representative small break size (5%). Assuming that the operator isolates steam generator blowdown flow within 15 minutes of reactor trip, the calculation concludes that the maximum post-trip hot leg temperature would be 558.9°F, which is less than the criterion of 600°F. Without the assumption of loss of offsite power at the time of the reactor trip, main feedwater would continue, level in both steam generators would not decrease further in any steam generator since there is no nuclear heating, and the main feedwater would be more than adequate to remove the decay heat. Therefore, it is concluded that FWLB events of less than 10% break sizes progress similarly to the loss of feedwater flow events. EOP-00 post-trip actions are adequate to address very small (<10%) break sizes. Hence, no additional administrative processes need to be considered or added.

**FCS - AFAS Verification – FWLB - Inputs to Support Answers to NRC RAI**

AREVA Document 51-9023091 - 000

AREVA

# ENGINEERING INFORMATION RECORD

Document Identifier 51 - 9023091 - 000

Title FCS - AFAS Verification - FWLB - Inputs to Support Answers to NRC RAI

**PREPARED BY:**

Name R. C. Gottula

Signature R. C. Gottula Date 6/26/06

**REVIEWED BY:**

Name T. R. Lindquist

Signature TR Lindquist Date 6/26/06

Technical Manager Statement: Initials for S. Frantz

Reviewer is Independent.

Remarks:

This document provides a summary of Feedwater Line Break calculations which support responses to an NRC RAI regarding verification of the Auxiliary Feedwater Actuation Signal.

# FCS - AFAS Verification – FWLB – Inputs to Support Answers to NRC RAI

Omaha Public Power District
Fort Calhoun Station

AREVA
Document 51-9023091-000
June 2006

## Record of Revisions

| Revision | Description | Date | Changed Pages |
|---|---|---|---|
| 000 | Initial release | 06/2006 | |

## Table of Contents



## List of Tables

## 1.0 Summary

The purpose of this document is to provide a summary of Feedwater Line Break (FWLB) calculations which support responses to an NRC Request for Additional Information (RAI) (Reference 1) regarding Auxiliary Feedwater Actuation Signal (AFAS) verification.

The original FWLB analysis with S-RELAP5 analyzed feedwater line break sizes of 100% (0.90 $ft^2$), 90% (0.81 $ft^2$), 80% (0.72 $ft^2$), 70% (0.63 $ft^2$), and 60% (0.54 $ft^2$). The analysis conservatively assumed that steam generator blowdown flow was isolated on a Steam Generator Low Pressure Signal, rather than on a Containment Isolation Actuation Signal (CIAS) based on either a Containment Pressure High Signal (CPHS) or a Pressurizer Pressure Low Signal (PPLS). The steam generator isolation times used for these cases are shown in Table 1.

**Table 1 SG Blowdown Flow Isolation Times Used in FWLB Calculations**

| Break Size (%) | Time of SG Blowdown Flow Isolation Based on a SG Low Pressure Signal (sec) |
|---|---|
| 100 | 30.14 |
| 90 | 36.26 |
| 80 | 49.03 |
| 70 | 60.53 |
| 60 | 75.25 |

The above times for steam generator blowdown flow isolation were demonstrated to be conservative by comparing to the predicted times of a CPHS based on a Fort Calhoun GOTHIC containment model for the 100% (0.90 $ft^2$) and 60% ($ft^2$) break sizes. Table 2 shows a comparison of the time of a CPHS based on the GOTHIC model to the time of steam generator blowdown flow isolation used in the FWLB analysis.

**Table 2 Comparison of GOTHIC CPHS Times to SG Blowdown Flow Isolation Times Used in FWLB Calculations**

| Break Size (%) | Time of CPHS Based on GOTHIC Model (sec) | Time of SG Blowdown Flow Isolation Based on GOTHIC Model Prediction of CPHS Plus a 20 sec SG Blowdown Valve Stroke Time (sec) | Time of SG Blowdown Isolation Used in S-RELAP5 FWLB Calculations Based on SG Low Pressure Signal (sec) |
|---|---|---|---|
| 100 | 2.7 | 22.7 | 30.14 |
| 60 | 4.3 | 24.3 | 75.25 |

Thus, the times used for steam generator blowdown isolation in the FWLB calculations were conservative relative to predictions based on the Fort Calhoun GOTHIC model.

A simplified S-RELAP5 containment model was developed to evaluate times to a CPHS for break sizes less than 60% (0.54 $ft^2$). The simplified S-RELAP5 containment model is the same as that developed for Main Steam Line Break (MSLB) analyses and which was previously approved as part of the Reference 2 methodology by the NRC in Reference 3. Following is a description of the S-RELAP5 containment model.

The model consists of a single volume representing the containment. The volume of the containment is the same as that used for Appendix K Large Break LOCA calculations. A heat structure is attached to that volume, representing containment structures. The left side of the heat structure is attached to the containment volume and the S-RELAP5 code is allowed to calculate the heat transfer coefficients for a convective boundary. The heat transfer area is based on the total heat transfer area from the Large Break LOCA ICECON containment model. The right side of the heat structure has heat transfer coefficients and boundary temperature conditions that produce a conservatively low containment pressure. The model was previously benchmarked against an Appendix K Large Break LOCA containment model (ICECON), which included pressure suppression systems such as sprays and fan coolers and which is biased to produce a conservatively low containment pressure. The S-RELAP5 containment model was shown to be equal to or conservative (low) relative to the ICECON model and is applicable for a steam generator blowdown period of up to 150 sec.

S-RELAP5 calculations, incorporating the S-RELAP5 containment model, were performed for the 100% (0.90 $ft^2$) and 60% (0.54 $ft^2$) break sizes to demonstrate that the S-RELAP5 simplified containment model provides a conservative prediction of the time to a CPHS relative to GOTHIC model predictions. Table 3 shows a comparison of the predicted times to a CPHS between the GOTHIC model and the simplified S-RELAP5 containment model including a 20 sec steam generator blowdown valve stroke time.

**Table 3 Comparison of GOTHIC to S-RELAP5 CPHS Times**

| Break Size (%) | Time to CPHS – GOTHIC Model[a] (sec) | Time to CPHS – S-RELAP5 Model[a] (sec) |
|---|---|---|
| 100 | 22.7 | 24.1 |
| 60 | 24.3 | 26.9 |

It can be seen that the times to a CPHS are larger with the S-RELAP5 model for break sizes from 100% to 60%. Thus, the S-RELAP5 containment model is expected to provide reasonable and conservative times to a CPHS for break sizes less than 60%

A FWLB calculation was performed for a break size of 10% (0.09 $ft^2$), incorporating the S-RELAP5 containment model, to demonstrate that a CPHS would occur for break sizes down to 10% such that automatic isolation of steam generator blowdown flow would occur. This calculation showed that a CPHS would occur at 132.5 sec (including a 20 sec steam generator blowdown valve stroke time). The maximum post-scram hot leg temperature for this case was 549.9°F, which is less than the criterion of <600°F.

It was assumed that a CPHS will not occur for break sizes less than 10%. For such cases, it was assumed that an operator action would be required to isolate steam generator blowdown flow within 15 minutes after reactor trip, similar to the Loss of Feedwater Flow event. In fact, EOP-00 Step 13 directs the operator to isolate steam generator blowdown flow following a reactor trip if main feedwater is not restoring level in at least one steam generator to 35 – 85% NR (73 – 94% WR). If offsite power is lost at the time of reactor trip, main feedwater would also be lost at the time of reactor trip such that the level in both steam generators would be decreasing. For this case, the operators would isolate steam generator blowdown flow. If offsite power is not lost at the time of reactor trip, MFW would also not be lost such that the level

[a] Includes 20 sec steam generator valve stroke time.

in both steam generators would be maintained, decay heat would be removed, and isolation of steam generator blowdown would not be required.

A calculation was performed for a representative very small break size [5% (0.045 $ft^2$)], assuming steam generator blowdown is isolated 15 min. after reactor trip by operator action, to demonstrate that the criterion (maximum post-scram hot leg temperature <600°F) would be satisfied. The results indicated that the maximum post-scram hot leg temperature was 558.9°F, which is less than the criterion, similar to the larger break sizes, and several degrees less than the value for the Feedwater Flow event (566.9°F). A FWLB event with a very small break size progresses similar to the Loss of Feedwater Flow event.

Table 4 provides a summary of the maximum post-scram hot leg temperature for the various FWLB cases analyzed.

**Table 4 Summary of FWLB Maximum Post-Scram Hot Leg Temperatures (Break Spectrum)**

| Break Size (%) | Maximum Post-Scram Hot Leg Temperature (°F) |
|---|---|
| 100 | 556.6 |
| 90 | 554.8 |
| 80 | 556.8 |
| 70 | 555.0 |
| 60 | 554.3 |
| 10 | 549.9 |
| 5 | 558.9 |

In conclusion, these calculations demonstrate that the simplified S-RELAP5 model is satisfactory to predict the time of a CPHS. The results in Table 4 indicate that the maximum post-scram hot leg temperature is similar for all cases analyzed. Break sizes that do not experience a CPHS require operator action within 15 minutes of reactor trip to isolate steam generator blowdown flow. The maximum post-scram hot leg temperature for all cases analyzed is slightly less than that for a Loss of Feedwater event (566.9°F). The criterion of maximum post-scram hot leg temperature of <600°F is satisfied for all FWLB cases and more than 20°F subcooling margin is maintained.

## 2.0 References

1. AREVA Document 38-9023507-000, Letter, NRC to OPPD, "Fort Calhoun Station Unit 1 – Request for Additional Information Related to License Amendment Request for Updated Safety Analysis Report Clarification of Operator Action During Loss of Main Feedwater Event (TAC No. MC7524)," May 5, 2006.

2. AREVA Document EMF-2310(P)(A) Revision 1, *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*, May 2004.

3. Letter, H. N. Berkow (NRC) to J. F. Mallay (AREVA), Final Safety Evaluation for Topical Report EMF-2310(P), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors" (TAC No. MC0329), May 19, 2004. [This letter is contained in Reference 2 above.]

**Fort Calhoun Station – Emergency Operating Procedure – Standard Post Trip Actions**

**(EOP-00)**

**Revision 18**

(pages 3, 16, 17, 18, 19, and 20, applicable to steam generator blowdown isolation)

# 4.0 PRECAUTIONS

The following specific cautions, notes and procedure expectations apply prior to or throughout this procedure.

## A. CAUTIONS

1. Common cause failure of a standby pump or component is possible if started following a pump or component failure. The cause of the failure should be determined prior to starting or restarting a standby pump or component.

## B. NOTES

1. Minimizing DC Loads requires operator action within 15 minutes of loss of either battery charger.

2. S/G Blowdown must be isolated within 15 minutes of a reactor trip when a loss of Main Feedwater to both steam generators occurs.

3. Do **NOT** adopt manual operation of automatically controlled systems unless a malfunction is apparent or the automatic system operation will **NOT** support the maintenance of a safety function.

4. Systems shifted to manual operation must be monitored frequently to ensure correct operation.

5. An incident may cause inconsistencies between instruments. A least **TWO** independent indications should be used, when available, to evaluate and verify a specific plant condition.

## C. PROCEDURE EXPECTATIONS

1. The Reactor Operators are expected to report the status of the Reactivity Control Safety function (Steps 1 and 2) to the CRS/SM immediately following the reactor trip.

2. With the exception of Steps 1, 2, and 16, steps may be performed in any order as dictated by plant conditions.

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

**NOTE**

If Instrument Air to valves HCV-1105, HCV-1106, HCV-1107A/B and HCV-1108A/B is not available, throttling of these valves is not possible. Open or close operation of these valves is possible for a minimum of three cycles.

13. Verify Main Feedwater is restoring level in at least one S/G to 35-85% NR (73-94% WR) by performing the following:

    a. Ensure **BOTH** of the Feed Reg Valves have ramped to 7-9%:

       - FCV-1101
       - FCV-1102

    b. Ensure no more than one Feed Pump is operating.

    c. Ensure no more than one Condensate Pump is operating.

    d. Stop all operating Heater Drain Pumps, FW-5A/B/C.

    e. Place the "43/FW" Switch in "OFF".

(continue)

13.1 **IF** Main Feedwater lost,

**THEN** isolate S/G Blowdown by closing **BOTH** sets of Blowdown Isolation Valves:

- HCV-1387A/B
- HCV-1388A/B

(continue)

INSTRUCTIONS

13. (continued)

(continue)

CONTINGENCY ACTIONS

******************************************************

**CAUTION**

When S/G level has dropped below the Feed Ring (29% NR), feeding at a rate of less than or equal to 150 gpm per S/G for the first five minutes will minimize the possibility of Feed Ring damage from water hammer.

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13.2 **IF** Main Feedwater is **NOT** restoring S/G level **AND** SGLS has not actuated, **THEN** establish Feedwater by performing step a, b, c, d or e:

a. **IF** a Main Feedwater Pump is operating,
**THEN** restore level in at least one S/G to 35-85% NR (73-94% WR) by manual control of Main Feedwater through the Feed Ring.

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

13. (continued)

(continue)

13.2 (continued)

b. Initiate AFW using FW-54, Diesel AFW Pump to the Feed Ring:

1) Start FW-54.

2) Restore level in at least one S/G to 35-85% NR (73-94% WR).

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**CAUTION**

S/G levels less than 85% NR (94% WR) may cause undesired thermal cycles or water hammer.

*****************************************************

c. Initiate AFW using FW-6 or FW-10, AFW Pumps to the AFW Nozzles:

1) Start at least one of the following AFW Pumps:

- FW-6
- FW-10

(continue)

13. (continued)

(continue)

13.2.c. (continued)

2) Restore level in at least one S/G to 85-95% NR (94-98% WR) via the AFW Nozzles.

d. Initiate AFW using FW-6 or FW-10, AFW Pumps to the Feed Ring:

1) Start at least one of the following AFW Pumps:

- FW-6
- FW-10

2) Open HCV-1384, "AUX FW/FW HEADER CROSS CONNECT VALVE".

3) Restore level in at least one S/G to 35-85% NR (73-94% WR).

(continue)

INSTRUCTIONS

13. (continued)

(continue)

CONTINGENCY ACTIONS

13.2 (continued)

****************************************************

**CAUTION**

S/G levels less than 85% NR (94% WR) may cause undesired thermal cycles or water hammer.

****************************************************

e. Initiate AFW using FW-54, Diesel AFW Pump to the AFW Nozzles:

1) Start FW-54.

2) Unlock and close FW-170, "AUX FEEDWTR X-TIE VLV HCV-1384 OUTLET VALVE" (Room 81).

3) Open HCV-1384, "AUX FW/FW HEADER CROSS CONNECT VALVE".

4) Restore level in at least one S/G to 85-95% NR (94-98% WR).