AFW

Corrective

Actions

Book 1

A/15

## STATE CHANGE HISTORY

Initiate → AR Screening Que
12/19/2001 9 36 04 PM
Owner SCOTT PFAFF
by JULIE KREIL

## SECTION 1

| | | | |
|---|---|---|---|
| Activity Request Id: | CAP001415 | | |
| Activity Type: | CAP | Submit Date: | 11/29/2001 1:00.00 AM |
| One Line Description: | Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW | | |

**Detailed Description:**

While performing an update to the Auxiliary Feed Water (AFW) System model in the PRA, a procedural shortcoming was identified in AOP 5B with regards to the availability of the minimum recirculation valve with the loss of instrument air. This issue was documented in CR 01-2278 with a recommendation to upgrade the procedure. Upon further review of this issue with PRA engineers, Operations, and Design Engineering, it was discovered that this issue has further reaching affects as documented below.\\Instrument air (IA) can be lost primarily by two failure mechanisms. The first, and most likely, is a loss of off-site power where the IA and Service Air (SA) compressors are stripped from the bus and not automatically re-loaded. The second less likely scenario is a random loss of the instrument air system due to equipment failure without potential for short term recovery. When IA is lost, the minimum flow recirculation valves for AFW fail closed.\\During these two transients, the AFW pumps will start injecting into the steam generators. Early in the EOPs, the operator is directed to control flow to the steam generators to maintain desired level. This may include shutting off flow to one or both steam generators if level is above the desired band If flow from any auxiliary feed pump is reduced too low (as would occur if the auxiliary feed regulating valves are closed) without functional recirculation valves, the pump will fail in a very short period of time. This common mode of failure (common loss of instrument air and common response to high steam generator level) could result in simultaneous failure of all AFW pumps.\\PRA has estimated the risk associated with this issue. The total risk increase due to both the loss of off-site power and loss of instrument air contribution is approximately a factor of 4 times higher than our assumed base risk with an overall increase in the area of 2E-4 CDF per year (base risk is around 5E-5 CDF per year).\\WHY DID EVENT/ISSUE OCCUR? Current design of plant - deficiency not previously recognized.\\SIGNIFICANCE/REQUIREMENT NOT MET: See description. Potential common failure mode for all auxiliary feed pumps under certain initiating events.\\CORRECTIVE ACTIONS TAKEN: Operations has been notified and action is being taken to brief operation crews and provide temporary instruction for the operation of the AFW discharge valves.\\RECOMMENDATIONS: 1) Engineering needs to further evaluate and determine long term corrective action.\\2) PRA needs to evaluate and provide guidance for short term Maintenance Rule risk monitoring until new model is implemented.

| | | | |
|---|---|---|---|
| Initiator: | MASTERLARK, JAMES | Initiator Department: | EPN Engineering Programs Nuclear Safety Analysis PB |
| Date/Time of Discovery: | 12/19/2001 9:36.04 PM | Date/Time of Occurrence: | 12/19/2001 9.36:04 PM |
| Identified By: | (None) | System: | (None) |
| Equipment # (1st): | (None) | Equipment Type (1st): | (None) |
| Equipment # (2nd): | (None) | Equipment Type (2nd) : | (None) |
| Equipment # (3rd): | (None) | Equipment Type (3rd) : | (None) |
| Site/Unit: | Point Beach - Common | | |
| Why did this occur?: | | | |
| Immediate Action Taken: | | | |
| Recommendations: | | | |
| Notify Me During Eval?: | N | SRO Review Required?: | N |

## SECTION 2

| | | | |
|---|---|---|---|
| Operability Status: | (None) | Compensatory Actions: | N |
| Basis for Operability: | | | |
| Unplanned TSAC Entry: | N | External Notification: | Y |

## SECTION 3

| | | | |
|---|---|---|---|
| Screened?: | Y | Significance Level: | A |
| INPO OE Reqd?: | N | Potential MRFF?: | N |
| QA/Nuclear Oversight?: | N | Licensing Review?: | N |
| Good Catch/Well Doc'd?: | NA | | |

## SECTION 4

| | | | |
|---|---|---|---|
| Inappropriate Action: | | | |
| Process: | (None) | Activity: | (None) |
| Human Error Type: | (None) | Human Perf Fail Mode: | (None) |
| Equip Failure Mode: | (None) | Process Fail Mode: | (None) |
| Org/Mgt Failure Mode: | (None) | Group Causing Prob: | (None) |
| Hot Buttons: | (None) | | |

## SECTION 5

| | | | |
|---|---|---|---|
| CAP Admin: | SCOTT PFAFF | Prescreener: | (None) |
| Project: | Corrective Action Process (CAP) | | |
| State: | AR Screening Que | Active/Inactive: | Active |
| Submitter: | JULIE KREIL | Owner: | SCOTT PFAFF |
| AR Type: | Parent | Last Modified Date: | 9/12/2002 9.44:02 AM |
| Last Modifier: | RICHARD FLESSNER | Last State Change Date: | 12/19/2001 9.36:04 PM |
| Last State Changer: | JULIE KREIL | Close Date: | |
| NUTRK ID: | CR 01-3595 | | |
| # of Children: | 4 | | |
| References: | CR 01-2278<br>RCE 01-069<br>GOOD CATCH | | |

Update: È\\(20011204 PB2171 JMK1) Operability Determination (OD) Part I, Revision 0, of CR 01-3595 was approved on 11/30/01. Operable But Degraded - or Operable But Nonconforming - meets the minimum required level of performances, compensatory measures ARE required
\\Operability Determination (OD) Part I, Revision 1 of CR 01-3595 was approved on 12/01/01. Operable But Degraded - or Operable But Nonconforming - meets the minimum required level of performances, compensatory measures ARE required.
Operability Determination (OD) Part 1, Revision 2 of CR 01-3595 was approved by a DSS on 04/27/02 at 2030. Operable - fully meets performance requirements. No further action

required A copy of the approved version is attached below.

**Prescreen Comments:** Operability Status: Operable \Operability Basis: AFW system has passed all required testing and is operable. An OD has been requested by plant staff. \\SCREENER COMMENTS: Temp info tags will be placed at appropriate locations on the control boards to address this problem This is only a short term fix. \Notification to NRC made at 1705 CST.

**Import Memo Field:**

**OPR Completed?:** N

**OLD_ACTION_NUM:**

**sub_tsid:** 0 **original_project_id:** 32

**original_issue_id:** 001415

**Site:** Point Beach

**Cartridge and Frame:**

## ATTACHMENTS AND PARENT/CHILD LINKS

Linked To Child 'ACE000314'

Linked To Child 'CA002592'

Linked To Child 'CA002593'

Linked To Child 'CA002594'

Linked To OTH003541

CA003691: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003692: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003693: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003694: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003695: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003696: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003697: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003698: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003699 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003700. Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003701: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003702 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003703 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003704 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003705: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Linked To CA003982

Linked To CA003983

CA004279: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Linked To CA004388

Linked To OTH004389

OD Part 1 rev 2 for CR 01-3595 approved.pdf (1984402 bytes)

Linked To OTH004510

Principal to CA026222: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Principal to CA026223. Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Principal to CA026224: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Principal to CA026225: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Principal to OTH026285: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

## STATE CHANGE HISTORY

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| Initiate → by JAMES MASTERLARK | Conduct Work 12/20/2001 9.57.31 PM Owner RICHARD FLESSNER | Return → by RICHARD FLESSNER | Assign Work 1/7/2002 5.06 01 PM Owner LARRY PETERSON | Assign → by LARRY PETERSON | Conduct Work 1/18/2002 12.54.54 PM Owner RICHARD FLESSNER | Work Complete → by RICHARD FLESSNER | Review & Approval 2/5/2002 4 15:10 PM Owner LARRY PETERSON |
| Approved → by LARRY PETERSON | Quality Check 2/6/2002 1.23 07 PM Owner JULIE KREIL | Return → by JULIE KREIL | Assign Work 2/14/2002 2:32 56 PM Owner LARRY PETERSON | Update → by MARYBETH ARNOLD | Quality Check 2/19/2002 11:05 11 AM Owner JULIE KREIL | Complete and Close → by MARYBETH ARNOLD | Done 2/19/2002 2 53:33 PM Owner (None) |
| Update → by MARYBETH ARNOLD | Conduct Work 4/27/2002 12:13.36 PM Owner RICHARD FLESSNER | Work Complete → by RICHARD FLESSNER | Review & Approval 5/14/2002 3 39.50 PM Owner LARRY PETERSON | Approved → by RICHARD FLESSNER | Quality Check 5/14/2002 4.57.47 PM Owner PBNP CAP Admin | Complete and Close → by MARYBETH ARNOLD | Done 5/15/2002 1.11:53 PM Owner (None) |

## SECTION 1

**Activity Request Id:** ACE000314

**Activity Type:** Apparent Cause Evaluation

**Submit Date:** 12/4/2001 1:00:00 AM

**Site/Unit:** Point Beach - Common

**Activity Requested:** ROOT CAUSE EVALUATION: Perform a Root Cause Evaluation for this issue in accordance with the Root Cause Evaluation guideline (OEG 001). CARB review of this Root Cause Evaluation is required. \\DESCRIPTION: \While performing an update to the Auxiliary Feed Water (AFW) System model in the PRA, a procedural shortcoming was identified in AOP 5B with regards to the availability of the minimum recirculation valve with the loss of instrument air. This issue was documented in CR 01-2278 with a recommendation to upgrade the procedure. Upon further review of this issue with PRA engineers, Operations, and Design Engineering, it was discovered that this issue has further reaching affects as documented below.\\Instrument air (IA) can be lost primarily by two failure mechanisms. The first, and most likely, is a loss of off-site power where the IA and Service Air (SA) compressors are stripped from the bus and not automatically re-loaded. The second less likely scenario is a random loss of the instrument air system due to equipment failure without potential for short term recovery. When IA is lost, the minimum flow recirculation valves for AFW fail closed.\\During these two transients, the AFW pumps will start injecting into the steam generators Early in the EOPs, the operator is directed to control flow to the steam generators to maintain desired level. This may include shutting off flow to one or both steam generators if level is above the desired band. If flow from any auxiliary feed pump is reduced too low (as would occur if the auxiliary feed regulating valves are closed) without functional recirculation valves, the pump will fail in a very short period of time. This common mode of failure (common loss of instrument air and common response to high steam generator level) could result in simultaneous failure of all AFW pumps.\\PRA has estimated the risk associated with this issue The total risk increase due to both the loss of off-site power and loss of instrument air contribution is approximately a factor of 4 times higher than our assumed base risk with an overall increase in the area of 2E-4 CDF per year (base risk is around 5E-5 CDF per year).\\WHY DID EVENT/ISSUE OCCUR? Current design of plant - deficiency not previously recognized.\\RECOMMENDATIONS: 1) Engineering needs to further evaluate and determine long term corrective action.\\2) PRA needs to evaluate and provide guidance for short term Maintenance Rule risk monitoring until new model is implemented.

| | | | |
|---|---|---|---|
| CATPR: | N | Initiator: | PETERSON, LARRY |
| Initiator Department: | EX Engineering Processes PB | Responsible Group Code: | (None) |
| Responsible Department: | Engineering | Activity Supervisor: | LARRY PETERSON |
| Activity Performer: | RICHARD FLESSNER | | |

SECTION 2

| | | | |
|---|---|---|---|
| Priority: | 2 | Due Date: | 5/27/2002 |
| Mode Change Restraint: | (None) | Management Exception From PI?: | N |
| QA/Nuclear Oversight?: | N | Licensing Review?: | N |
| NRC Commitment?: | N | NRC Commitment Date: | |

SECTION 3

**Apparent Cause Evaluation:** (Note: This RCE required revision because additional information and insight were developed during preparations for the NRC regulatory conference held on this issue.)

Purpose:
The purpose of this investigation is to determine the root and contributing causes of why the emergency operating procedural inadequacies existed that contributed to the increased core damage frequency (CDF) for the Auxiliary Feedwater System during a loss of instrument air event, and why these inadequacies where not identified previously.

Event Synopsis:
During a review of the AFW PRA model in June 2001, it was discovered that the AFW recirculation valves were not modeled. Subsequent discussions disclosed that under a loss of instrument air condition (IA), operators might close the AFW discharge valves to stop AFW flow. Because the recirculation valves fail close on loss of IA, these actions could deadhead the AFW pumps and result in pump damage. Initially the procedural concern was directed at AOP-5B, but it was later realized that the AOP was not the only concern. Operator actions could be taken earlier in an accident scenario to control or stop AFW flow, as directed by steps in EOP-0.1, prior to taking manual actions directed by AOP-5B. PRA modeling of the AFW system continued and on 11/26/01 a factor of 2.3 risk increase in CDF was identified. As discussions with site personnel continued, additional initiating events were identified and on 11/28/01 a revised PRA model was run that changed the risk estimate to a factor of 4 to 5 increase in CDF. Condition report CR 01-3595 was initiated at 1445 on 11/29/01 and an NRC event notification was made at 1705 the same day.

Conclusions:
The investigation found that the EOP validation process is the barrier that failed, causing the weakness in EOP-0.1. The EOP validation process failed because it did not evaluate the interaction among design, procedures, and human error timeline analysis. It was only from this integrated perspective that a loss of instrument air causing the recirculation valves to fail closed, combined with a possibility that an operator could close the discharge valve on an AFW pump, and the timing of this action prior to implementation of the abnormal procedure for loss of instrument air (AOP-5B) could the potential be seen to damage multiple AFW pumps. The combination of FMEA, timeline studies, and human error analysis is a recently implemented practice in the industry unique to PRA. Without the use of these combined analyses, it was not reasonable that previous evaluations would have identified this vulnerability.

Nuclear Safety Significance:
Preliminary PRA results show that the vulnerability described in this report, prior to the procedural changes, was potentially risk significant. Although the initiating event frequencies are low to moderate, when an unrecoverable IA scenario is considered risk becomes significant due to the consequences of a total loss of all AFW pumps requiring feed and bleed without the pressurizer PORVs. The risk results are highly dependant upon human interactions. PBNP operators are trained on AFW system operations and have experience with degraded IA scenarios. Because of this training and experience, it is reasonable to assume that operators would have successfully handled this combination of conditions in the unlikely event that it would have occurred.

Root Cause:
The root cause of the EOP procedural weaknesses was the failure of the original EOP validation process barrier to identify that specific operator directions were needed to ensure the operator would properly control or stop AFW flow under a loss of instrument air condition. This barrier failed because the analytical tools needed to identify this vulnerability did not exist at that time. This resulted in a misalignment between plant design and procedural guidance.

Significant contributing causes to this condition continuing to exist were:

-The original PRA model fault trees evaluated system performance primarily on functions described in design documents and only considered operator actions taken to mitigate a failure

-Previous evaluations focused on delivery of the minimum required AFW flow for providing decay heat removal

Corrective Action Synopsis:
-EOP-0, EOP 0.1 and ECA-0.0 revised to address AFW control under loss of IA
-Back-up pneumatic supply added to AFW recirculation valves
-AOP-5B revised to incorporate back-up pneumatic supply for recirculation valves
-EOP validation process revised to include PRA
-Simulator enhanced to model potential for AFW pump failure on loss of IA
-Evaluated EOP steps to ensure successful implementation on loss of IA
-Completed detailed evaluation of PRA model for the four top risk-significant systems
-Validated PRA assumptions on next two risk-significant systems (these six systems comprise 80% of CDF risk)
-Continuing detailed evaluations of PRA model for other risk-significant systems
-Enhancing CDF risk reduction by incorporating PRA human error reduction methods into operator training and operating procedures

**Activity Completed:**

1/18/2002 12:52PM - LARRY PETERSON:
Due date extended as requested and approved by F. Cayia in prior update. Retruned to R. flessner for completion.

1/18/2002 12:54PM - LARRY PETERSON:
Reassigned to R. Flessner for completion following extension.

2/5/2002 4:15PM - RICHARD FLESSNER:
RCE report completed on 2/5/02 and forwarded to LJP for approval. Actions items generated and report attached.

2/6/2002 1:23PM - LARRY PETERSON:
RCE reviewed and approved. Routed for CA Mgr and CARB approval.

2/14/2002 2:32PM - JULIE KREIL:
RCE copy not received by CAP Manager as of 2/14/2002, per RCE Coordinator

2/19/2002 11:05AM - MARYBETH ARNOLD:
RCE electronic copy had been attached to this record; however, it is believed that Windows 2000 migration may have caused a problem with the opening of this document. Document unable to be opened. Assessment group is contacting the RCE evaluator for an additional electronic copy. Hard copy is presently in the hands of the CAP manager awaiting other groups reviews/comments prior to his approval signature.

2/19/2002 2.53PM - MARYBETH ARNOLD:

A copy of the electronic RCE is attached below. Actions were created by the evaluator as followon to this RCE. See CA003691 through CA003705 for these items. OTH003541 tracks the presentation and acceptance by CARB. This item is considered CLOSED.

4/27/2002 12:13PM - MARYBETH ARNOLD:
This item is being re-opened per the request of the Activity Performer and Plant Management. A revision to the Root Cause Evaluation is being prepared.

5/14/2002 3:39:50 PM - RICHARD FLESSNER:
The revised report was approved on 5/14/02 by the Engineering Director and sent to the CAP Manager.

5/14/2002 4:57.47 PM - RICHARD FLESSNER
Larry Peterson and Lori Armstrong approved Rev. 1 of the RCE on 5/14/02. Approval to close granted by Larry Peterson. RCE forwarded to CAP Manager for final approval.

5/15/2002 1:11:53 PM - MARYBETH ARNOLD.
RCE 01-069, Revision 1 was approved by CAP on 05/15/02. Actions were created from the Revision 0 RCE.

5/15/2002 1:13:07 PM - MARYBETH ARNOLD:
OTH003541 tracks the presentation of Revision 1 to the CARB. CLOSED.

6/14/2002 8:08:57 AM - JULIE KREIL:
At 6/04/2002 CARB meeting, CARB accepted Revision 1 of this RCE with no further actions or editorial changes to be made (reference NPM 2002-0292).

## SECTION 4

| | | | |
|---|---|---|---|
| QA Supervisor: | (None) | Licensing Supervisor: | (None) |
| ACE Event Description Grade: | 0 | ACE Extent of Condition Grade: | 0 |
| ACE Corrective Actions Grade: | 0 | ACE CATPR Grade: | 0 |
| ACE Apparent Cause Grade: | 0 | | |

## SECTION 5

| | | | |
|---|---|---|---|
| Project: | Apparent Cause Evaluation (ACE) | | |
| State: | Done | Active/Inactive: | Inactive |
| Owner: | (None) | AR Type: | Daughter |
| Submitter: | JAMES MASTERLARK | Assigned Date: | 1/18/2002 |
| Last Modified Date: | 9/3/2002 7:12:54 PM | Last Modifier: | RICHARD FLESSNER |
| Last State Change Date: | 5/15/2002 1:11:53 PM | Last State Changer: | MARYBETH ARNOLD |
| Close Date: | 5/15/2002 1:11:53 PM | | |
| One Line Description: | Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW | | |
| NUTRK ID: | CR 01-3595 | | |
| Child Number: | 1 | | |
| References: | CR 01-2278<br>RCE 01-069<br>GOOD CATCH | | |

NPM 2002-0292

**Update:** (12/05/01 LJP) Received Action into Group. SEP\Responsible Person. RAF:RICHARD FLESSNER Due Date: 01/03/2002\\(20011205 XX4869 LJP) Set Work Priority to 2. Assigned priority 2 based on level B CR requiring RCE.\\(12/12/01 RAF) Changed the Due Date from: 01/03/2002 to 01/10/2002\The due date for this RCE was established in the Charter by the Plant Manager as 1/10/2002 based on the normal 30 day period plus an additional week to account for the holiday season. Approval of the Charter by Fred Cayia on 12/4/2001 satisfies the requirement for a SVP Direct Report approval of a due date different from the NP. A copy of the Charter has been forwarded to the PLA for documentation. (12/05/01 LJP) RECEIVED ACTION INTO GROUP. SEP\RESPONSIBLE PERSON: RAF:RICHARD FLESSNER DUE DATE: 01/03/2002\\(20011205 XX4869 LJP) SET WORK PRIORITY TO 2. ASSIGNED PRIORITY 2 BASED ON LEVEL B CR REQUIRING RCE.\\(12/12/01 RAF) CHANGED THE DUE DATE FROM. 01/03/2002 TO 01/10/2002\THE DUE DATE FOR THIS RCE WAS ESTABLISHED IN THE CHARTER BY THE PLANT MANAGER AS 1/10/2002 BASED ON THE NORMAL 30 DAY PERIOD PLUS AN ADDITIONAL WEEK TO ACCOUNT FOR THE HOLIDAY SEASON. APPROVAL OF THE CHARTER BY FRED CAYIA ON 12/4/2001 SATISFIES THE REQUIREMENT FOR A SVP DIRECT REPORT APPROVAL OF A DUE DATE DIFFERENT FROM THE NP. A COPY OF THE CHARTER HAS BEEN FORWARDED TO THE PLA FOR DOCUMENTATION.

4/27/2002: This RCE required revision because of additional information and insight developed during prepartions for the NRC regulaory conference held on this issue. Action item OTH004389 was created to track completion of the revised RCE.

**Import Memo Field:** An extension request was granted by Fred Cayia on 1/24/02 to a revised due date of 2/7/02. A draft report was provided to key personnel on 1/22/02. Comments received from Operations/Training on 1/23 and L. Peterson on 1/24 require additional time for resolution.
A new due date of 5/27/2002 has been set based on 30 days from the date it was reopened, to revise the RCE. (RAF)

**CAP Admin:** PBNP CAP Admin **Site:** Point Beach

**OLD_ACTION_NUM:**

**Cartridge and Frame:**

**NOTES/COMMENTS**

Note created during 'Return' transition by **RICHARD FLESSNER** (1/7/2002 5:06:01 PM)
A due date extension to 1/24/02 has been approved verbally by Fred Cayia and Rick Mende in response to the following e-mail (dated 1/7/02):

Fred,

The Root Cause Team met for several hours today to review progress on the RCE report. We were at the stage of reviewing event and causal factor information for events in the 1979 to 1990 time frame. The events were primarily due to modifications and responses to industry operating experience (NRC and INPO). It became evident that several areas need additional research in order to understand what the causal factors involved were. These areas are FSAR content over time, IST program changes, IPE program development, training program influences, and DBD changes.

I have concluded that the committed date for the RCE completion of 1/10/02 cannot be met. There are 2 factors influencing this conclusion: 1) I underestimated the impact the holiday season would have on the Team's momentum, and 2) the new areas of research identified in today's meeting will take additional time to develop and evaluate.

The remaining scope of work consists of evaluating these new areas of research, developing E&CF information for 1990-2001, identification of appropriate causal factors, determination of root and contributing causes, and development and negotiation of corrective actions. The Team feels that this work can be accomplished by 1/24/02. The Team wants to do a thorough job on this RCE because of its safety significance and the level of scrutiny that it will receive by the interested parties. I am therefore requesting a 2 week extension of the committed due date to 1/24/02. I have discussed this extension with Rick Mende and have his concurrence.

Respectfully,

Rich Flessner
Team Leader

Note created during 'Return' transition by JULIE KREIL (2/14/2002 2 32:56 PM)
Copy of RCE not received by CAP Manager as of 2/14/2002, per RCE Coordinator.

ATTACHMENTS AND PARENT/CHILD LINKS

Linked From Parent 'CAP001415'

OTH003541: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003691: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003692: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

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CA003703 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003704: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA003705: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

CA004388 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

OTH004389: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

RCE 01-069 Rev 1 (419840 bytes)

Linked from CA026222 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Linked from CA026223 Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Linked from CA026224: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Linked from CA026225: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

2137

NMC
Committed to Nuclear Excellence

# INTERNAL CORRESPONDENCE

NPM 2002-0252

**To:** M. E. Warner

**From:** D. A. Hettick

**Date:** May 15, 2002

**Subject:** ROOT CAUSE EVALUATION 01-069, Revision 1

**Copy To:**

| | | | |
|---|---|---|---|
| S. J. Nikolai | D. D. Schoon | A. J. Cayia | T. Coutu - KNPP |
| J. Purcell - KNPP | S. J. Thomas | L. J. Armstrong | V. A. Kaminskas |
| J. R. Anderson | G. A: Corell | T. Taylor | D. Weaver |
| R. Repshas - KNPP | T. Sullivan (P458) | J. J. Walsh | K. Peveler |
| R. Nicolai - KNPP | B. Day | M. McCarthy | S F. Putman – KNPP |
| R. Milner | M. B. Arnold | T. L. Zifko | R. Wood |
| L. Peterson | C. Krause | J. P. Schroeder | T. Staskal |
| R. Flessner | L. A. Schofield/JOSRC | | T. Y Fessler/OSRC |
| File | | | |

Attached is Root Cause Evaluation (RCE) 01-069, Revision 1, for your review. This RCE is an evaluation of Increased CDF in AFW PRA Model Due to Procedural Inadequacies Related to Loss of Instrument Air. The corrective actions associated with this RCE will be tracked under CR 01-3595.

If you have any questions or would like to discuss the report, please call me at Extension 6498.

Approved: FOR DAH
D. A. Hettick

tlz

Attachment

NMC
Committed to Nuclear Excellence

# Point Beach Nuclear Plant

# Increased CDF in AFW PRA Model Due to Procedural Inadequacies Related to Loss of Instrument Air

## RCE 01-069
Revision 1

## (CR 01-3595)

## Event Date: November 29, 2001

## Report Date: May 14, 2002

Principal Investigators:

R. Flessner – Team Leader
C. Krause
J. P. Schroeder
T. Staskal
R. Wood

Approved By:

Group Lead – Larry Peterson — Date 5/14/02

Issue Manager – Lori Armstrong — Date 5/14/02

CAP Manager – Dennis Hettick — Date 5/15/02

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# I. Executive Summary

(Note: This RCE required revision because additional information and insight were developed during preparations for the NRC regulatory conference held on this issue.)

**Purpose:**

The purpose of this investigation is to determine the root and contributing causes of why the emergency operating procedural inadequacies existed that contributed to the increased core damage frequency (CDF) for the Auxiliary Feedwater System during a loss of instrument air event, and why these inadequacies where not identified previously.

**Event Synopsis:**

During a review of the AFW PRA model in June 2001, it was discovered that the AFW recirculation valves were not modeled. Subsequent discussions disclosed that under a loss of instrument air condition (IA), operators might close the AFW discharge valves to stop AFW flow. Because the recirculation valves fail close on loss of IA, these actions could deadhead the AFW pumps and result in pump damage. Initially the procedural concern was directed at AOP-5B, but it was later realized that the AOP was not the only concern. Operator actions could be taken earlier in an accident scenario to control or stop AFW flow, as directed by steps in EOP-0.1, prior to taking manual actions directed by AOP-5B. PRA modeling of the AFW system continued and on 11/26/01 a factor of 2.3 risk increase in CDF was identified. As discussions with site personnel continued, additional initiating events were identified and on 11/28/01 a revised PRA model was run that changed the risk estimate to a factor of 4 to 5 increase in CDF. Condition report CR 01-3595 was initiated at 1445 on 11/29/01 and an NRC event notification was made at 1705 the same day.

**Conclusions:**

The investigation found that the EOP validation process is the barrier that failed, causing the weakness in EOP-0.1. The EOP validation process failed because it did not evaluate the interaction among design, procedures, and human error timeline analysis. It was only from this integrated perspective that a loss of instrument air causing the recirculation valves to fail closed, combined with a possibility that an operator could close the discharge valve on an AFW pump, and the timing of this action prior to implementation of the abnormal procedure for loss of instrument air (AOP-5B) could the potential be seen to damage multiple AFW pumps. The combination of FMEA, timeline studies, and human error analysis is a recently implemented practice in the industry unique to PRA. Without the use of these combined analyses, it was not reasonable that previous evaluations would have identified this vulnerability.

**Nuclear Safety Significance:**

Preliminary PRA results show that the vulnerability described in this report, prior to the procedural changes, was potentially risk significant. Although the initiating event frequencies are low to moderate, when an unrecoverable IA scenario is considered risk becomes significant due to the consequences of a total loss of all AFW pumps requiring feed and bleed without the pressurizer PORVs. The risk results are highly dependant upon human interactions. PBNP operators are trained on AFW system operations and have experience with degraded IA scenarios. Because of this training and experience, it is reasonable to assume that operators would have successfully handled this combination of conditions in the unlikely event that it would have occurred.

**Root Cause:**

The root cause of the EOP procedural weaknesses was the failure of the original EOP validation process barrier to identify that specific operator directions were needed to ensure the operator would properly control or stop AFW flow under a loss of instrument air condition. This barrier failed because the analytical tools needed to identify this vulnerability did not exist at that time. This resulted in a misalignment between plant design and procedural guidance.

Significant contributing causes to this condition continuing to exist were:

- The original PRA model fault trees evaluated system performance primarily on functions described in design documents and only considered operator actions taken to mitigate a failure

- Previous evaluations focused on delivery of the minimum required AFW flow for providing decay heat removal

**Corrective Action Synopsis:**

- EOP-0, EOP 0.1 and ECA-0.0 revised to address AFW control under loss of IA
- Back-up pneumatic supply added to AFW recirculation valves
- AOP-5B revised to incorporate back-up pneumatic supply for recirculation valves
- EOP validation process revised to include PRA
- Simulator enhanced to model potential for AFW pump failure on loss of IA
- Evaluated EOP steps to ensure successful implementation on loss of IA
- Completed detailed evaluation of PRA model for the four top risk-significant systems
- Validated PRA assumptions on next two risk-significant systems (these six systems comprise 80% of CDF risk)
- Continuing detailed evaluations of PRA model for other risk-significant systems
- Enhancing CDF risk reduction by incorporating PRA human error reduction methods into operator training and operating procedures

## II. Event Narrative

In June, 2001 the PRA group was reviewing and revising the AFW portion of the PRA model. During this review it was discovered that the minimum flow recirculation valves were not modeled within the PRA. Therefore, a failure modes and effects analysis was performed to determine potential failure modes. A discussion was held with past operations personnel about how the system was operated within the AOPs and EOPs. It was then determined that upon a complete loss of instrument air, the operators may use the EOPs and stop AFW flow by closing the discharge MOV or the flow control valve. However, since the recirculation valve fails closed on a loss of instrument air, the AFW pump would not have adequate recirculation flow. This issue was discussed with a design engineer who informed the PRA group that the AFW pumps could be damaged in a short period of time without adequate recirculation flow.

This issue was then discussed with Operations Training personnel who reviewed the EOPs and discussed what operator actions would be. The operator actions were also confirmed with an Operations crew. The actions assumed were that upon a complete loss of instrument air, entry would be made into EOP-0, Reactor Trip or Safety Injection, and then into EOP-0.1, Reactor Trip Response. Steps in these procedures would ensure that at least one AFW pump was available. In EOP-0.1, if S/G level is high the operator is directed to STOP flow. If flow were stopped, by closing the discharge valve, the AFW pump would fail due to lack of minimum flow caused by the recirculation valve failing closed. The potential exists that this same evolution could be repeated on additional AFW pumps. Since this is a dual unit event with both units in a similar configuration, the same problem could also happen on the second unit.

It was noted that AOP-5B, Loss of Instrument Air, had a specific note to gag open the recirculation valves, but the information was located well into the procedure and timing showed that it would not be adequate to preclude closing the discharge valves. PRA personnel understood that this failure mode had the potential to be risk significant even though the actual significance was not known since the PRA model development was not yet completed. PRA personnel initiated CR 01-2278 on 7/6/01 to document this problem and identify potential corrective actions to place steps addressing the need to gag the recirculation valves open earlier in the sequence of AOP-5B. It was assumed that the AOP was sufficient to address the concern, but the timing of the action could be improved to ensure that the action would be successful.

An action item was created on 7/10/01 for the Operations Procedure group with a recommendation to move the step (AOP-5B step 24) to a more prominent position in the procedure and consider using a foldout page. The action item priority was set at 4 and the due date was established as 8/21/01. Discussions were held between PRA and Operations personnel and it was expected that a PRA group evaluation to determine the significance of the issue would be completed by 8/20/01. Initial Operations review of AOP-5B indicated that the procedure was laid out in a priority to restore instrument air, which is the correct response for that procedure. The evaluation of the risk significance of the as found configuration of the procedure is dependent on quantifying the entire PRA

model. This was not completed until October, due to the complexity of developing a complete two-unit model. The original model used a single unit and simplified common systems. The PRA group informed Operations on 8/20/01 that the evaluation was not completed as expected and additional time was required to evaluate the actual significance and the type of action that should be done. At that time modifications and procedural changes were being considered.

The PRA group completed some preliminary modeling on 10/19/01 that indicated the potential for a high risk was involved and informed Operations that the AFW pump recirculation valves should be procedurally addressed. Based on further discussion, it was decided that a change to the Alarm Response Procedure for instrument air low header pressure (ARP C01 A 1-9) could address the concern. The PRA group was to submit a procedure feedback form for the desired change. The original action item was closed on 11/14/01 and a new action item was created on 11/14/01 to track the changes to the ARP and assigned to Operations. Operations discussed the request with PRA personnel and gave the new action item a priority of 3 with a due date of 12/26/01, based on expected completion of the PRA model and Safety Monitor update in December. During that discussion some concerns were raised by Operations about the adequacy of procedural changes to address the issue. Specifically, the concern was that the ARP may not be the most effective way of protecting the AFW pumps during high activity in the Control Room, i.e., the loss of instrument air may not take priority and the ARP may not be referred to.

Additional discussions took place between Operations, PRA and a design engineer concerning the appropriate corrective actions and what risk might be involved if the procedural remedy was not completed or was inadequate. On Monday, 11/26/01, the PRA modeling adjustments were completed and a factor of 2.3 risk increase in Core Damage Frequency (CDF) was identified, which is considered high. Additional discussions took place between Engineering and Operations to determine further actions that may be appropriate.

A meeting between Operations and Engineering was held at 1300 on Wednesday, 11/28/01, to discuss significance and actions. During the discussion it was discovered that the loss of instrument air was more than just a random loss, a loss of offsite power (LOOP) or other events could also initiate the event. A re-evaluation of risk including the LOOP event resulted in an estimated factor of risk increase of 4 to 5 in CDF. Operability was also discussed. It was concluded that there was no operability concern because no equipment degradation, failure, or non-conformance had been identified. Regardless, the level of concern was great enough that further prompt actions were felt to be justified. The Design Engineering Manager briefed the Operations Manager on the situation later that afternoon. The Operations Manager also updated the Plant Manager on the situation.

On Thursday morning, 11/29/01, the Operations Manager briefed the NRC Resident Inspectors on the issue and informed them that we were evaluating this apparent vulnerability and the risk significance. Operations decided that use of temporary

information tags and briefing of all watch standers would be an important step to reduce risk; an evaluation of possible procedure enhancements was also initiated. At 1000, PRA personnel briefed the STA and Shift Manager on the issue and discussed potential wording for temporary placards to be placed on the control panels.

At 1100, PRA personnel discussed potential reportability concerns with Licensing. It was not clear if this issue was reportable because it involved a procedure and was not an equipment issue – additional discussions were needed. At 1130, PRA personnel briefed the NRC Resident Inspector on the issues and answered questions regarding risk impact and human error probabilities. During the afternoon, Licensing and Engineering personnel evaluated the reportability aspect further. It was concluded that the conservative decision would be to report the issue, even though a specific reporting criteria could not be identified. At 1445, PRA personnel initiated Condition Report 01-3595 and brought it to the Work Control Center for SRO screening at 1538. The Operations Manager took part in discussions involving operability and the need for an Operability Determination (OD). Since the issue identified in CR 01-3595 did not affect equipment, the decision was made that an OD was not required; however, the details of those discussions were not captured in either the CR or the screening comments. The SRO screening was completed at 1553 with the event determined to be reportable as a procedural inadequacy and not requiring an OD.

At 1520, the oncoming crew was briefed on the concerns of this potential event and temporary information tags were placed adjacent to the controls for 1/2P-29 and P-38 A/B that provided a reminder of the minimum flow requirements for each AFW pump.

At 1700, the Operations Manager provided the Plant Manager with an update on the issue. At 1705, Event Notification EN 38525 was made to the NRC via the ENS phone. (See Section V. for details)

On Friday morning, 11/30/01, the Licensing Manager received a phone call from the acting NRC-NRR Project Manager for Point Beach, concerning confusion over the event notification. A return conference call was made with Engineering personnel to address NRR questions. A decision was made to provide a supplemental event notification providing additional details. The Operations Manager had additional conversation with the NRC Resident Inspectors and concluded that to formally document the operability of the AFW system, an OD would be initiated to capture the discussions held during the previous 24 hours. Operations requested that Engineering provide an OD and informed the Shift Manager that it was expected to be completed that afternoon. At Noon, the Operations Manager met again with the NRC Resident Inspectors and their supervisor to address NRC concerns regarding AFW operability prior to 11/29/01 and in its current configuration. The Plant Manager and Operations Manager had a conference call with NRC Region III to discuss operability of the AFW system.

At 1400, a simulator scenario was run to obtain information on plant response to a loss of offsite power coincident with a rapid loss of instrument air pressure. Additional scenarios were run on 11/30 and 12/1.

At 1645, temporary procedure changes were completed for EOP-0 and EOP-0.1 to reflect the guidance provided earlier to operators via the temporary information tags.

At 1700, the Plant Manager was informed that a 5 person NRC incident investigation team would arrive on 12/3/01. At 1746, a supplemental event notification was made to the NRC to clarify the discussion on the potential for an AFW system failure as described in the original event notification (EN 38525).

At 1755, Engineering completed Revision 0 of the OD that concluded that the AFW system was Operable but Non-Conforming. This was based in part on a statement in the FSAR that "each pump has an AOV controlled recirculation line back to the condensate storage tanks to ensure minimum flow to dissipate pump heat." The compensatory actions already in effect were listed in the OD as required actions. The Plant Manager and Operations Manager reviewed the OD content and then briefed the Senior NRC Resident Inspector. The OD was then brought to the Control Room and accepted at 2015. On Friday evening, just-in-time (JIT) training was provided to the swing shift crew on the simulator on this event; JIT was also provided to the mid-shift crew on the simulator prior to assuming the watch.

On Saturday, 12/1/01, at 0720 JIT was provided to the oncoming dayshift crew on the simulator prior to assuming the watch. A staff meeting was held from 0930 to 1200 to prepare for the NRC inspection team. A revised OD was prepared at 1500 to expand the discussion on AFW pump motor duty cycles. The Control Room accepted it at 1515.

On Monday, 12/3/01, CR 01-3595 was screened and assigned to Engineering to perform an apparent cause evaluation. Another meeting was held from 1000 to 1200 in preparation for the NRC inspection team. At that meeting it was decided that a root cause evaluation would be a more appropriate response to this event. The Plant Manager approved the RCE Charter on 12/4/01.

The NRC Inspection Team arrived onsite on 12/3/01 and conducted a technical debrief on 12/7/01. A preliminary exit meeting was held on 12/13/01.

An expert on Human Error Probabilities was brought onsite on 12/4/01 to help quantify the risks associated with the procedural weaknesses that were identified. His evaluation estimated that there was about a 50% chance that the operator would shut the discharge valve and fail to recognize that the minimum flow recirculation valve did not open when flow was stopped as S/G levels rose above 65% on the narrow range.

On 12/4/01, CR 01-3633 was initiated by Engineering on the ability of the Motor Driven Auxiliary Feedwater Pumps (MDAFWP) to respond to an Appendix R fire coincident with a loss of offsite power and instrument air because of a lack of documentation related to the potential for closure of the recirculation valves due to loss of instrument air. CR 01-3648 was initiated by Engineering on 12/5/01 on the same issue when four specific fire zones were identified as having the potential to cause an AFW pump auto-start

coincident with discharge and recirculation valve closure, resulting in pump damage. An OD was completed for CR 01-3468 on 12/7/01 that concluded the MDAFW Pumps were Operable but Non-Conforming, with the required compensatory measures of performing hourly fire rounds in the specified fire zones. An event notification on this issue was made at 1926 on 12/05/01 (EN #38541)

Permanent revisions to EOP-0 and EOP-0.1 were implemented on 12/14/01. As PRA reviews continued, it was recognized that the closure of the AFW recirc valves could occur after an operator had already taken action to put the pumps in the recirculation mode. Additional changes were made to those procedures and ECA-0.0 on 12/20/01 to address this concern. As additional information becomes available, procedure improvements are often implemented to continually improve their quality.

## III. Extent of Condition Assessment

The root cause of this event is attributed to a weakness in the original EOP validation process where the effects of a loss of instrument air were not adequately evaluated. This occurred because the validation process did not evaluate the interaction between design, procedures and human error timeline analysis. It was only from this perspective that a loss of IA causing the recirc valves to fail closed combined with a possibility that an operator could close a discharge valve on an AFW pump and the timing of this action prior to implementation of the abnormal procedure for loss of IA (AOP-5B) could the potential be seen to damage multiple AFW pumps. This validation process was believed to be consistent with industry practices.

Because of this event, the previously held belief that AOP-5B, Loss of Instrument Air, adequately directed the required operator actions was found to be faulty because actions were required while in an EOP, prior to performing AOP subordinate actions. This event identified a specific concern with AFW control, but there may be other operator actions that are unique to a loss of instrument air condition that were not adequately considered in the EOPs. A review of EOP steps was performed to ensure that the stated operator actions could be performed under a loss of instrument air condition.

The original PRA model fault trees evaluated system performance primarily on functions described in design documents and did not adequately consider human actions. The current PRA model review uses a methodology that integrates system performance with potential human actions to obtain a spectrum of plant responses. This more rigorous approach should identify any other assumptions used in risk-significant systems that have not adequately considered human actions and any risk-significant vulnerabilities in the emergency operating procedures. The four highest risk-significant systems have had a detailed review of the PRA model completed already. The assumptions for operator actions for the next two highest risk-significant systems have also been validated. These six systems comprise 80% of the CDF risk. The detailed review of the PRA model for the remaining risk-significant systems is continuing.

The lack of integration of human error reduction methods into operations training and emergency procedure development processes may allow situations to exist where PRA risk reduction has not been optimized. Procedures and training associated with high-risk human error events will be reviewed against human error reduction methods to ensure that reasonable risk reduction has been achieved.

## IV. Nuclear Safety Significance

Any complete loss of IA for a significant time is expected to result in a reactor trip and an AFW start signal due to a loss of normal feedwater (the normal feed water regulating valves fail closed on loss of air). Under this postulated condition, all components of the AFWS are now and continue to be fully capable of performing their design functions supporting automatic starting and supplying sufficient flow to the steam generators to mitigate any transient or accident by removal of decay heat. It is the continued function of the AFWS, in response to directed operator actions to control AFWS flow, and the lack of specific guidance contained within the EOPs regarding a loss of IA, that is the issue identified in this report.

A PRA assessment of the possible failure modes and effects associated with an IA failure identified a previously unrecognized vulnerability. This failure would have been caused by a combination of a design limitation, a specific sequence of postulated operator actions, and a lack of clear guidance within the EOPs. This combination could result in failure of one or more of the AFW pumps due to aggressive AFW flow reduction (as may be expected in response to a steam generator overfill or RCS over-cooling) after automatic system start and flow had been established. The likelihood of success or failure in the postulated scenario is highly dependent upon plant transient response (which may vary with the nature of the initiating event, initial power levels, etc.) and operator response. Operator response is highly dependent upon prior training, procedural usage, system knowledge and awareness, experience, and other human effectiveness (HE) factors. It should be noted that a control board alarm is provided (Instrument Air Header Pressure Low) to alert the operator to the existence of an initiating condition for this event and that established plant procedures direct the restoration of IA (both Emergency Operating Procedures and Abnormal Operating Procedures), and the manual gagging open of the minimum flow recirculation valves in the event that IA cannot be promptly restored (AOP-5B). PBNP has experienced partial losses of IA, including one event involving the loss of all off-site power and another involving a low IA header pressure alarm following a reactor trip. In each of these cases the operators demonstrated the ability to cope with the loss of IA casualty and recover IA header pressure before it had an adverse affect on plant equipment or response.

Preliminary PRA results show that the vulnerability described in this report, prior to the procedural changes, was potentially risk significant. Although the initiating event frequencies are low to moderate, the unrecoverable IA scenario was risk significant due to the consequences of a total loss of all AFW pumps requiring feed and bleed

without the pressurizer PORVs. The risk results are highly dependant upon human interactions. PBNP operators are trained on AFW system operations and have experience with degraded IA scenarios. Because of this training and experience, it is reasonable to assume that operators would have successfully handled this combination of conditions in the unlikely event that it would have occurred.

Although the AFWS met, and continues to meet, all of its design and licensing requirements, the postulated initiating event of a loss of IA, in conjunction with a misaligned procedure, had the potential to affect redundant trains of the AFWS, a safety related system. Since it could be postulated that the same operator action could have impacted all the AFWS pumps, the result could be the complete loss of the AFWS safety-related function. Accordingly, this event has also been identified as a possible safety system functional failure (SSFF).

## V. Report to External Agencies

Condition Report 01-3595 was initially brought to the PBNP Work Control Center for an SRO screening at 1538 on November 29, 2001. During this screening, a determination was made that this event should conservatively be reported to the NRC in accordance with 10 CFR 50.72(b)(3)(v) as a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:...(D) Mitigate the consequences of an accident. This is an eight-hour non-emergency notification. During the discussion of reportability it was noted that 10 CFR 50.72 Paragraph (b)(3)(vi) clarifies paragraph (b)(3)(v) by noting that, "Events covered in paragraph (b)(3)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies." The last of these items appeared as though it may be applicable in this situation. The following elements also entered into the notification determination:

- NUREG-1022 notes that the level of judgment for reporting an event is a reasonable expectation that the event or condition could lead to preventing fulfillment of a safety function. The intent of these criteria is to capture those events regardless of whether there was an actual demand.

- Example (20) in NUREG-1022 Page 64 directs that system interactions that are found as a result of ongoing routine activities may be reportable.

- When in doubt concerning issues of reportability, it is our policy (consistent with the directions in NUREG-1022) to make the report.

The NRC notification was made using the Emergency Notification System (ENS) telephone at 1705 on November 29$^{th}$. Event number EN 38525 was assigned to this notification.

On the morning of November 30$^{th}$, as a courtesy, the PBNP acting Project Manager at NRC-NRR was telephoned to advise him of the event notification. He had several

questions that were answered in a follow-up call later in the morning. At 1746 on November 30, 2001, the ENS event notification was supplemented to further clarify the discussion of the specific failures postulated and to reiterate that the potential failure would involve only the AFWS pump recirculation valves.

A Licensee Event Report (LER 266/2001-005-00) was submitted within 60 days of this event as required by 10 CFR 50.73.

## VI. Data Analysis

### *Information & Fact Sources*

Document Review Results

Modifications

- **M-623 / 624 - TDAFP Alternate Bearing Cooling Supply, issued 9/1/79**
  Description: In response to an NRC Evaluation of the AFW system, this MR provided a cooling water supply to the TDAFP bearing coolers that is independent of AC power. The supply is taken from the diesel powered Fire Water system.

  Evaluation: The MR enabled the TDAFP to cope with a SBO. Since the TDAFPs are the only pumps available for decay heat removal during the first hour of the SBO, operation of the pumps at low flows requiring recirculation flow is not probable. This modification was performed prior to the original EOP-0.1 being issued in 1985. Therefore, it is not reasonable that this modification would have identified the EOP procedural vulnerability.

- **IC-274 - Modify Logic To Keep Recirculation Valves Open, issued 2/1/80 (Canceled 8/32/82)**
  Description: Modify the control scheme of the recirculation valves to keep valves normally open. The reason for this change was to provide a path for the first off check valve leakage back to the CST. This change would prevent the leakage from lifting the pump suction relief. The modification was canceled since it was only solving a symptom of the real problem; check valve leakage. The modification still intended to have the recirculation valves fail to the shut position.

  Evaluation: The modification was attempting to resolve symptoms associated with check valve leakage. The modification would not have permitted a continuous recirculation path. This modification was originated and cancelled prior to the original EOP-0.1 being issued in 1985. Therefore, it is not reasonable that reviews associated with this modification would have identified the EOP procedural vulnerability.

- **MR 83-104 - AFW System Discharge MOV Controls, issued 8/1/83**
  Description: The MDAFP discharge valves were modified to provide automatic actuation of the valves similar to the automatic starting logic for the MDAFPs.

  Evaluation: The MR was a response to NUREG-0737 to ensure AF is provided to the S/Gs without operator action. The recirculation valves either failed on loss of air or shut as flow to the S/G increased therefore, these valves were already in compliance with the NUREG. This MR deals with eliminating an operator action and the design limitation of the recirculation valves is not introduced until an operator action is taken (i.e. throttling AF discharge flows). This modification was performed prior to the original symptom-based EOP-0.1 being issued in 1985. Therefore, it is not reasonable that this MR would identify the EOP procedural vulnerability.

- **MR 88-099 - AFW Pump Mini-Recirculation Line Improvements, issued 7/7/88**
  Description: In response to NRC IE Bulletin 88-04 and GL 89-04, the recirculation line flows were increased to prevent pump degradation due to hydraulic instability. The minimum pump flow prior to this MR was 30 gpm. The MR increased this to minimum flow to 70 gpm for the MDAFPs and 100 gpm for the TDAFPs. The MR did not change the operation of the recirculation valves.

  Evaluation: PBNP did a design review of the recirc capacity needed for adequate long-term protection of the AFW pumps. This modification was initiated to increase the recirc flow capacity to the required levels. The modification did not alter the operating modes of the recirc valves. System operating procedures were reviewed to the extent that this design change impacted them. Therefore, this very specific design change and review would not identify the EOP procedural vulnerability.

- **MR 92-091/092/093 - IST Testability of AF Recirculation Line AOVs, issued 6/19/92**
  Description: In order to simplify stroke testing of these AOVs, bypass valves were installed around the control solenoid.

  Evaluation: The MR was small scope focusing only on the need to bypass the solenoid to allow stroke testing of the valve. At this time, the IST Program had already identified the shut position as the safety related position for these valves. The scope of this MR was not an opportunity to identify the issue.

- **MR 97-038*A/B - MDAFP Discharge Pressure Control Valve Backup Nitrogen Supply and Cable Separation, issued 4/15/97**
  Description: The MR prevented redundant failures of the AOVs (common electrical fault) and pump runout due to loss of IA (Ref. LER 97-014-00). MR 97-038*B provided physical separation for electrical cables associated with the discharge pressure control valves (AF-4012 and AF-4019) and their associated control components. MR 97-038*A installed nitrogen bottles as a backup pneumatic supply. The design description for MR 97-038*B states that one of the functions of the discharge AOVs is to allow enough flow to the S/Gs to cool the associated pump

during a scenario when pump recirculation is required and the associated recirculation valve fails closed.

Evaluation: The intent of the MR was to prevent pump runout due to a failed open discharge AOV as a result of a loss of instrument air and low S/G pressures. It appears the focus of the MR was to ensure control capability of the discharge pressure control valves. The MR does recognize that the discharge AOVs are needed to provide pump cooling flow if the recirculation valves fail shut. This appears to support the idea that the flow to the S/G is the safety related function and failure of the recirculation valves is acceptable. System operating procedures were reviewed for the impact of this design change. Since the recirc valves were not being modified, it was not reasonable to review procedures associated with those valves. The failure modes and effects analysis of the system performed on this modification did not consider failures caused by operator actions. The ability to throttle the pump discharge flow during a loss of instrument air provides another opportunity (in addition to throttling the MOV) for operator action to cause pump damage.

Procedures

- **AOP-5B, Loss of Instrument Air:** This AOP was first issued on 5/2/86. The procedure contained an "immediate action – manual" step (step 6.0) emphasizing the understanding that AOVs may not function depending on IA header pressure and referred the operator to Appendix A for individual system information. Section R of Appendix A was for Auxiliary Feed, and listed the AFW pump recirculation valves as failing shut with a corresponding note on manual gag override. The additional information in that section included monitoring of AFW pumps for sufficient flow to prevent overheating due to no "minirecirc", and to use the manual gag on the "minirecirc" valve to provide maximum recirculation unless continuous feed was verified through each AFW pump. The procedure content remained essentially the same until Revision 11 was issued on 9/26/97, which moved time critical actions from the appendices into the main body of the procedure. At that time a specific step (step 21) was added for control of AFW flow. A note was placed before that step informing the operator "the manual gag on each AFW pump mini-recirc valve must be used to provide minimum recirc flow if continuous flow through the pump can NOT be verified." The current procedure content is equivalent.

  Evaluation: The AOP contained sufficient information identifying the correct failure mode of the AFW pump recirculation valves on loss of IA, the required manual actions, the concern with pump overheating, and the need to monitor pump flow. The content of the note that directed the operator to continuously monitor pump flow and use the manual gag if flow could not be verified, met the requirements of OM 4.3.1 for note content. OM 4.3.1 allows notes to advise on actions to be taken in the event of changing plant conditions (see discussion on OM 4.3.1 below).

- **EOP-0.1, Reactor Trip Response:** Emergency Operating Procedures, specifically EOP-0.1, is the PBNP procedure that would be used in the event of concern; EOP-0.1

is based on a WOG ERG. Neither EOP-0.1 nor the WOG ERG has ever addressed the function of the AFW mini recirc flow valves. EOP-0.1, in one step (step 3), directs the operator to use main feedwater regulator bypass valves for feed flow control. As a response-not-obtained (RNO) action, AFW use is directed, and has a substep to "verify AFW alignment". The mini recirc valves are not included. A NOTE containing the flow rate at which AFW pumps will trip due to over current induced by pump runout precedes the feedwater flow control step. In another step (step 4), the operator is directed to stabilize S/G level but is not provided details on how to accomplish the task. The RNO action specified is to "stop feed flow to that S/G." This action applies whether feed flow is being provided by main feedwater (via the bypass) or by the auxiliary feedwater pumps. There is also reference to controlling feed flow in step 1 related to maintaining RCS temperature.

The steps on S/G level stabilization and feed flow control have basically existed since the symptom-based EOPs were created in July 1985, as a result of NUREG-0737. They have never addressed the impact of loss of instrument air on the mini recirc valves. The effect of excessive AFW flow (i.e., pump runout) was introduced in about 1995.

The WOG ERGs for Reactor Trip Response do not address loss of instrument air, nor do they specifically address AFW pump mini recirc flow capability. The WOG considers such aspects to be plant specifics, to be addressed by the owner. The original WOG developmental guidance from 1984 contains little information on what (plant specific) systems should be addressed, or how. This trend continues through 1997, Rev 1C, which does generically identify that plant specific electrical loads (which covers one major cause of IA loss, compressors) should be a plant specific list. AFW and S/G level control specifics are not addressed. The WOG has always recognized that plant specific information is needed in EOPs and the Deviation and Background Document concepts were provided to manage such information.

At various times throughout the history of EOP-0.1 the importance of AFW in general (but not mini recirc flow in particular) has been recognized at PBNP. For example in Rev 7, 10/11/91, checking AFW actuation was step number 1 of EOP-0.1. AFW pump runout concerns were added in 1995. Loss of IA due to electrical bus availability was addressed similarly to AFW. For example in Rev 11, 11/22/94 (prior to the development of AOP-18A and -18B for train specific equipment operation) Appendix A to EOP-0.1 contained a list of Priority Electrical Loads, which included an IA compressor. Appendix A was deleted when AOP-18A & -18B were created.

Evaluation: PBNP EOP-0.1 is based appropriately on ERG guidance. The ERGs consider that plant specific information may need to be included in EOPs and provides means and mechanisms to document the same (Background and Deviation documents). The verification and validation (V&V) process described by the ERG procedure development process is intended to identify plant specific needs to be included in the plant specific EOPs. PBNP did not include operator guidance in EOP-0.1 on AFW minimum recirc flow under a loss of IA condition.

- **OM 4.3.1, AOP and EOP Writers' Guide:** The Writers' Guide contains the usage rules for notes and cautions that specify (in part):
  - A note is used to present advisory or administrative information necessary to support performance of the subsequent step(s).
  - Each document should provide enough information to accomplish the purpose of the document without relying on information contained in notes or cautions.
  - Notes and cautions should be declarative statements of fact and not commands or action statements unless they are advising on actions to be taken in the event of changing plant conditions.

The references listed in OM 4.3.1 were reviewed with the following results:

- NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures – 8/82: Note statements provide operators with supplemental information concerning specific steps or sequences of steps in the EOP. These statements should provide operators with enough information, and be located so as to ensure that they can easily relate the note to the step or steps to which it applies. Because they are supplemental, notes should not direct operators to perform actions. (p24)
- NUREG-1358, Lessons Learned From the Special Inspection Program for Emergency Operating Procedures – 4/89: In many cases action statement were found embedded in notes and cautions. Again, this increases the chance that the step will be overlooked and that an error will occur. (p4) Cautions and notes are not intended to direct operator action, but rather to warn of possible consequences or to provide supplemental information to the procedure steps. Inclusion of actions in a caution or note can be disruptive and confusing to an operator. More importantly, the action could be entirely overlooked if embedded in a caution or note. Any cautions or notes containing operator actions, including conditional actions or transitions, should be restructured so as to provide an action step plus a caution or note. (pC-3)
- NUREG-1358, Supplement 1, Lessons Learned From the Special Inspection Program for Emergency Operating Procedures – 10/92: Cautions and notes: notes (1) provide only supplemental information, and (2) no actions included. (P16)
- NUREG/CR-2005, Checklist for Evaluating Emergency Operating Procedures in Nuclear Power Plants – 4/83: Do explanatory notes avoid the use of action statements? (Statements directing personnel to perform actions must not be imbedded in explanatory notes.) (p7)
- PBNP Procedures Writers' Guide – 11/27/00: Cautions and notes shall <u>NOT</u> direct or infer actions. All required actions shall be stated in action steps. (p50) This procedure is not applicable to the AOPs or EOPs.
- WOG ERG Writers Guide – 7/1/87: Because the present action step wording is reduced to the minimum essential, certain additional information is sometimes desired, or necessary, and cannot be merely included in a background document. This non-action information is presented as either a NOTE or a CAUTION. (p22) NOTE is used to present advisory or administrative information necessary to

support the following action instruction. A CAUTION or NOTE may also be used to provide a contingent transition based on changes in plant condition. As a general rule, a CAUTION or NOTE will not contain an instruction/operator action. However, passive action statements in CAUTIONS or NOTES, which typically contain the words should, may or must, may be appropriate under certain conditions. An example is when continuous monitoring of a specific plant condition and an associated action is required.

Evaluation: OM 4.3.1 guidance on the content of a note is consistent with the WOG ERG Writers Guide, but contradictory to all of the other references cited. Some statements within the OM contradict others; specifically, the statement that "Each document should provide enough information to accomplish the purpose of the document without relying on information contained in notes or cautions" contradicts the intent of "unless they are advising on actions to be taken in the event of changing plant conditions."

Training

- **Continuing Training:** The overall content of the continuing training program is determined based on a two-year cycle. Presently the 2001/2002 LOR (license operator requalification) Long Range Training Plan is in effect. The Long Range Plan concept is very organized and structured with respect to content of the topics to be covered; it has been used since the mid-1990s. The content of the Long Range Training Plan is based, in part, on PRA information and includes a focus on systems with high safety significance. Prior to the Long Range Plan implementation, the content of LOR training was determined in a much less rigorous manner and on a much shorter time frame, typically on a 6 week-to-6 week cycle. Content was based on needs suggested by students, operations management and instructors plus inputs based on current events (such as design change implementation, procedure changes, plant and industry events).

  The 2001/2002 plan contains a number of topics pertinent to the issue of concern. The tasks for Loss of Instrument Air and Loss of Offsite Power were covered as well as a system review of Auxiliary Feedwater. The training devices used by instructors to cover the topics are LPs (Lesson Plans) and SGs (simulator guides). Both these devices present information in outline form, containing topical areas to be covered. The LPs are primarily oriented for classroom environment, whereas SGs are targeted for the simulator, mostly the instructor/ simulator operator. LPs clearly identify references and materials to be used as handouts. Typical support documents are drawings, procedures and OE documents. The LPs used in continuing training are the same LPs used for initial training. Training personnel indicated that LPs and SGs are reviewed prior to use and, to the best ability of the individual trainer, are updated to be current.

- **Initial Operator (CO and SRO) Training:** The highest-level document in Initial Training is the Program requirements (TRPR). They are position based. For example

TRPR 18 is Control Operator Trainee. The TRPRs are primarily administrative documents rather than technical. The TRPRs do identify the Training Courses (TRCRs) that comprise the Program. The TRCRs are a little more technical than Programs in that they identify some general areas of knowledge that the trainee needs to cover. For example, under TRPR 18, two of the courses are TRCR 52, Secondary Systems and TRCR 55, Integrated Operations. The TRCRs identify LPs. The LPs are the same as those used in continuing training. Some of the LPs specific to the event are LP 0169 AFW system, LP 0405 Reactor Trip or SI Response (which includes EOP 0.1), LP 0338 Instrument and Service Air (which includes AOP-5B) and LP 2439 Secondary Coolant System Malfunctions (AFW is one of those).

- Evaluation: LPs contain enough specific information about auxiliary feedwater and instrument air systems to accurately describe system operations, causes and effects. Training documents do not contain extremely specific details on specific evolutions. For example, the specific method for controlling steam generator level as directed in EOP-0.1 in concert with compounding events such as loss of IA, is not covered nor is the need to locally gag an AFW pump mini recirc valve upon loss of instrument air. Instructors review material to be taught in advance and are able to make changes in course content in order to add information, including current events and to change areas of emphasis. The Simulator Guide topics used in continuing training appear to be marginally related to the topic area they are listed under. PRA and human performance information is not included in LPs. PRA and CDF values are used as input to select the content of the Long Range Training Plan for continuing training.

Other Documents

- **DBD-01, Auxiliary Feedwater System Design Basis Document:** Revision 0 of DBD-01 was issued on 4/4/94. In Section 4.8, AFW Pump Recirc Flow Control Valves, there was a statement under Safety-Related Functions that "These valves shall open automatically and remain open to provide a recirculation flowpath from AFW pump discharge to the CST when flow in the AFW discharge line is insufficient to prevent pump damage." The reference cited was MR 88-099. The DBD also stated "These valves shall close automatically to prevent the unnecessary diversion of AFW pump discharge during high-flow conditions where adequate pump discharge flow is removing pump heat." Section 4.8.4 addressed these competing requirements stating "Since this valve has a safety function to close, and a less significant function to open (long-term pump protection) it is most reliable therefore to have the valve fail (upon loss of power or instrument air) to the closed position. This section also discussed a potential worst-case flow condition with both the recirculation valve closed (due to loss of IA) and the associated discharge MOV closed (single active failure), but concluded that this was outside the system design and licensing basis. This worst-case concern was based on NUREG-0800 assumptions, but was not considered applicable since PBNP had not incorporated NUREG-0800 into its licensing basis.

Revision 1 of DBD-01 was issued on 3/31/00. One of the major changes included was "Deleted safety-related function to OPEN for mini-recirculation valves for AFW pumps." The worst-case flow condition discussion remained in the DBD.

Evaluation: The basis for including an OPEN safety-related function for the recirculation valves in Revision 0 was cited as MR 88-099, the modification that increased the recirculation flow orifice size for AFW pump protection. A review of the modification paperwork did not identify any statement declaring a safety-related function for the valves to OPEN. A review of the DBD validation documentation indicated that in-service testing of the valves checked recirculation valve position. Testing of the recirculation valves in the OPEN direction was added to the IST Program in 1991. (See discussion of IST Program below.)

Revision 1 of DBD-01 deleted this OPEN safety-related function. This appears to be a result of actions coming from CR 97-3363 (discussed later). Testing of the valves in the OPEN direction was deleted from the IST Program on 9/30/98, also as a result of CR 97-3363. Overall, the basis for adding and deleting this function to the DBD was not well documented or justified.

- **EOP Verification and Validation (V&V):** The original EOPs issued in 1985 were verified by a multi-disciplined verification team using an approved procedure with a detailed checklist of attributes to be evaluated. That effort generated over 2500 discrepancy sheets and involved a series of more than 40 team meetings over a period of several years. The discrepancy sheets generated for EOP-0.1 did not raise any concerns with the step for controlling feed flow or stopping feed flow to a S/G if a level increase above the desired value occurred.

  The validation process involved a WOG review of the basic version of the ERGs at the Calloway simulator in 1982 and on the Revision 1 ERGs at the Seabrook simulator in 1983. Early drafts of some of the plant specific procedures were taken to the Zion simulator in March and April of 1983, which generated many suggested procedure changes. The procedures were then put through the previously described verification process. Following this, the procedures were used by operating crews at the Kewaunee simulator (8/84-11/84). Each crew spent a week mitigating accidents using the procedures. No concerns were raised regarding the actions to control feed flow or stop feed flow if S/G level increased above the desired level range. Finally, a portion of the detailed control room design review was expanded to provide another validation of the EOPs. A full size photographic mock-up of the PBNP control room was created and fourteen scenarios (increased from the typical 5 or 6) were evaluated in an attempt to ensure that every EOP was used. Operators performed walkthroughs of the EOPs during these scenarios, which were also videotaped for later review, and then interviewed for their comments (1985). EOP-0.1 was validated using a Reactor Trip without SI scenario (without a concurrent loss of instrument air). Again, no concerns were raised regarding the actions to control feed flow or stop feed flow if S/G level increased above the desired level range.

The EOP V&V process was also part of a NUREG-0737 Supplement 1 (GL 82-33) commitment. The EOP procedure generation package (PGP) was submitted to the NRC on 6/1/84. The NRC responded with a draft SER on 5/7/87 that found the PGP to be unacceptable. The PBNP revisions to the draft SER were submitted back to the NRC on 11/10/87, addressing each of the identified concerns. The NRC issued the final SER on 4/9/90 that contained additional programmatic improvements identified by the staff. The SER transmittal letter also referred to the June 1989 NRC Inspection of the EOPs and recommended that PBNP consider both the results of that inspection and the SER discussion and utilize them as appropriate in the next major revision of the EOPs. Current procedures governing the EOP V&V process are OM 4.3.2, EOP Verification Procedure, and OM 4.3.3, EOP Validation.

Evaluation: During the development of the PBNP EOPs from the WOG ERGs, information was to be included to address differences between the reference plant used by WOG and the Point Beach plant. Following development of those procedures, verification and validation reviews were applied to ensure the adequacy of those procedures. Validation is the process of evaluating the EOPs for usability by the operators and operational correctness (e.g., compatibility with plant hardware and control board layout). EOP-0.1 was operationally incorrect for a loss of IA condition. Therefore, it was the validation step in the EOP development and implementation process that failed. The need to evaluate EOP-0.1 using a loss of instrument air condition was not recognized because the validation process did not evaluate procedures, design and human error/timeline analysis concurrently.

- **EPRI Report TR-100259, An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment – 6/92:** This document is used by the PRA group in evaluating human interactions for the probability of an error. It identifies attributes of certain failure mechanisms that influence the overall probability that the mechanisms will contribute to a human interaction (HI). One mechanism, Relevant Step in Procedure Missed, has four attributes that are considered and evaluated in a decision tree:
  - Obvious vs. Hidden: Is the relevant instruction a separate, stand-alone numbered step, in which case the upper branch is followed, or is it "hidden" in some way that makes it easy to overlook, e.g., one of several statements in a paragraph, in a note or caution, or on the back of a page?
  - Single vs. Multiple: At the time of the HI, is the procedure reader using more than one text procedure or concurrently following more than one column of a flowchart procedure?
  - Graphically Distinct: Is the step governing the HI in some way more conspicuous than surrounding steps?
  - Place Keeping Aids: Are place keeping aids, such as checking off or marking through completed steps and marking pending steps used by all crews?

A hidden step had a 10% probability of being missed, whereas a procedure step exhibiting the best of all four attributes had a probability of only 0.1%, a reduction by

a factor of 100. The worst probability for an obvious step is only 1.3%, which is about a factor of 8 lower than a hidden step.

- **FSAR:** The FSAR did not include a description of AFW recirculation line features until updates were made in 6/97 and 6/98. The 1997 update involved the addition of a paragraph describing the diversion of AFW flow via the recirculation line to the CST for a 3-minute period following pump start. This was an original design feature that had never been included in the FSAR description of the AFW system. The 1998 update was an extensive change resulting from the FSAR Review and Upgrade Project that provided a more detailed description of the AFW system and its licensing basis. This change added the wording that each pump had an AOV controlled recirculation line back to the condensate storage tanks to ensure minimum flow to dissipate pump heat. This change also revised the time period for AFW flow diversion during pump start from 3 minutes to 45 seconds.

- **Individual Plant Evaluation, Revision 0 dated 6/30/93:** The original IPE for Point Beach was developed from a snapshot of the plant and procedures as of 9/5/90. Many of the success criteria for systems in the IPE PRA model were based on design basis assumptions. In the original PRA system notebook for Auxiliary Feedwater, it was recognized that the minimum recirculation flow valves failed closed on a loss of instrument air. However, this was not included in the PRA model as a failure mode for AFW because it was assumed that these valves failing to open did not result in pump failure. Assumption 22 in Section 4.6.7.1 of the notebook states:

  > The discharge lines of the AFW pumps have recirculation lines back to the CSTs. These lines are normally isolated by AOVs that fail closed on loss of power or instrument air. Although they receive open signals upon a pump start and when pump flow is low, it is assumed that failure to open does not fail the AFW pump. Failure of one of these AOVs in a full open position is assumed to fail the associated AFW train due to diversion of pump flow.

  The potential to damage the AFW pumps with lack of flow was mentioned briefly in the notebook. In Section 4.6.2.2 on Support Systems, the following discussion is found under the "Instrument Air" heading:

  > The mini-recirculation valves on both the turbine-driven AFW pumps (AF-4002) and the motor-driven AFW pumps (AF-4007 and AF-4014) fail shut on a loss of instrument air. This could cause overheating of these pumps on low flow conditions with no recirculation flow available.

  These two sections seem to contradict each other. However, controlling (reducing) AFW flow was assumed to take place later in the transient so there was plenty of time for the operators to perform this action correctly. This was based on decay heat removal curves. Again, there appeared to be an emphasis on ensuring that enough flow was available in the transient initially and it was not recognized how early in the event that AFW flow needed to be reduced to prevent overfilling the Steam Generators. This is evidenced by Assumption 13 where operator actions to control

AFW flow later in the transient are discussed. No mention is made of ensuring a minimum flow path is available:

> Operator actions to control AFW flow later in an accident sequence are not explicitly modeled in the AFW system fault trees. Operator actions are necessary to prevent the AFW system from overfilling the steam generators as their pressures decrease and AFW flow likewise increases. This was not modeled since there is a long time available and the function would be alarmed. In addition, the operator would have to successfully supply an alternate source of water to the suction of the AFW pumps (not automatic) and then forget to control flow or check steam generator level.

It seems from these statements in the notebook that some injection flow was always assumed to be required. The need for the operator to shut off flow to the Steam Generators entirely from one or more AFW pumps at some time in the event was apparently not considered.

In Section 4.6.4.2 of the notebook, initiating event impacts on the system are discussed. Under the "Loss of Instrument Air" heading, only the discharge valves for the motor driven pumps are considered. The closure of the mini-recirculation valves for the AFW pumps was not documented as a possible effect of the Loss of Instrument Air event:

> A loss of instrument air will degrade the operators' ability to throttle the flow rates of that portion of the AFW system associated with the motor-driven AFW pumps. The discharge pressure control valves, which are intended to limit flow to 200 gpm per pump, (AF-4012, 4019) are air-operated and would fail open on a loss of instrument air. Under this condition the operator is directed to use the turbine-driven pump to supply feed per AOP-5B, "Loss of Instrument Air" (Reference 4.6-12) or use the local gag to control AF-4012 and AF-4019 per OI-62A, "Motor-Driven Auxiliary Feedwater System (P-38A&B)".

The notebook also contains a discussion of potential common cause failures for the AFW system. This review did not identify the closure of each pumps minimum recirculation valve on a loss of instrument air as a potential failure mechanism. However, this is consistent with the assumption that failure of these valves to open does not fail the AFW pumps.

Updates to the original IPE PRA model (1990) were based on snapshots of the plant taken in 1993 and again in 1996, and implemented a few years later (due to the long time required to perform the model update). The focus of these updates was to incorporate new plant-specific failure data and to incorporate model changes that reflected plant modifications. The PRA model update being completed this year is the first time since the original IPE effort that critical systems were examined from the ground up in a detailed review to ensure all failure modes are captured. This was accomplished in part by use of detailed failure modes and effects fault trees. Adding this detail was considered to be necessary at this point to make the model more

flexible for risk-informed applications. It was the use of this approach that identified the concern with operator actions to control AFW flow.

- **IST Program:** In December, 1990 the 3rd interval program (Revision 0) was implemented. There is a line item in the general valve section that states "Due to isolation of S/G by EOPs, it may be necessary for an operating pumps recirc path to be available." The testing to verify the open function was not included in the tabular section of the IST program that identified the actual testing to be done. A valve program relief request (VRR-28) was added to the IST Program under Revision 1 on 5/28/91 that described the recirculation valves function to be "These valves open to ensure minimum recirculation flow from the pumps to prevent pump damage." A cold shutdown test frequency was being sought.

  The NRC issued a Technical Evaluation Report (TER) on 4/17/92 that denied the relief request because the valves had a safety function in the closed position and noted that the recirculation valves were not tested by the IST Program in the open position. The TER referenced the VRR-28 function statement and went on to state "The program should be revised to address these valves' safety function in the open direction." PBNP responded to the NRC on 7/30/92 to clarify that the valves could not be stroked except by use of hand wheels until modifications were made that allowed manual stroking using air. The response also stated "Since the AF pumps are capable of delivering feedwater at any steam generator pressure, the minimum flow valves are not required to open to protect the AF pumps under any anticipated accident conditions. The valves will, nevertheless, be stroke time tested in the open direction, as well as in the shut direction, once the modification to permit stroke time testing is completed." A follow-up letter dated 3/2/93, informed the NRC that the modifications would be completed by the completion of the spring 1993 refueling outage and VRR-28 relief request was being withdrawn. Revision 3 to the IST Program was implemented on 3/30/93 deleting relief request VRR-28.

  On 10/15/97, CR 97-3363 raised a question about a discrepancy between the open function testing of the AFW recirculation line check valves (not in the IST Program) compared to the recirculation flow control valves (in the IST Program). The evaluation of this concern concluded on 2/5/98 that there was no safety related function for the recirculation valves or check valves to open, and the IST Program would be revised. Revision 5 of the IST Program was issued on 9/30/98 and deleted the open function testing of the recirculation flow control valves.

## Interview Results

Personnel Statements: Written statements were obtained from key personnel involved in the evolution of this issue covering the period of initial discovery to its reporting to the NRC. The information derived from those statements has been incorporated into the timeline included in Attachment B and involved the following personnel:

- PRA Engineer
- Design Engineer

- Design Engineering Manager
- Regulatory Compliance Engineer
- AFW System Engineer
- Operations Manager
- PRA Supervisor

Interviews: Interviews were conducted with the following individuals to obtain additional information:

PRA Engineer: An interview was conducted with the PRA Engineer that identified the concern with operator actions to control AFW flow. That interview identified the following points:

- The PRA group reviewed the effect of the EOP change made (addition of foldout page information) but did not make recommendations on the best method of accomplishing the incorporation of that information. Use of the foldout page resulted in a reduction of the Human Error Probability (HEP) from 0.5 to 0.05.

- Use of a foldout page is treated as a continuous step with some additional credit for other control room personnel and training; it does not have as high of a CDF reduction factor as a specific check.

- Credit was given in the recovery factor calculated for use of a procedure reader; it was treated the same as an extra crew.

- The PRA Engineer received information in June or July 2001 that operators stop AFW flow by using valves versus stopping pumps. The information was obtained during discussions with an operating crew. This information was verified later via operator interviews conducted by the HEP expert.

- The PRA group provides feedback to Training, via informal communications, on high-risk accident sequences, but not on specific procedure steps that have high HEPs.

EOP Coordinator: An interview was conducted with the EOP coordinator and identified the following points:

- The direct work item system is a process that allows procedure changes to be made. Direct work items are changes that are issued by the WOG after review by the appropriate WOG subcommittee. Essentially they are revisions to the ERGs. Any member of WOG can initiate a possible direct work item but it does not become one until issued by the WOG.

- Changes to the EOPs can also be initiated internally without going through the WOG using the procedure feedback process. When this mechanism is used, the EOP Coordinator and an Operations Procedure Writer evaluate the request to decide if it should be processed, and the EOP set changed. There is no procedurally defined process that describes the evaluation methodology. There does not seem to be any

guidance on determining specific technical content of a change if it is outside the ERG.

- Foldout page content is expected to be memorized by the operator. Foldout page information is intended to trigger operator memory. The addition of foldout page information to EOP-0 and EOP-0.1 is applicable at all times to continually control AFW flow correctly; this includes transition out of EOP-0 and EOP-0.1. The EOP Coordinator did not consider the PRA value of foldout page use versus other methods of incorporating the desired actions into procedures when the decision to use a foldout page was made.

- No formal V&V was performed on the foldout page change to the EOPs; a serial review was performed.

- The EOP Coordinator believes that Operations generally keeps Training informed of training needs.

- The EOP Coordinator thinks the changes made to the EOPs are done to streamline the procedures.

**Other Information**

During preparations for the NRC Regulatory Conference held on this issue, discussions with the participants identified the following:

- The timing of operator actions for S/G level control assumed in the original IPE was based on decay heat curves. Diversion of flow (by gagging open the recirc valves) was not envisioned earlier in the accident scenario. The timing of operator actions to throttle AFW flow to a level requiring a recirculation flow path due to S/G overfilling or RCS overcooling concurrent with a loss of IA was not recognized.

- The EOP procedure weakness was very difficult to identify. It was only from an integrated perspective of evaluating AFW system design, procedural guidance, and FMEA, overlaid with human error probability analysis and timeline studies that the issue could be identified.

- The PBNP instrument air system has multiple cross-ties between units and redundancies that requires a dual unit event to cause a complete loss of IA. The EOPs are single-unit emergency procedures and do not consider dual unit casualties.

- During a SBO event, based on the required condition for decay heat (100% power for 100 days), the need to throttled AFW flow to levels requiring the recirculation valves to open would not occur for about 5 hours, well after the time that IA is restored. Therefore, the review of this event would not identify the EOP procedural vulnerability.

Industry and Station Operating Experience

Internal Operating Experience

- **CR 97-3363, IST Program Design Basis for AFW Minimum Flow Recirculation Valves:** This CR was initiated on 10/15/97 to address a concern with a conflict between the IST Program and the AFW DBD. The IST Program stated that the AFW recirculation line check valves did not have an active safety function to open and that the minimum flow recirculation lines were not needed since there was always adequate flow to the S/Gs under accident conditions. This conflicted with the AFW DBD that did not address the check valves, but had an open safety function for the recirculation valves. The IST Program tests the recirculation valves in the open and close directions. The DBD group performed an evaluation on 2/5/98 that concluded the check valves have no safety related function in either direction and that the recirculation valves only have a safety related function in the closed direction. The basis stated that the main safety related function of AF was to supply water to the S/Gs and that flow to the S/Gs was the most important flow path to maintain. The mini-recirc line was considered a diversion path, and since the AF system was capable of a cold start, a recirculation path was not necessary. The potential to deadhead a pump was considered, but establishment of a flow path through the discharge lines was used to eliminate the concern and the mini-recirc path was deemed to not be needed for pump protection. The evaluation noted that DBD-01 (Rev. 0) was being revised to reflect that there was no open safety function. The evaluation went on further to consider an AFW pump scenario where the associated discharge MOV failed to open or the pressure control valve inadvertently closed along with the recirculation path being blocked. In this event, the recirculation line would be required to prevent pump destruction, but the emergency function to feed the S/Gs is defeated anyway. This active single component failure scenario would only apply to one pump, so it would be acceptable and recirculation flow for AFW pumps was not a required safety related function.

- **QCR 99-0115, Code Testing Conflict With the Aux Feedwater Mini-Flow Recirc Check Valves:** This CR was initiated on 5/24/99 and addressed a concern that conflicting information existed about the safety related function of check valves AF-115 and AF-117 to OPEN compared to the AFW recirculation valves that have a safety related function to CLOSE. Further, the IST Program did not include these check valves. An evaluation performed on 5/27/99 concluded that the concern identified was in error and had already been addressed by CR 97-3363. Additional evaluation on 6/15/99 concluded that some clarification to the IST Program documentation was needed to address how AFW single failure affected the decision on testing. A new action item was generated to revise the IST Program documentation and closed on 6/19/00 with issuance of Revision 4 of Appendix A of the IST Background Document.

- **RCE 98-148, P-38A AFW Pump Recirc Valve Found Failed Shut, dated 1/29/99:** This RCE documented an event where an operator was in the process of starting an

AFW pump and noted that the recirculation line valve did not open as expected and then quickly secured the pump. This event showed that operators monitor recirculation valve position during AFW system manipulations.

INPO Operating Experience

- **SEN 174 - Loss of Nonvital Bus Causes Dual Unit SCRAM and Degraded Auxiliary Feedwater System, dated 11/10/97**
  Description: At the McGuire plant, a loss of non-safety related 120V AC instrument and control power caused both units to SCRAM. Also, the recirculation valves for all 3 U-1 AF pumps failed shut. The control board indication for these valves was also lost. As water level in the S/G was recovered, operators eventually shut the pump discharge valves. The pumps were operated for 20 to 60 minutes with their discharge and recirculation valves shut. Valve leakage was adequate to prevent pump damage.

  Evaluation: This event is very similar to our case. Our evaluation of the SEN focused only on the power supply failure. AF pump operation without recirculation flow was discussed in the SEN and one question raised was "what procedures require operators to ensure that adequate pump flow is maintained?" This question was not addressed in the evaluation of the SEN. CARB requested that this SEN be reviewed again. CA004279 was initiated to track this evaluation.

- **SOER 88-1 – Instrument Air System Failures, dated 5/18/88**
  Description: This document provides a review and evaluation of industry events associated with failures and degradations of instrument air systems. Recommendations 1, and 2 from this SOER are relevant to this event.

  Recommendation 1 (Operations) was to provide procedures to assist operators in the identification, control, and recovery from partial or total loss of instrument air events. A list of attributes that the operating, abnormal, and emergency procedures should provide included (in part) the following: identification of critical components operated by instrument air and the positions in which they fail, expected system and plant responses to a loss of IA and the consequences of these responses, actions to take if critical components do not fail in their intended position, and manual actions the operator should be expected to take to respond to a loss of IA event. The PBNP response was that AOP-5B, Loss of Instrument Air, contained the necessary instructions and information to assist operators in the identification, control, and recovery from partial or total loss of IA, and fully satisfied that recommendation. At that time, AOP-5B had an appendix for the AFW system that identified the recirculation valves as failing shut and requiring a manual gag override to open.

  Recommendation 2 (Training) from the SOER was to provide classroom and simulator training on loss of IA events to operators. The training was to provide the bases for such things as failure modes of critical components and expected operator actions, so that the operators would understand the major concerns involved in a loss of IA event. The PBNP response was to initiate Training Needs Analysis (TNA) 88-

0425 for the PBNP Training group to evaluate. The result was that classroom training on loss of IA was included in cycle 89-8 of AO, RO, SRO, and DTA continuing training. LP 1782, Revision 0 dated 11/1/89, Instrument and Service Air was developed and approved to address this need. That lesson plan included a section that lists concerns with a loss of IA that focused on four areas: heat removal, auxiliary feedwater, inadvertent safety injection, and containment isolation. For AFW, the lesson plan identified that on the electric driven AFW pumps, the PCV fails open, and on all AFW pumps the recirculation valves fail closed. No simulator training on loss of IA was provided because PBNP was using the KNPP simulator then and loss of IA could not be adequately modeled on it.

Evaluation: The PBNP response to recommendation 1 addressed the need for information in abnormal operating procedures, but did not directly address operating and emergency procedures. The reliance on AOPs for addressing specific plant conditions and using EOPs for general response and mitigation probably influenced the scope of the review. The classroom training specifically identified that the AFW pump recirculation valves failed close on loss of IA, but did not identify concerns with pump damage or the need to gag open the valves, as dictated by AOP-5B. However, there was a notation relating to the SI recirculation/test line isolation valves failing shut causing pump overheating in a few minutes and reference to an OPS Special Order 85-05 that had the valves currently gagged open. Simulator training was not performed due to modeling difficulties. Overall, the response did address the issue of the AFW recirculation valves failing closed on loss of instrument air. The reliance on AOP-5B for operator actions resulting from a loss of instrument air was reasonable based on what was known at that time.

- **OE 10727 – PRA Risk Insight to Improve Operator Actions, dated 9/11/00**
  Description: This document describes an event at another utility where the NRC identified that they did not effectively use PRA risk insight to improve the timeliness and reliability of mitigating operator actions prior to an actual event resulting in loss of all RCP seal cooling to 2 RCPs. For this event, it was determined that PRA updates were not being used to train operators on plant vulnerabilities to core damage.

  Evaluation: At PBNP, procedure ESG 5.1, PRA Maintenance and Update Guidelines, requires the generation of a condition report whenever new vulnerabilities are identified. However, there were no provisions in the ESG that addressed who should be trained. In response to OE 10727, a revision to ESG 5.1 was issued on 12/19/00 that specified what groups should receive training on PRA updates and newly identified vulnerabilities.

Other Operating Experience

- **Zion Station LER 90-002, 1A Auxiliary Feedwater Pump Cavitation, dated 2/15/90:** This LER describes an event where the 1A turbine-driven AFW pump was run in a deadheaded condition resulting in pump damage. Due to a combination of management error and procedural deficiency, the AFW pump was operated with both

the discharge valve and recirculation valve shut for a period of about eight minutes until an operator stationed locally at the AFW pump noted an abnormal temperature rise on the pump's thrust bearing, water hammer sounds, and that the oil cooling water relief valve had lifted. This event demonstrates that pump damage can occur in a short period of time when operating a pump in a deadheaded condition. The pump impeller was found to be damaged and required replacement.

NRC Generic Communications

- **Generic Letter 81-14, Seismic Qualification of AFW Systems, dated 2/10/81**
  The purpose of this GL was for licensees to determine the extent to which their AFWS are seismically qualified and to walk-down the non-seismic portions of the system and identify deficiencies. Our original response was submitted on July 16, 1981, in which we concluded that the PBNP AFWS is adequately protected for a seismic event. No specific mention was made of the AFWS recirculation valves or piping. In a response to the NRC follow-up request for additional information dated May 4, 1982, we specifically noted that the recirculation piping connections to the seismic AFWS piping were inspected and that the recirculation valves close upon receipt of a pump discharge flow signal. The NRC's Technical Evaluation Report (TER) of November 12, 1982, concluded that the PBNP AFWS did not provide reasonable assurance to perform its SR function following a seismic event. In our response dated December 15, 1982, we stated that the recirculation valves fail closed and the discharge AOVs fail open and concluded that the instrument air system that powers these valves is not required for AFWS functioning. Because of the questions concerning the recirculation piping not being well supported, we committed in this letter to independently support each air operated recirculation valve. Finally, in our letter dated April 26, 1985, we responded to the NRC request for comments on their revised TER. In the TER the staff postulated a failure during a seismic event of the non-seismic AFWS piping or a failure of the pump recirculation valves to shut following the switchover of the AFWS supply to service water. In our response we stated that under either condition the operator are trained to recognize off normal condition and that adequate time existed for manual operator actions.

  Evaluation: PBNP performed a design review that evaluated the seismic adequacy of foundations, supports and structures associated with the AFWS. Review of system operating procedures was not a reasonable response to the Generic Letter. Therefore, this very specific design review would not identify the EOP procedural vulnerability.

- **Information Notice 87-28, Air Supply Problems at US Light Water Reactors, dated 6/22/87**
  The internal evaluation of this IN consisted of a review of all systems that perform safety functions and contain air operated valve operators, for the effect that the loss of air would have on those safety functions. The failure positions of the AFWS valves are identified. The concern for pump damage or failure due to less than minimum pump flow with the recirculation valves failing shut is also discussed. However, the focus of the evaluation was on demonstrating that the AFWS pumps would always

feed the S/Gs with sufficiently high flow to protect the pump. This was documented in calculation N 87-041. At that time the discharge AOV for the electric AFW pumps failed open on loss of air; therefore, there was no identified concern with the recirculation valves failing shut.

Evaluation: PBNP verified the performance of safety-related functions with a loss of IA and that the AFW recirc valves must fail closed to assure the AFW safety-related function of providing flow to the S/Gs. It was also verified that adequate procedures existed (AOP-5B) to address a loss of IA, including the manual actions needed to gag open the recirc valves. Since PRA tools were not available yet, it is not reasonable that the EOP procedural vulnerability would have been identified.

- **NRC Bulletin No. 88-04, Potential Safety-Related Pump Loss, dated 5/5/88**
  This bulletin requested licensees to investigate and correct as appropriate two mini-flow design concerns. The first concern was the potential for deadheading one or more pumps that have a common mini-flow line. The second concern is whether or not the installed mini-flow capacity is adequate to prevent damage to safety related pumps. In a response dated June 28, 1988, we acknowledged that each of the pumps in the AFWS have their own recirculation lines with an AOV isolation valve and an orifice upstream of the common return line to the CST. We discussed the logic of the recirculation valves to open or shut dependent on AFWS forward flow but did not address the potential to lose recirculation on an instrument air failure. We also acknowledged that the flow orifice for the pumps will need to be replaced with higher flow orifices to ensure sufficient flow for indefinite pump cooling via the recirculation lines.

  Evaluation: PBNP did a design review of the recirc capacity needed for adequate long-term protection of the AFW pumps. Modifications were initiated to increase the recirc flow capacity to the required levels. Review of system operating procedures was not a reasonable response to this Bulletin. Therefore, this very specific design review would not identify the EOP procedural vulnerability.

- **10 CFR 50.63 Loss of All Alternating Current Power, effective 7/21/88**
  The NRC amended its regulations at 10 CFR 50.63 to require all nuclear power plants to be capable of withstanding and recovering from a station blackout (SBO) of a specified duration. Our initial response to this regulation, which addressed the appropriate guidance from Reg. Guide 1.155 and NUMARC 87-00 was submitted on April 17, 1989. In that response we stated that no air-operated valves are required to operate to cope with a SBO for one hour. We also completed an analysis on condensate inventory necessary to cope with the one hour SBO. We concluded that we had sufficient CST inventory, along with the initial S/G fluid inventory to maintain S/G decay heat removal capability. Clearly, for a SBO, only the TDAFW pumps would be available. The concern appeared to be assurance that sufficient water would be fed to the S/Gs until AC power was restored and AFW could be shifted to the safety related service water supply. The first NRC SER on SBO was dated October 3, 1990. The NRC agreed, based on our statement, "that the

compressed air is not needed to cope with an SBO for one hour and, after 1 hour, the Alternate AC power source will supply the compressed air." The Technical Evaluation Report (TER Page 16) also stated agreement that operation of the AFWS is independent of AC and IA for one hour. Indeed the concern identified in the Technical Evaluation Report was that the minimum volume of 10,000 gallons in the CST per unit, was insufficient and ultimately we had to revise our Technical Specifications to change that minimum CST volume to 13,000 gallons.

Evaluation: During a SBO event, only the TDAFW pumps are available (one per unit). The conditions for this event assume a decay heat load based on 100 days of operation at 100% power. Based on the high decay heat load and one TDAFW pump, it is not credible to stop or reduce AFW flow to a point where pump damage is incurred in the first hour. Therefore, it is not reasonable that the EOP vulnerability would have been found during reviews associated with a SBO event.

- **Generic Letter 88-14, Instrument Air Supply System Problems Affecting Safety Related Equipment, dated 8/8/88**
  In a February 20, 1989, response to this GL we stated that all safety related pneumatic equipment at PBNP is designed to fail to a safe condition with the safety function being tested in the PBNP IST Program. The AFWS discharge AOVs were specifically discussed and the concern expressed that the fail open position could potentially lead to over feeding of the S/Gs. There was additional correspondence to the NRC on July 27, 1989, in the form of a supplemental response concerning the potential problem with the discharge valves failing open. We also responded to an inspection report dated January 16, 1991, in which the NRC determined that PBNP had not fully complied with statements in our original GL response regarding testing of safety related AOVs. In this response we clarified that safety related valves with "passive" functions (do not perform a mechanical motion during the course of accomplishing a system safety function) were excluded from IST fail safe testing. We also noted that since the 1989 submittal the IST program was revised and reissued for the third 10-year interval and that the AFWS mini-recirculation valves were now fail safe tested.

  Evaluation: PBNP verified the performance of safety-related functions with a loss of IA and that the AFW recirc valves must fail closed to assure the AFW safety-related function of providing flow to the S/Gs. It was also verified that adequate procedures existed (AOP-5B) to address a loss of IA, including the manual actions needed to gag open the recirc valves. Since PRA tools were not available yet, it is not reasonable that the EOP procedural vulnerability would have been identified.

- **Generic Letter 89-04, Guidance on Developing Acceptable In-service Testing Programs, dated 4/3/89**
  The attachment to the GL listed eleven specific generic deficiencies related to IST programs and procedures. Item 9 addressed pump testing using minimum flow return line with or with out flow measuring devices. The concern for this item was for those pumps that could only be IST tested using minimum flow return. In our response

dated October 3, 1989, we confirmed that SI, RHR and AFW are tested in compliance with the GL position 9. The GL advised licensees that meeting the guidelines for Code testing does not supercede the thrust of Bulletin 88-04 (See discussion above).

Evaluation: This review of this issue does not appear to be a missed opportunity for evaluation of the EOP procedural vulnerability.

## Evaluation Methodology & Analysis Techniques

The analytical techniques used in this root cause evaluation were:
- Document Review
- Interviewing
- Event and Causal Factor Charting (Attachment D)
- Timeline Development (Attachment B)
- Why Staircase Development (Attachment C)

## Data Analysis Summary

### Identification of Causal Factors

A "Why Staircase" was constructed based on the information obtained in the Information & Facts Sources section of this report. This technique results in a repetitive asking of the question "why" until a detailed understanding of the problem is obtained. The "Why Staircase" for this event is provided in Attachment C. This approach identified three main causal factors that contributed to this event.

EOP-0.1 contains a step (step 1) to CONTROL feed flow because of RCS cool down considerations and another step (step 4) to STOP feed flow to a steam generator if an increasing level cannot be maintained below the desired setpoint – these steps do not specify the method to be used to CONTROL or STOP flow. (It is postulated that an operator could throttle the AFW discharge valves closed and with a loss of instrument air when the recirculation valves are failed closed, the running pumps would dead-head and destroy themselves in short period of time; a potential common mode failure.)

There were two reasons influencing why specific information was not provided in the EOP. First, reliance had been placed on AOP-5B for providing specific operator actions for a loss of instrument air scenario, and second, closure of the AFW discharge valves due to operator action was not previously considered as a possible failure mechanism.

Reliance on AOP-5B:
Reliance on AOP-5B was faulty because operator action to control AFW flow (under loss of instrument air conditions) was needed in the early steps of EOP-0.1. This need had not been identified prior to this event. A key opportunity to have identified this need was via the EOP validation process. The original validation of EOP-0.1 steps was done in 1985 using a Reactor Trip w/o SI scenario. This scenario did not include a concurrent loss of instrument air condition. Consequently, it would not matter what method an operator

used to control flow since either throttling flow or shutting off pumps would be successful. These steps have not changed since Revision 0, so additional validation would not have been required. It was the EOP validation barrier that failed. Validation was to ensure the operational correctness of the EOPs. The reason the barrier failed was because the interaction between design, procedures, and human error/timeline analysis was not evaluated concurrently, and the need for specific operator actions under a loss of instrument air condition was not recognized. **The Human Error/Timeline Analysis method was not available at the time the EOPs were originally validated.**

Another key opportunity to identify the need for operator action while in EOP-0.1 was when the initial PRA model was developed to support the IPE submittal in 1993. The original PRA model did not model operator actions to control AFW flow in the system fault trees because it was assumed (based on decay heat removal requirements) that there was a long time available and the function (S/G overfill) would be alarmed (assumption 13). The flaw in this assumption was not identified during the PRA model review because the fault trees were based primarily on functions described in design documents. Also, only operator actions taken to mitigate a failure were evaluated. The selection of the evaluation method using fault trees focused on design functions over other FMEA methods was based on an assumption that the design function approach was more conservative. The current PRA model review uses a methodology that integrates system performance with potential human actions to obtain a spectrum of plant responses. **The original PRA Model was based on system functions, and only operator actions to mitigate failures were evaluated.**

Finally, routine performance of accident scenarios on the PBNP simulator should also have provided an opportunity to identify this need for operator action. Simulator Guides are presented in outline form and do not contain detailed information on evaluation of all actions performed during the scenario. PRA information has been used to identify which scenarios are important to teach from a risk perspective, but information on which steps in emergency procedures are risk-significant has not been incorporated into scenario evaluation criteria. The operator action to control AFW flow had not been identified as a human interaction with a human error probability assigned to it **(because Human Error/Timeline Analysis was not available yet).** Consequently, scenarios often went quickly through the loss of air condition to other conditions such as loss of secondary heat sink without evaluating the intermediate steps such as S/G level control. **The interface between the PRA and Training programs is less than adequate.**

<u>Operator Action was not Previously Considered as a Possible Failure Mechanism</u>
Previous evaluations of the effects of the AFW recirculation valves failing closed on loss of IA concluded that the AFW pumps would not be damaged because forward flow was always available. Closure of a single discharge valve due to component failure concurrent with the AFW recirculation valve failing closed was evaluated and considered to be outside the design and licensing basis. (This used NUREG-0800 assumptions and PBNP was not committed to that NUREG.) Closure of all the discharge valves due to operator action was not considered. The two reasons identified for not considering

operator actions were the lack of integrating human actions into failure mode analyses and the lack of insight that a specific operator action could result in pump damage.

Although the concept of determining the potential failures that could result from human errors has been around since at least the TMI accident, it is most often utilized in the PRA area. The current design process does not prompt an evaluation of failure modes from a human action perspective. When the MDAFW pump discharge AOVs were modified with a nitrogen back-up system, a throttling capability was created that did not exist on that valve before (under a loss of instrument air condition). Throttling of the MOVs that direct AFW flow to the respective steam generators had already existed, so this was an additional opportunity to perform that same action on another component. Only recent use of failure mode fault tree tables in the PRA program allowed identification of the concern on AFW control. The knowledge learned from evaluating human interactions in the PRA program has not been transferred into the failure modes and effects analysis element of the design control program. **The interface between the PRA and Design Control programs is less than adequate.**

Insight was needed to understand that the actual operator response to a "CONTROL or STOP feed flow" command under a loss of instrument air scenario would be closure of the discharge valves instead of stopping the AFW pumps. The expected operator response to the "CONTROL or STOP feed flow" command under a loss of instrument air scenario was not clearly stated in training documents. Knowledge that operation of the AFW discharge valves had a human error probability associated with it could have resulted in focused training on that evolution that may have identified the potential for pump damage. However, the information on risk-significant human interactions was not effectively incorporated into the training program. **The interface between the PRA and Training programs is less than adequate.**

## Other Conclusions

The assumptions used by the PRA group in evaluating human interactions are based on industry guidelines that determine how the effectiveness of procedures is established. These same rules have not been applied to our process for procedure writing. One example is the use of action steps in notes. The industry guidance is clearly not to include actions in notes. However, the AOP and EOP Writers' Guide (and WOG ERG Writers Guide) allows the use of condition monitoring that initiates an action in a note. Under PRA rules, little credit is given for an action embedded in a note. Procedure effectiveness can be improved by incorporating PRA rules into our procedure development process. **The interface between the PRA and procedure development processes is less than adequate.**

ESG 5.1, PRA Maintenance and Update Guideline, is the governing document for administration of PRA updates. That procedure contains interfaces with departments outside of Engineering. The use of a higher tier document may be more appropriate for this process. **Organizational interfaces for the PRA update process lack formality.**

There was a lack of consistency between different design basis and licensing documents regarding the description and function of the AFW recirculation valves. The predominant position taken in various licensing correspondence was that AFW flow could always be provided to the S/Gs and the recirculation valves were not required to provide an open safety function. However, the initial AFW DBD (1994) contained a statement that the valves had an open safety function, and the basis was not clear. The open function was removed from the AFW DBD in 2000. The IST program did not include an open safety function, but did test the valves in the open direction based on prior NRC correspondence (1992). That testing was removed from the IST program in 1998. The FSAR did not include any discussion of the recirculation line function until updates made in 1997 and 1998. **Consistency between AFW licensing and design basis documents is less than adequate.**

The subject of AFW flow and recirculation capability was part of many prior evaluations. However, the combined evaluation of design, procedures and human error timeline analysis only occurred during the recent PRA model update process. **Without the use of these combined analyses, it was not reasonable that previous evaluations would have identified this vulnerability.**

### *Failure Mode Identification*

| | |
|---|---|
| RR5 | Actions Not Tied to Another Process When Necessary – Actions required by one program not belonging to any program, which is needed to ensure consistency. |

- Information on risk-significant human interactions was not effectively incorporated into the operations training program, including scenario development
- Knowledge learned from evaluating human interactions in the PRA program has not been transferred into the failure modes and effects analysis element of the design control program
- PRA concepts are not included in the emergency procedure development process
- Consistency in the licensing and design basis for the AFW system was not maintained between the FSAR, AFW DBD and IST program

| | |
|---|---|
| RR2 | Actions Not Clear – Inadequate program design |

- The original validation of EOP-0.1 steps done in 1985 using a Reactor Trip w/o SI scenario did not include a concurrent loss of instrument air condition because the analytical tools (Human Error/Timeline Analysis) needed to identify this were not available at that time

| | |
|---|---|
| F2 | Inadequate Communications Among Organizations – Lack of interface formality |

- The PRA update interface requirements with other organizations are contained in an Engineering Supplemental Guideline, and lack formality

| | |
|---|---|
| J4 | Wrong Assumptions – Erroneous assumptions used in decision making |

- Only operator actions taken to mitigate failures were evaluated in the original PRA model
- The selection of the original PRA model evaluation method using fault trees focused on design functions over other FMEA methods was based on an assumption that the design function approach was more conservative

## VII. Root Causes & Contributing Factors

### Conclusions

The investigation found that the EOP validation process is the barrier that failed, causing the weakness in EOP-0.1. The EOP validation process failed because it did not evaluate the interaction among design, procedures, and human error timeline analysis. It was only from this integrated perspective that a loss of instrument air causing the recirculation valves to fail closed, combined with a possibility that an operator would close the discharge valve on an AFW pump, and the timing of this action prior to implementation of the abnormal procedure for loss of instrument air (AOP-5B) could the potential be seen to damage multiple AFW pumps. The combination of FMEA, timeline studies, and human error analysis is a recently implemented practice in the industry unique to PRA. Without the use of these combined analyses, it was not reasonable that previous evaluations would have identified this vulnerability.

### Root Cause

The root cause of the EOP procedural weaknesses was the failure of the original EOP validation process barrier to identify that specific operator actions were needed to properly control or stop AFW flow under a loss of instrument air condition. This barrier failed because the analytical tools needed to identify this vulnerability did not exist at that time. This resulted in a misalignment between plant design and procedural guidance.

### Contributing Causes

Significant contributing causes to this condition continuing to exist were:

- The original PRA model fault trees evaluated system performance primarily on functions described in design documents and only considered operator actions taken to mitigate a failure

- Previous evaluations focused on delivery of the minimum required AFW flow for providing decay heat removal

Other causes that were not significant contributors were:

- The failure to consider human actions during FMEA reviews in the design control processes,
- The lack of integration of human error reduction methods into the operations training process,
- The lack of integration of human error reduction methods into the emergency procedure development process,
- The lack of formality of organizational interfaces in the PRA update process, and
- The inconsistencies between the FSAR, AFW DBD, and the IST program concerning the description and function of the AFW recirculation valves.

## VIII. Corrective Actions

**Interim Corrective Actions (mitigation)**

- CA #1 Responsible Group: Operations, Completion Due Date: Complete

  Revise EOP-0, EOP-0.1 and ECA-0.0 to address AFW control under loss of instrument air conditions.

**Corrective Actions to Prevent Recurrence (CATPRs)**

- CA #1 Responsible Group: Engineering (PRA), Priority: 2, Completion Due Date: Complete [CA003691]

  Assist Operations in determining what initiating events should be included in the EOP validation process by formally providing information on which initiating events considered risk-significant for each EOP.

- CA #2 Responsible Group: Operations, Priority: 2, Completion Due Date: 8/5/2002 (90 days after CATPR #1 is completed) [CA003692]

  Revise the EOP validation process to ensure that appropriate initiating events are included. Utilize PRA input in determining what initiating events are applicable.

**Corrective Actions to Restore (broke – fix)**

- CA #1 Responsible Group: Engineering (PRA), Priority: 3, Completion Due Date: 10/4/2002 [CA003693]

  Complete the analysis portion of the PRA model review to identify any other risk-significant vulnerabilities in the current EOPs.

- CA #2 Responsible Group: Operations, Priority: 3, Completion Due Date: Complete [CA003694]

  Review the operator actions specified in AOP-5B to determine if they should be included in applicable EOPs to ensure timeliness of the actions, and initiate revisions as required.

- CA #3 Responsible Group: Engineering (PRA), Priority: 3, Completion Due Date: 6/5/2002 [CA003695]

  Formally provide Operations and Training with an updated list of high-risk human error events based on the PRA model.

- CA #4 Responsible Group: Engineering (PRA), Priority: 3, Completion Due Date: 6/5/2002 [CA003696]

  Formally provide Operations and Training with a description of the human error reduction methods used in evaluating operator actions in the PRA model.

- CA #5 Responsible Group: Operations, Priority: 3, Completion Due Date: 10/4/2002 (120 days after CA #2 and CA # 3 are completed) [CA003697]

  Review EOPs and AOPs containing high-risk human error events against human error reduction methods used in the PRA model and revise where appropriate to achieve significant CDF risk reduction.

- CA #6 Responsible Group: Operations, Priority: 3, Completion Due Date: 10/4/2002 (120 days after CA # 3 is completed) [CA003698]

  Revise OM 4.3.1, AOP and EOP Writers' Guide, to incorporate human error reduction methods used in the PRA model that can significantly reduce CDF risk.

- CA #7 Responsible Group: Training, Priority: 3, Completion Due Date: 10/4/2002 (120 days after CA #2 and CA # 3 are completed) [CA003699]

  Review initial operator training materials and methods associated with high-risk human error-events against human error reduction methods used in the PRA model and revise where appropriate to achieve significant CDF risk reduction.

- CA #8 Responsible Group: Training, Priority: 3, Completion Due Date: 10/4/2002 (120 days after CA # 3 is completed) [CA003700]

  Revise operator training procedures to incorporate human error reduction methods used in the PRA model that can significantly reduce CDF risk.

- CA #9 Responsible Group: Engineering (PRA), Priority: 3, Completion Due Date: 6/5/2002 [CA003701]

  Revise the AFW PRA model to accurately reflect system performance.

- CA #10 Responsible Group: Engineering (Systems), Priority: 3, Completion Due Date: 6/5/2002 [CA003702]

  Review the description of the AFW recirculation line function in the FSAR, DBD-01, and the IST Program for consistency and accuracy, and initiate revisions as required.

- CA #11 Responsible Group: Engineering (Design), Priority: 3, Completion Due Date: 6/5/2002 [CA003703]

  Revise the design process to include consideration of human action induced failure modes.

- CA #12 Responsible Group: Engineering (PRA), Priority: 3, Completion Due Date: 6/5/2002 days [CA003704]

  Evaluate if an Engineering Supplemental Guideline is the appropriate procedural method for controlling PRA updates, or if a higher tier document such as a Nuclear Procedure (NP) should be used considering the interfaces involving other departments. Initiate any procedure changes resulting from that evaluation.

- CA #13 Responsible Group: Engineering (PRA), Priority: 3, Completion Due Date: 6/5/2002 [CA003705]

  Revise the procedure governing PRA updates to include identification of the formal methods to be used for providing information to other groups. Use of existing processes, such as training work requests and procedure feedback forms, should be used whenever possible.

- CA #14 Responsible Group: Assessment, Priority: 3, Completion Due Date: Complete [CA003982]

  Review SEN 174 response and re-open the OE items if not fully addressed.

- CA #15 Responsible Group: Operations, Priority: 3, Completion Due Date: Complete [CA004279]

  Review SEN 174 and verify that procedures exist for maintaining adequate pump flow, including pumps other than AFW.

- CA #16 Responsible Group: Engineering (PRA), Priority: 4, Completion Due Date: Complete [CA004388]

  Review operator action assumptions in the PRA model for validity for the top risk-significant systems.

- CA #17 Responsible Group: Training, Priority: 3, Completion Due Date: Complete

  Update the PBNP simulator to model AFW pump failure due to less than required minimum recirculation flow.

- CA #18 Responsible Group: Operations, Priority: 3, Completion Due Date: Complete

  Revise the EOP validation process to include PRA involvement. OM431 OM432

- CA #19 Responsible Group: Engineering (Design), Priority: 3, Completion Due Date: Complete

  Modify the AFW recirculation valves to provide a back-up pneumatic supply to allow time for operator actions.

## IX. References

AOP-5B, various revisions, Loss of Instrument Air
CR 97-3363, dated 10/15/97, IST Program Design Basis for AFW Minimum Flow Recirculation Valves
CR 01-2278, dated 7/6/01, AFW PRA Model for Loss of Instrument Air
CR 01-3595, dated 11/29/01, PRA for AFW System
CR 01-3633, dated 12/4/01, Response of MDAFWPs to an Appendix R Fire
CR 01-3641, dated 12/4/01, AFW Pumps Common Mode Failure Information for CR 01-3595 RCE
CR 01-3648, dated 12/5/01, Response of MDAFWPs to an Appendix R Fire
CR 01-3654, dated 12/6/01, AFW System DBD Missed Opportunity
DBD-01, Revision 0, dated 4/4/94, Auxiliary Feedwater System
DBD-01, Revision 1, dated 3/21/00, Auxiliary Feedwater System
DD-EOP-0, various revisions, Deviation Documents - Reactor Trip or Safety Injection
EOP-0, various revisions, Reactor Trip or Safety Injection
EOP-0.1, various revisions, Reactor Trip Response
EPRI Report TR-100259, dated 6/92, An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment

Event Notification Worksheet EN#38525, dated 11/29/01
Event Notification Worksheet EN#38525 Supplemental, dated 11/30/01
FSAR, Chapter 10, various revisions, Auxiliary Feedwater System (AF)
Individual Plant Evaluation, Revision 0, dated 6/30/93
IST Background Document –Appendix A, dated 5/17/00
IST Program – 3rd Interval, various revisions
Internal Memorandum, dated 12/3/01, CR 01-3595 Reportability Recommendations
NUREG-0899, dated 8/82, Guidelines for the Preparation of Emergency Operating Procedures
NUREG-1358, dated 4/89, Lessons Learned From the Special Inspection Program for Emergency Operation Procedures
NUREG-1358 Supplement 1, dated 10/92, Lessons Learned From the Special Inspection Program for Emergency Operation Procedures
NUREG/CR-2005, dated 4/83, Checklist for Evaluating Emergency Operating Procedures Used in Nuclear Power Plants
OD 01-3595 Rev. 0 dated 11/30/01, and Rev. 1 dated 12/1/01
OD 01-3648 Rev. 0 dated 12/7/01
OM 4.3.1, Revision 1, dated 6/4/99, AOP and EOP Writers' Guide
OM 4.3.2, Revision 1, dated 6/14/95, EOP Verification Procedure
OM 4.3.3, Revision 0, dated 7/30/93, EOP Validation
PRA System Notebook – AFW, Revision 0, dated 1991
QCR 99-0115, dated 5/24/99, Code Testing Conflict With the AFW Mini-Flow Recirc Check Valves
RCE 98-148, dated 1/29/99, P-38A AFW Pump Recirc Valve Found Failed Shut
S-A-ENG-01-03, PBNP PRA Peer Review Report (Draft Report – 7/01)
SEN 174, dated 11/10/97, Loss of Nonvital Bus Causes Dual Unit Scram and Degraded Auxiliary Feedwater System
WOG ERG Executive Manual
WOG ERG Writers Guide, dated 7/1/1987
WOG LP-ERGs
Zion Station LER 050-295/90-002-00, dated 2/15/90, 1A Auxiliary Feedwater Pump Cavitation

## X. Attachments

Attachment A: Team Charter

Attachment B: Timeline

Attachment C: Why Staircase

Attachment D: Event & Causal Factor Chart

## *Attachment A: Team Charter*

# Root Cause Investigation Charter

CR 01-3595
RCE 01-069

**Issue Manager:**

Rick Mende

**Problem Statement:**

Discovery during the review of the AFW PRA model for transients involving loss of instrument air that emergency and abnormal operating procedures may not adequately address maintaining minimum AFW pump recirculation flow to prevent AFW pump failure.

**Investigation Scope:**

Determine the following:

- the root cause of why the condition exists
- why the problem was not identified previously

Make recommendations for:

- correcting the problem
- preventing recurrence of the problem
- applicability of the root cause to other areas (extent of condition)

**Team Members:**

Team Leader – Richard Flessner, Engineering Processes
Team Member – R. Wood, PRA
Team Member – J.P. Schroeder, System Engineering
Team Member – T. Staskal, Site Assessment
Team Member – C. Krause, Licensing

**Milestones:**

Status Update – 12/11/01
Draft Report – 12/20/01
Final Report – 1/10/02

**Approved:** (Original signed by F. Cavia) **Date:** 12/4/2001
Fred Cayia, PBNP Plant Manager

## Attachment B: Event Timeline

| DATE / TIME | DESCRIPTION |
|---|---|
| 9/1/79 | M-623/624 TDAFP alternate bearing cooling supply modification issued |
| 2/1/80 | IC-274 AFW recirculation valve logic (keep open) modification issued |
| 2/10/81 | GL 81-14 issued on Seismic Qualification of AFW System (response is dated 7/16/81) |
| 5/4/82 | Additional response to GL 81-14 due to NRC RAI – response says that AFW recirc valves close on receipt of AFW pump discharge flow signal |
| 6/82 | WOG Basic ERGs validated on Calloway Simulator |
| 8/82 | NUREG-0899, Guidelines for the Preparation of EOPs, is issued |
| 8/31/82 | IC-274 AFW recirculation valve logic (keep open) modification cancelled |
| 11/12/82 | NRC issues TER concluding that PBNP AFW system did not provide reasonable assurance to perform its SR function following a seismic event |
| 12/15/82 | PBNP response to NRC TER on AFW – concluded that IA is not required for AFW system functioning (based on recirc valves FC and discharge valves FO); commit to independently supporting each recirc valve |
| 4/83 | NUREG/CR-2005, Checklist for Evaluating EOPs, is issued |
| 8/1/83 | MR 83-104 AFW system discharge MOV controls modification issued |
| 4/26/85 | PBNP response to revised NRC TER on AFW – conclude that AFW piping failure or failure of AFW recirc valves to close will be handed by operators trained to recognize off normal condition that adequate time exists for manual action |
| 7/1/85 | Revision 0 of the EOPs issued |
| 5/2/86 | AOP-5B, Loss of Instrument Air, Revision 0 issued |
| 6/22/87 | IN 87-28 issued on Air Supply Problems at US Light Water Reactors |
| 7/1/87 | WOG ERG Writers Guide issued |
| 12/20/87 | IN 87-28 Supplement 1 issued on Air Supply Problems at US Light Water Reactors |
| 3/23/88 | NPERS evaluation of IN 87-28 issued via NEPB 88-090 |
| 5/5/88 | IEB 88-04 issued on Potential SR Pump Loss (response is dated 6/28/88) |
| 5/18/88 | INPO issues SOER 88-01 on Instrument Air Failures |
| 7/7/88 | MR 88-099 AFW pump mini-recirculation line improvements modification issued |
| 7/21/88 | SBO Rule (10CFR50.63) became effective (response is dated 4/17/89) |
| 8/8/88 | GL 88-14 issued on Instrument Air Supply System Problems Affecting SR Equipment (response is dated 2/20/89) |
| 4/89 | NUREG-1358, Lessons Learned From the Special Inspections Program for EOPs, is issued |
| 4/3/89 | GL 89-04 issued on Guidance on Developing Acceptable IST Programs (response is dated 10/3/89) |
| 5/8/89 | MSS approves response to SOER 88-01 |
| 2/15/90 | Zion Unit 1 LER issued on AFW Pump Cavitation |
| 12/90 | 3rd interval IST Program is implemented |
| ~1991 | Original IPE Notebooks developed |

| DATE / TIME | DESCRIPTION |
|---|---|
| 5/28/91 | Revision 1 to IST Program adding VRR-28 on recirc valves |
| 4/17/92 | NRC issues TER on IST Program denying VRR-28 and requesting OPEN safety function be added for recirc valves |
| 6/92 | EPRI Report TR-100259, An Approach to the Analysis of Operator Actions in PRA, is issued |
| 6/19/92 | MR 92-091/092/093 IST testability of AFW recirculation line AOVs modifications issued |
| 7/30/92 | PBNP response to NRC TER clarifying that recirc valves are not required to OPEN to protect AFW pumps |
| 10/92 | NUREG-1358 Supplement 1, Lessons Learned From the Special Inspections Program for EOPs, is issued |
| 3/2/93 | PBNP informs NRC that mods will be completed for testing recirc valves and withdraws VRR-28 |
| 3/30/93 | Rev. 3 of IST deletes VRR-28 |
| 4/93 | DBD-01 validation considers worst-case flow (discharge and recirc valves closed) outside design and licensing basis |
| 6/30/93 | Revision 0 of IPE PRA model is issued |
| 4/4/94 | DBD-01, AFW System, Revision 0 is issued |
| ~1995 | Affects of excessive AFW flow introduced into EOPs |
| 4/15/97 | MR 97-038*A/B MDAFP discharge pressure control valve backup nitrogen supply and cable separation modifications issued |
| 6/97 | Update to FSAR adding AFW recirc feature for 3 minute closure on pump start |
| 9/26/97 | AOP-5B, Revision 11 issued that moved time critical steps from appendices to main body of the procedure |
| 9/30/97 | Revision 1C of WOG ERGs issued |
| 10/15/97 | CR 97-3363 initiated on IST Program Design Basis for AFW Minimum Flow Recirculation Valves (closed 10/5/98) |
| 11/10/97 | INPO issues SEN 174 on Loss of Nonvital Bus Causes Dual Unit Scram and Degraded AFW System (McGuire Units) |
| 1998 | Update to IPE PRA model is issued |
| 1/6/98 | Evaluation of SEN 174 completed – focus was on power supplies and did not address degradation of AFW recirculation valves |
| 6/98 | Update to FSAR adding detailed description of recirculation line function |
| 6/29/98 | CR 98-2575 (RCE 98-148) initiated on P-38A AFW Pump Recirc Valve Found Failed Shut |
| 9/30/98 | Rev. 5 of IST Program issued deleting testing of AFW recirc valves in the open direction |
| 5/24/99 | QCR 99-0115 initiated on Code Testing Conflict With the AFW Mini-flow Recirc Check Valves |
| 3/31/00 | DBD-01, AFW System, Revision 1 is issued |
| 9/11/00 | OE 10727 initiated on industry event involving PRA |

| DATE / TIME | DESCRIPTION |
|---|---|
| 7/6/01 | While revising the Probabilistic Risk Assessment (PRA) model for the Auxiliary Feedwater system, a potential procedural shortcoming was identified in AOP-5B, Loss of Instrument Air. Condition Report 01-2278 was originated to document the above finding |
| 7/10/01 | A CR action item #1 was created for Operations to move the step in AOP-5B, "Loss of Instrument Air," for gagging open the AFW minimum recirculation valves to an earlier location in the body of the procedure. (CR 01-2278) |
| 7/30/01 | Operations discussed issue with PRA group. PRA to run an evaluation to determine the significance of the issue. Analysis was expected to be completed by 8/20/01 (CR 01-2278) |
| 8/20/01 | The analysis is not ready yet. The evaluation is expected to determine the actual risk significance of the condition and address the type of actions that may be recommended. (CR 01-2278) |
| 10/19/01 | Per discussion with the PRA group, the PRA model is showing a higher risk and the recirculation valve should be procedurally addressed. The AOP is sequenced properly to address the loss of instrument air. PRA Group is requesting that the ARP for low instrument air pressure be changed to address this concern. This should be adequate rather than changing the sequence of the AOP. PRA will follow up with a procedure feedback. (CR 01-2278) |
| 10/24/01 | CR 01-2278 Action #1 was completed with direction to create a new action item to track issuance of a change to ARP C01 A 1-9 for low instrument air pressure. (CR 01-2278) |
| Early November, 2001 | Operations had discussions with PRA Group regarding whether procedure changes were adequate. |
| Week of Nov 13th, 2001 | PRA Group went to work to adjust the PRA model to evaluate the risk if the procedure change was not complete or would not be adequate. |
| 11/26/01 | Modeling adjustments were completed. A risk evaluation was done for the minimum recirculation valves. A factor of 2.3 risk increase was identified. This was considered high-risk significance. A discussion was held with Operations and Engineering . Decided we needed to determine what the scope of this was and what further actions may be appropriate. |
| 11/28/01 – 1300 | A meeting was held with Operations, Engineering and PRA personnel to discuss the significance and appropriate actions. The mechanistic details of the issue were well understood and developed by all present. The consensus was that this item represented a real possibility, and that it required further attention. Various possible actions were discussed, focusing primarily on enhancing Operator awareness of the system design, as well as modifications or procedural changes that may be desirable to eliminate it.<br>The subject of Operability was discussed during the meeting, and it was agreed that there was no operability concern because no equipment degradation, failure, or non-conformance had been identified. Regardless, the level of concern was great enough that further prompt action was felt justified. |

| DATE / TIME | DESCRIPTION |
|---|---|
| 11/28/01 – Late afternoon | The Operations manager had discussions with Engineering about this potential concern regarding significantly increased CDF risk resulting from an event where instrument air was lost and during the subsequent EOP actions, operators may take inappropriate action which could cause one or more AFW pumps to fail. |
| 11/29/01 – AM | Operations manager briefed the resident inspectors on the concerns of the issue and that we were evaluating the condition and risk. |
| 11/29/01 – Late AM | Following discussions with the staff SRO, operations concluded that use of temporary information tags and a briefing of all watch standers, would be an important step to reduce the risk of the event. We also started evaluating procedure changes that might help improve the safety of the plant and reduce the risk profile. |
| 11/29/01 – 10:00 | PRA briefed the STA and Shift Manager on the issue and discussed potential wording for control board placards. |
| 11/29/01 – 11:00 | PRA discussed potential reportability concerns with licensing. |
| 11/29/01 – 11:30 | PRA briefed the RI and provided estimated risk impact values. |
| 11/29/01 – 14:45 | CR 01-3595 documenting the increased risk was written. The CR was brought to the WCC and screened by an SRO. At that time, extensive discussion regarding whether an OD was required had already occurred, and extensive discussion on operability had occurred. My discussions with engineering and others focused on the fact that there was not an equipment problem, no equipment is degraded such that operability is in question, that this is a risk issue upon which we are relying on operator action to mitigate, and therefore, use of the OD was not appropriate. Those discussions were not captured in either the CR, or the associated screening. |
| 11/29/01 – 1520 | The oncoming crew was briefed and temporary information tags placed adjacent to the controls for 1/2P-29 and P-38A/B. This briefing summarized the concerns of this potential event. The temporary information tags provided a reminder that the minimum flow requirements for the AFW pumps are 50 GPM for the motor driven pumps and 75 GPM for the steam driven pumps. |
| 11/29/01 – 1553 | CR 01-3595 was screened by the WCC SRO (CR 01-3595) |
| 11/29/01 – 1700 | Operations Manager briefed Plant Manager on this issue. |
| 11/29/01 – 1705 | Event Notification 38525 made to NRC via ENS phone. |
| 11/30/01 – AM | Licensing manager received a call from the NRC-NRR backup PM concerning confusion over the event notification. A return conference call was made with engineering to address NRR questions. |

| DATE / TIME | DESCRIPTION |
|---|---|
| 11/30/01 – AM | Friday morning, after discussing this with the residents, Operations Manager concluded that to properly document the operability of the AFW system, we should initiate an operability determination to ensure the discussions we had the previous 24 hours regarding operability were properly documented. Engineering was requested to start on the OD. The Shift Manager was informed that an OD on the issue was being performing it and that it was expected to be completed mid to late afternoon. |
| 11/30/01 – Noon | Operations Manager met with Sr. Resident, Resident, and their supervisor to discuss situation. At that point NRC brought forward their concerns regarding whether AFW was operable in the condition that existed prior to Thursday afternoon and whether it was currently operable. The Plant Manager called NRC Region III along with the Operations Manager and had a discussion regarding operability of the system. |
| 11/30/01 – 1400 | Ran a simulator scenario to get information on plant response to a loss of offsite power coincident with a rapid loss of instrument air pressure. NOTE: Additional simulator scenarios were run on 11/30 and 12/1. |
| 11/30/01 – 1645 | Temporary procedure changes were completed to EOP-0 and EOP-0.1 to reflect the guidance provided earlier to operators on the temp info cards. |
| 11/30/01 - ~1700 | Plant Manager informed that a five-man incident investigation team would arrive on 12/3. |
| 11/30/01 – 1746 | A supplement to the Event Notification was provided to the NRC to clarify the discussion of the potential for an AFW failure as described in the original event notification 38525 |
| 11/30/01 - ~1830 | The OD was approved. This OD evaluated the current operability of the AFW system and included a discussion of the compensatory measures already taken to assure compliance with our licensing basis. |
| 12/1/01 – 0930 to 1200 | Staff meeting to prepare for NRC inspection team. |
| 12/1/01 – 1515 | Revision 1 to the Operability Determination was approved. The discussion of the AFW pump motor duty cycle was revised. |
| 12/3/01 – 0830 | CR 01-3595 screened as requiring an ACE. |
| 12/3/01 – 1000 to 1200 | Inspection Team meeting to prepare presentation for NRC entrance meeting. |
| 12/3/01 – 1200 | SVP and Plant Manager agree that CR 01-3595 requires a RCE. |
| 12/3/01 – 1400 | NRC Inspection Team has entrance meeting. |
| 12/4/01 | HEP expert onsite |
| 12/4/01 – 0700 | Initial RCE Team meeting held. |
| 12/4/01 – 1200 | Plant Manger approves RCE Charter. |
| 12/4/01 – 1620 | CR 01-3633 initiated on Appendix R concerns associated with MDAFW pump and LOOP and loss of IA and coincident fire. (CR 01-3633) |
| 12/5/01 – 1545 | CR 01-3648 initiated on response of MDAFW Pump to an Appendix R fire coincident with a LOOP and loss of IA. Potential existed for auto-start with discharge and recirc valves failed closed causing pump damage. (CR 01-3648) |
| 12/7/01 – 0900 | NRC Inspection Team has technical debrief. |

| DATE / TIME | DESCRIPTION |
|---|---|
| 12/13/01 – 1400 | NRC Inspection Team has exit meeting. |
| 12/14/01 | Permanent Revision to EOP-0 and EOP-0.1 implemented. |
| 12/20/01 | Additional revision made to EOP-0, EOP-0.1, and ECA-0.0 |

## *Attachment C: Why Staircase*

Problem: There is an increased CDF during a loss of instrument air scenario due to a common mode failure of all AFW pumps.

Why?: EOP-0.1 contains a step (step 1) to CONTROL feed flow because of RCS cool down considerations (RCS overcooling) and another step (step 4) to STOP feed flow to a steam generator if an increasing level cannot be maintained below the desired setpoint (S/G overfill) – these steps do not specify the method to be used to CONTROL or STOP flow. (It is postulated that an operator could throttle the AFW discharge valves closed and with a loss of instrument air when the recirculation valves are failed closed or fail closed later, the running pumps would dead-head and destroy themselves in a few minutes; a common mode failure.)

Problem: EOP-0.1 contains insufficient information to direct operators to take the correct actions for controlling AFW flow or stopping AFW flow to S/Gs under a loss of instrument air scenario.

Why1?: Reliance had previously been placed on AOP-5B for directing operator response to a loss of instrument air scenario; however, it was just recently recognized by the PRA group that action by operators would be required earlier in the scenario while still in EOP-0.1 (e.g., controlling S/G level without the availability of the AFW recirculation valves).

Problem1: The need for specific operator response actions for AFW flow control due to a loss of instrument air scenario while in EOP-0.1 was not previously identified.

Why1-1?: The original validation of EOP-0.1 did not evaluate the interaction between design, procedures and human error/timeline analysis. This analytical method was not available at that time. **(Human Error/Timeline Analysis Not Available)**

Why1-2?: The original PRA model did not model operator actions to control AFW flow in the system fault trees because it was assumed that there was a long time available and the function (S/G overfill) would be alarmed (assumption 13). The flaw in this assumption was not identified during the PRA model review because the fault trees were based primarily on functions described in design documents. Also, only operator actions taken to mitigate a failure were evaluated. The selection of the evaluation method using fault trees focused on design functions over other FMEA methods was based on an assumption that the design function approach was more conservative. The current PRA model review uses a methodology that integrates system performance with potential human

actions to obtain a spectrum of plant responses. **(PRA Model based on system functions)/(Only mitigating actions were evaluated)**

Why1-3?: The operator action to control AFW flow had not been identified as a human interaction with a human error probability assigned to it. **(Human Error/Timeline Analysis Not Available)**

Why2?: Previous evaluations of the effects of the AFW recirculation valves failing closed on loss of IA concluded that the AFW pumps would not be damaged because forward flow was always available. Closure of a single discharge valve due to component failure concurrent with the AFW recirculation valve failing closed was evaluated and considered to be outside the design and licensing basis. (This used NUREG-0800 assumptions and PBNP was not committed to that NUREG.) Closure of all the discharge valves due to operator action was not considered.

Problem2: Closure of the AFW discharge valves due to operator action was not previously considered as a possible failure mechanism.

Why2-1?: The consideration of human actions in failure modes and effects analyses has occurred primarily only in the PRA area and the integrated method of evaluating FMEA, human error probabilities, and timeline studies is a recent development. **(Human Error/Timeline Analysis Not Available)**

Why 2-2?: Insight was needed to understand that the actual operator response to a "CONTROL or STOP feed flow" command under a loss of instrument air scenario would be closure of the discharge valves instead of stopping the AFW pumps.

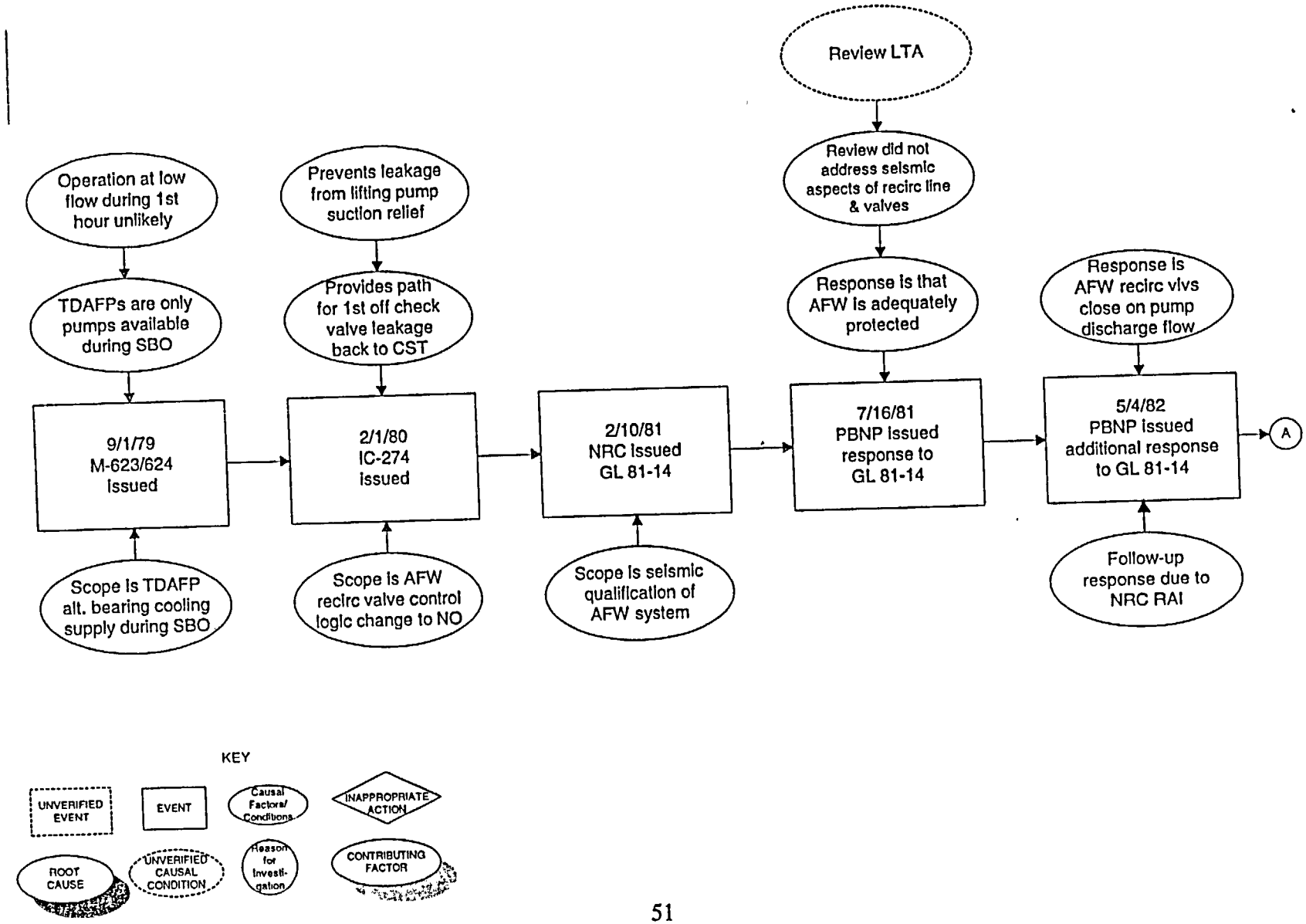
Problem: The expected operator response to the "CONTROL or STOP feed flow" command under a loss of instrument air scenario was not clear.

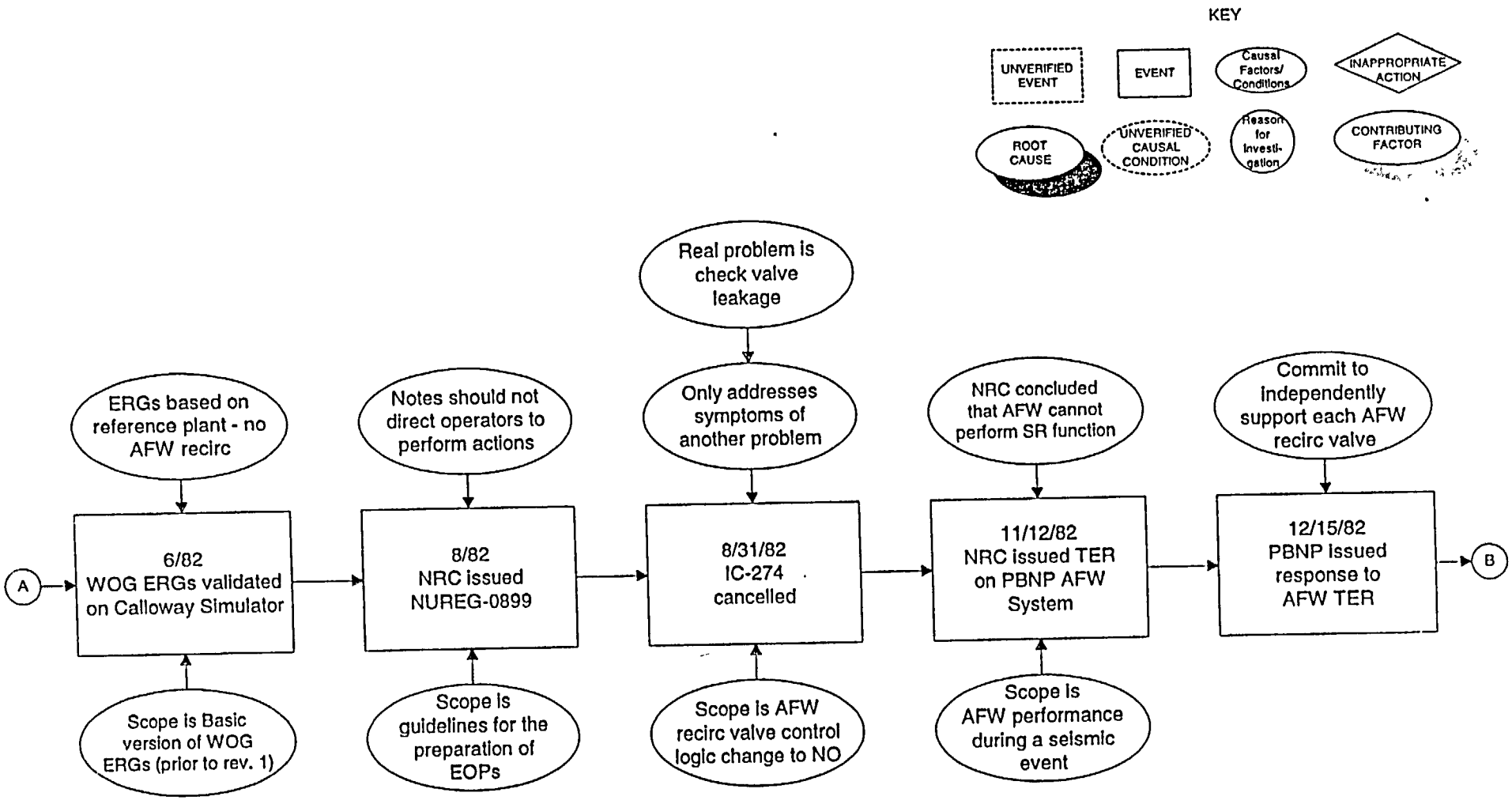
Why?: Training materials did not contain specific information on operator actions for controlling steam generator level (and AFW flow) under a loss of instrument air condition.

Problem: Training materials did not specify the actions required for successful control of AFW flow under loss of instrument air conditions.

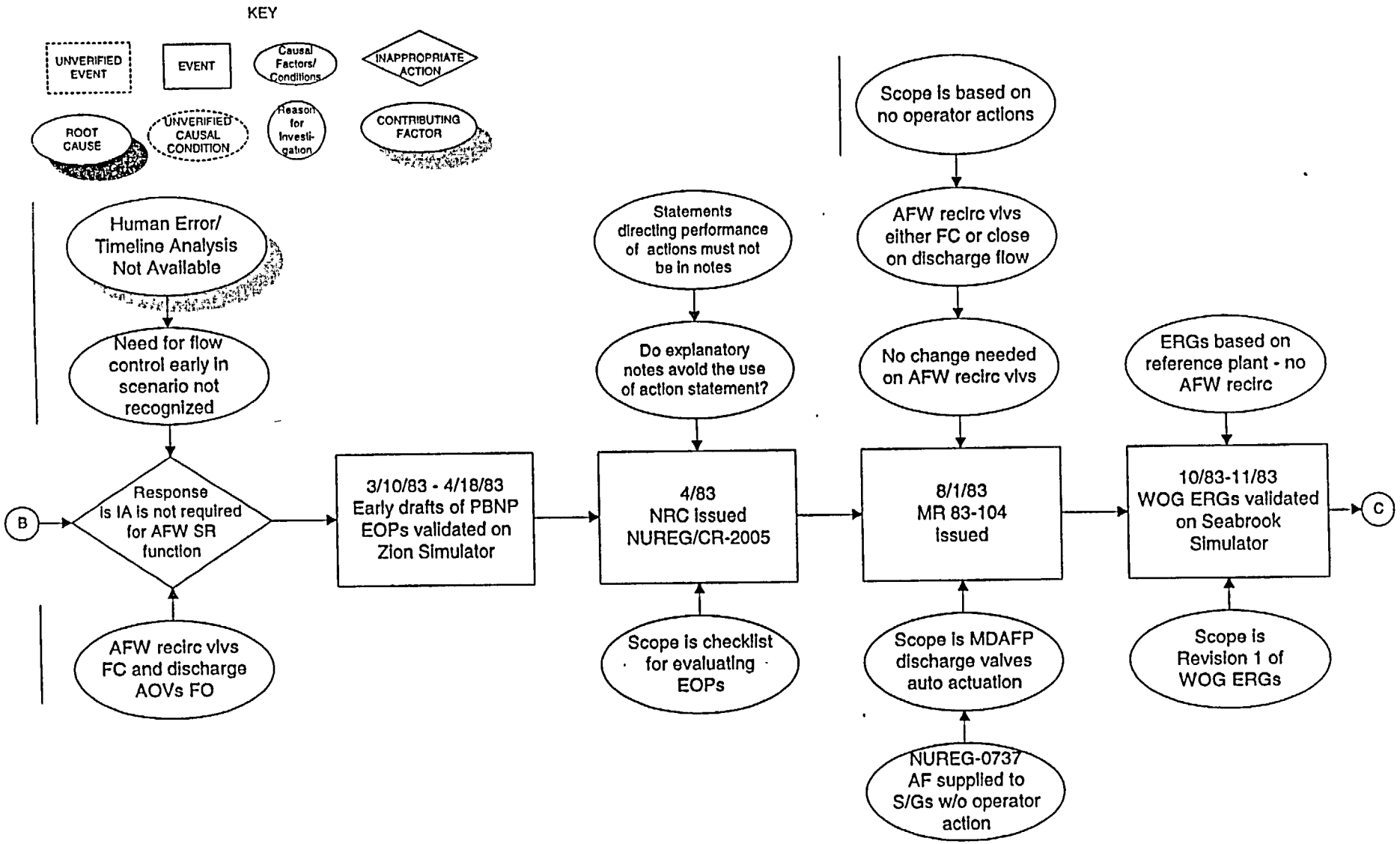
Why?: The importance of the AFW control evolution was not previously recognized. **(Human Error/Timeline Analysis Not Available)**

## Attachment D: Event & Causal Factor Chart




KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
A
ERGs based on reference plant - no AFW recirc
6/82 WOG ERGs validated on Calloway Simulator
Scope is Basic version of WOG ERGs (prior to rev. 1)
Notes should not direct operators to perform actions
8/82 NRC issued NUREG-0899
Scope is guidelines for the preparation of EOPs
Real problem is check valve leakage
Only addresses symptoms of another problem
8/31/82 IC-274 cancelled
Scope is AFW recirc valve control logic change to NO
NRC concluded that AFW cannot perform SR function
11/12/82 NRC issued TER on PBNP AFW System
Scope is AFW performance during a seismic event
Commit to independently support each AFW recirc valve
12/15/82 PBNP issued response to AFW TER
B

*In. ased CDF in AFW PRA Model Due to Procedural Inadequacies Related to Loss of Instrument Air*



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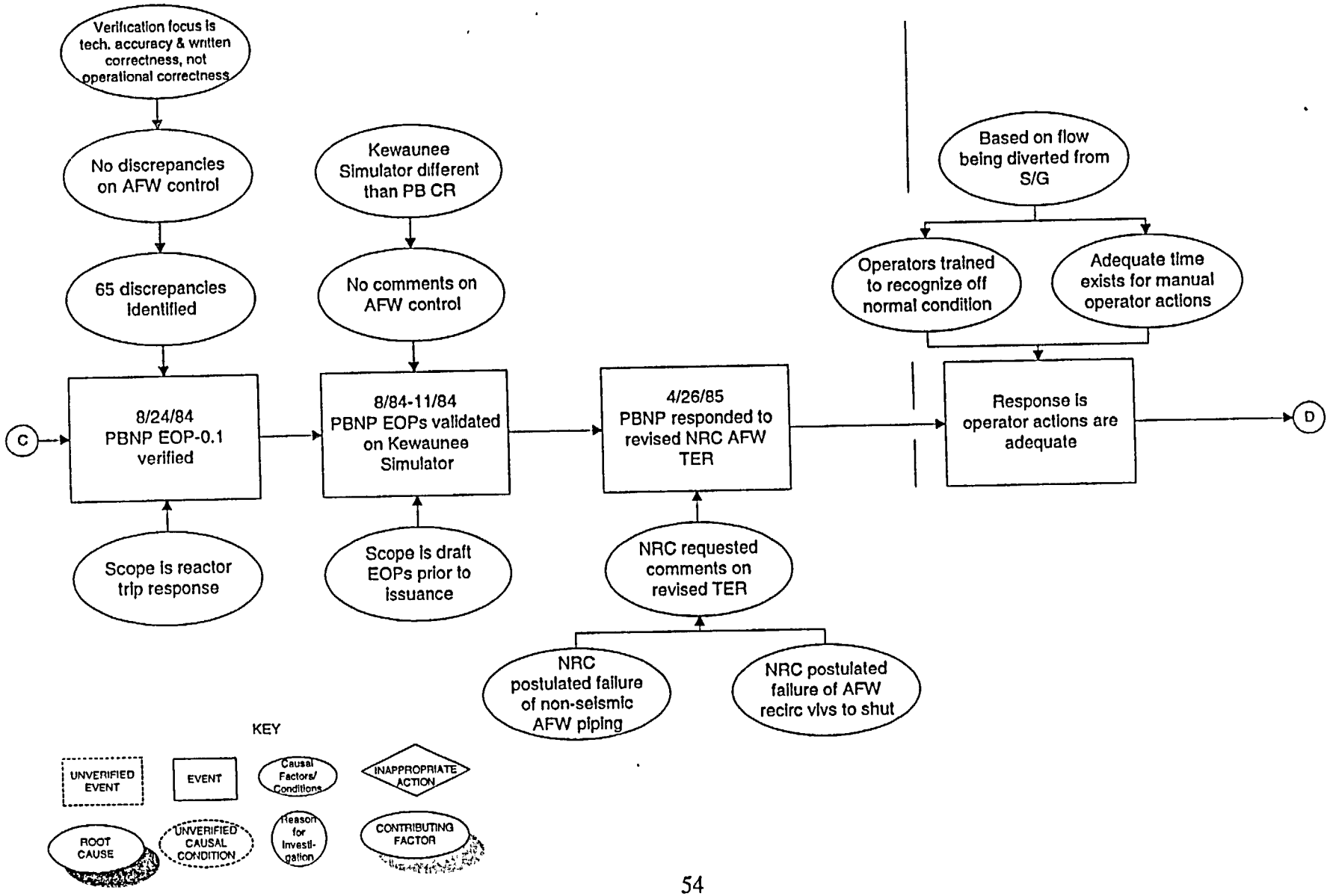

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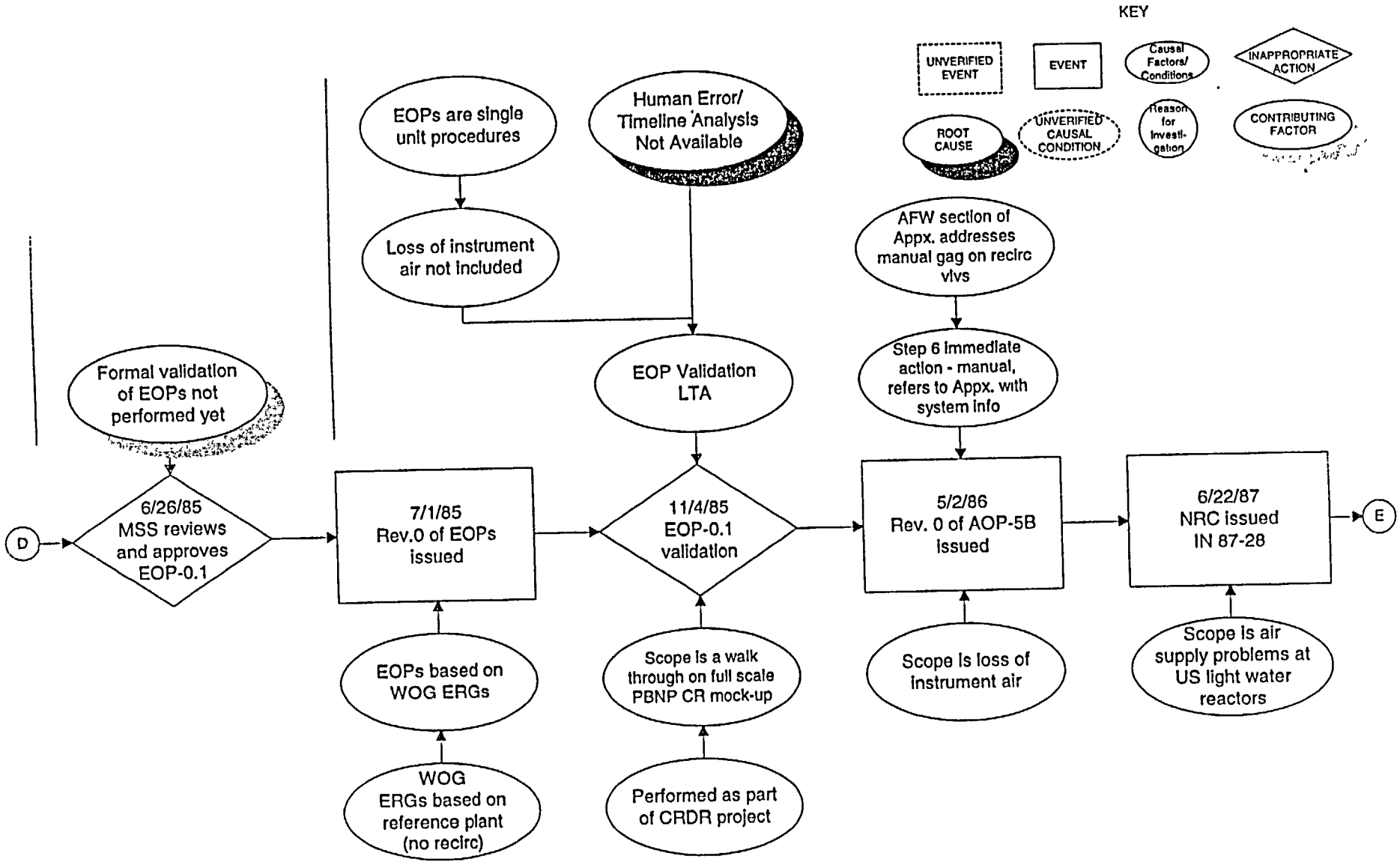
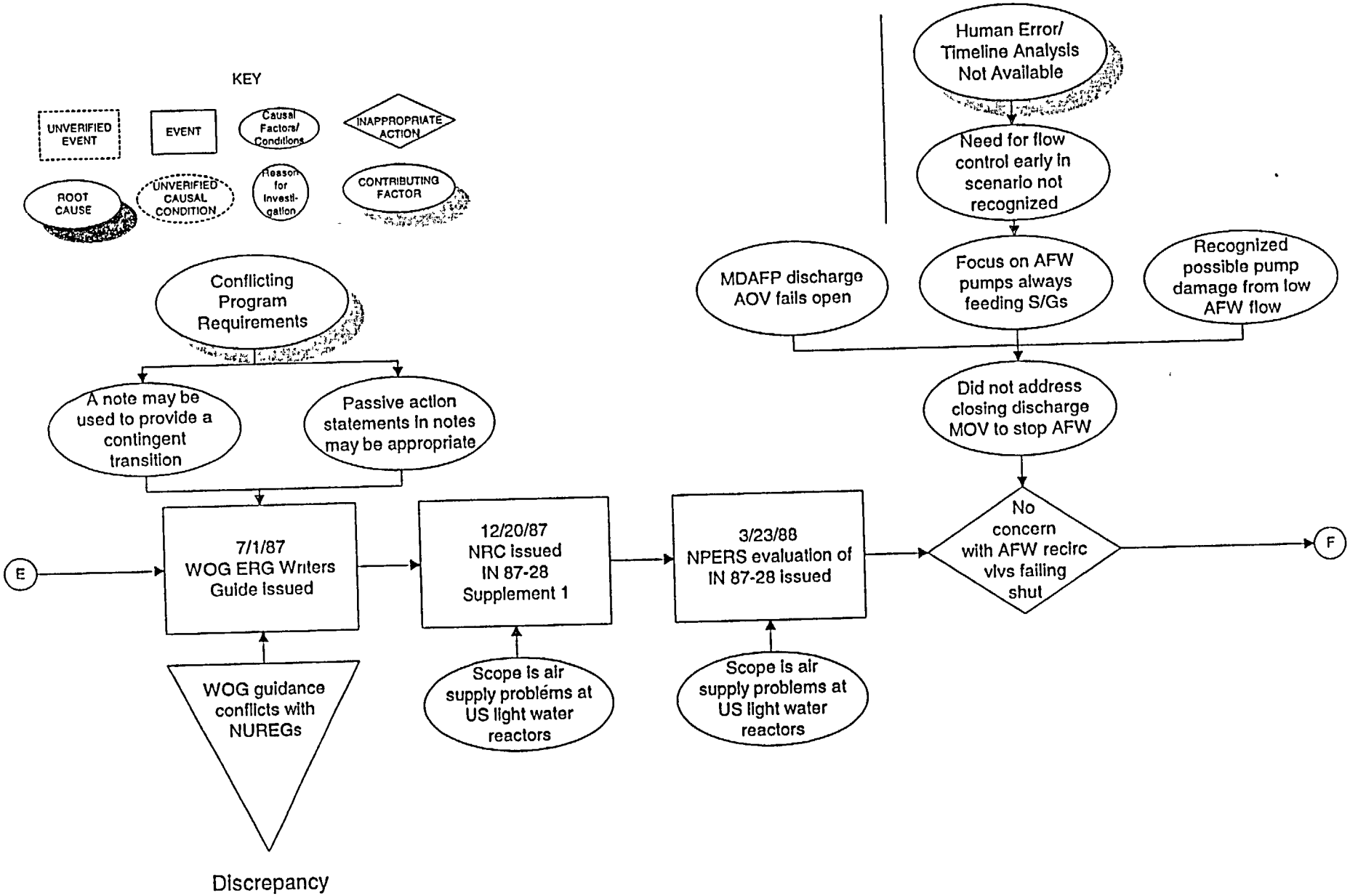
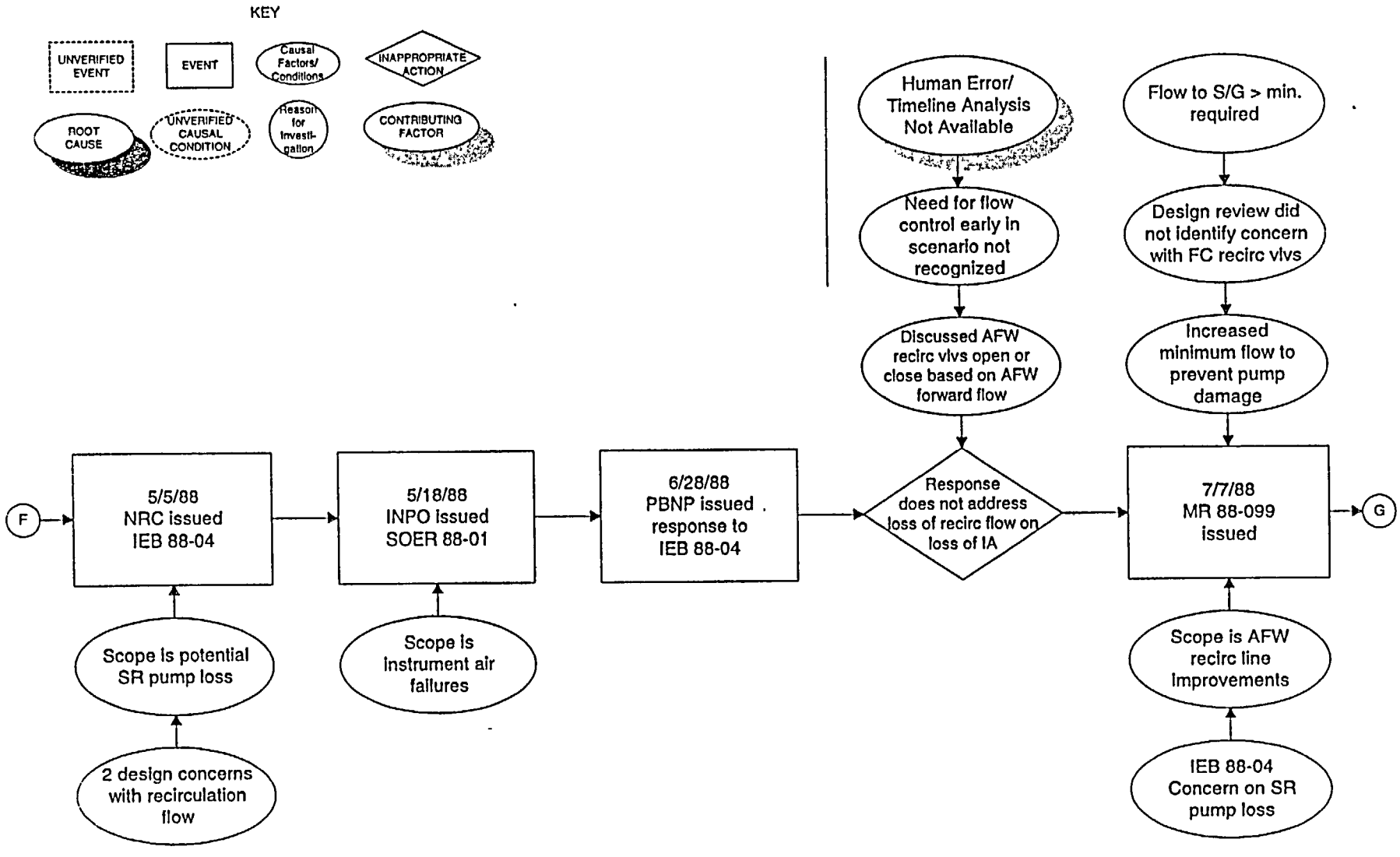
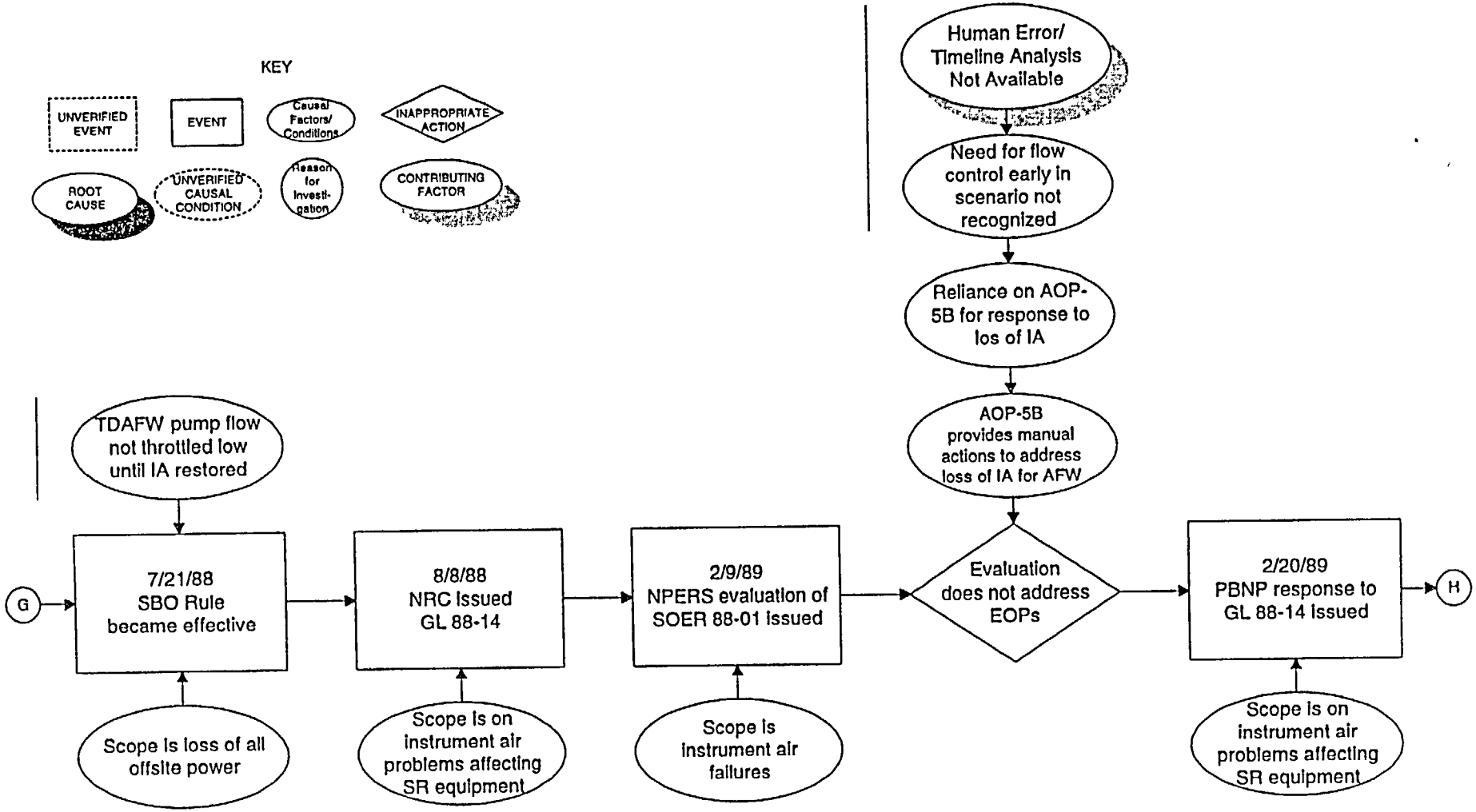


*Increased CDF in AFW PRA Model Due to Procedural Inadequacies Related to Loss of Instrument Air*




KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
F
5/5/88 NRC issued IEB 88-04
Scope is potential SR pump loss
2 design concerns with recirculation flow
5/18/88 INPO issued SOER 88-01
Scope is instrument air failures
6/28/88 PBNP issued response to IEB 88-04
Human Error/ Timeline Analysis Not Available
Need for flow control early in scenario not recognized
Discussed AFW recirc vlvs open or close based on AFW forward flow
Response does not address loss of recirc flow on loss of IA
Flow to S/G > min. required
Design review did not identify concern with FC recirc vlvs
Increased minimum flow to prevent pump damage
7/7/88 MR 88-099 issued
Scope is AFW recirc line improvements
IEB 88-04 Concern on SR pump loss
G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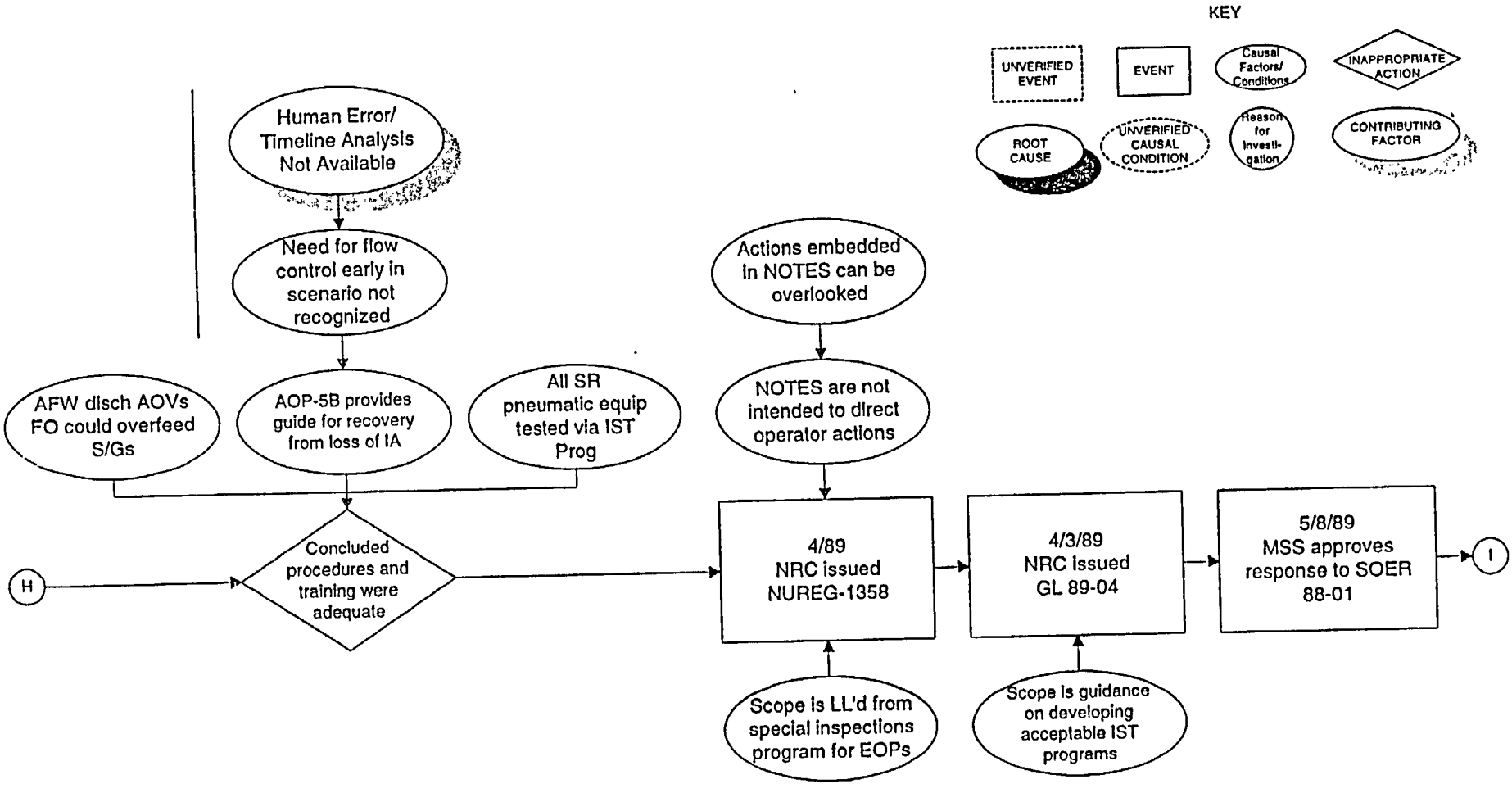
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
Human Error/ Timeline Analysis Not Available
Need for flow control early in scenario not recognized
Reliance on AOP-5B for response to los of IA
AOP-5B provides manual actions to address loss of IA for AFW
TDAFW pump flow not throttled low until IA restored
G
7/21/88 SBO Rule became effective
8/8/88 NRC Issued GL 88-14
2/9/89 NPERS evaluation of SOER 88-01 issued
Evaluation does not address EOPs
2/20/89 PBNP response to GL 88-14 issued
H
Scope is loss of all offsite power
Scope is on instrument air problems affecting SR equipment
Scope is instrument air failures
Scope is on instrument air problems affecting SR equipment

*Inc ;ed CDF in AFW PRA Model Due to Procedural Inadequacies Related to Loss of Instrument Air*



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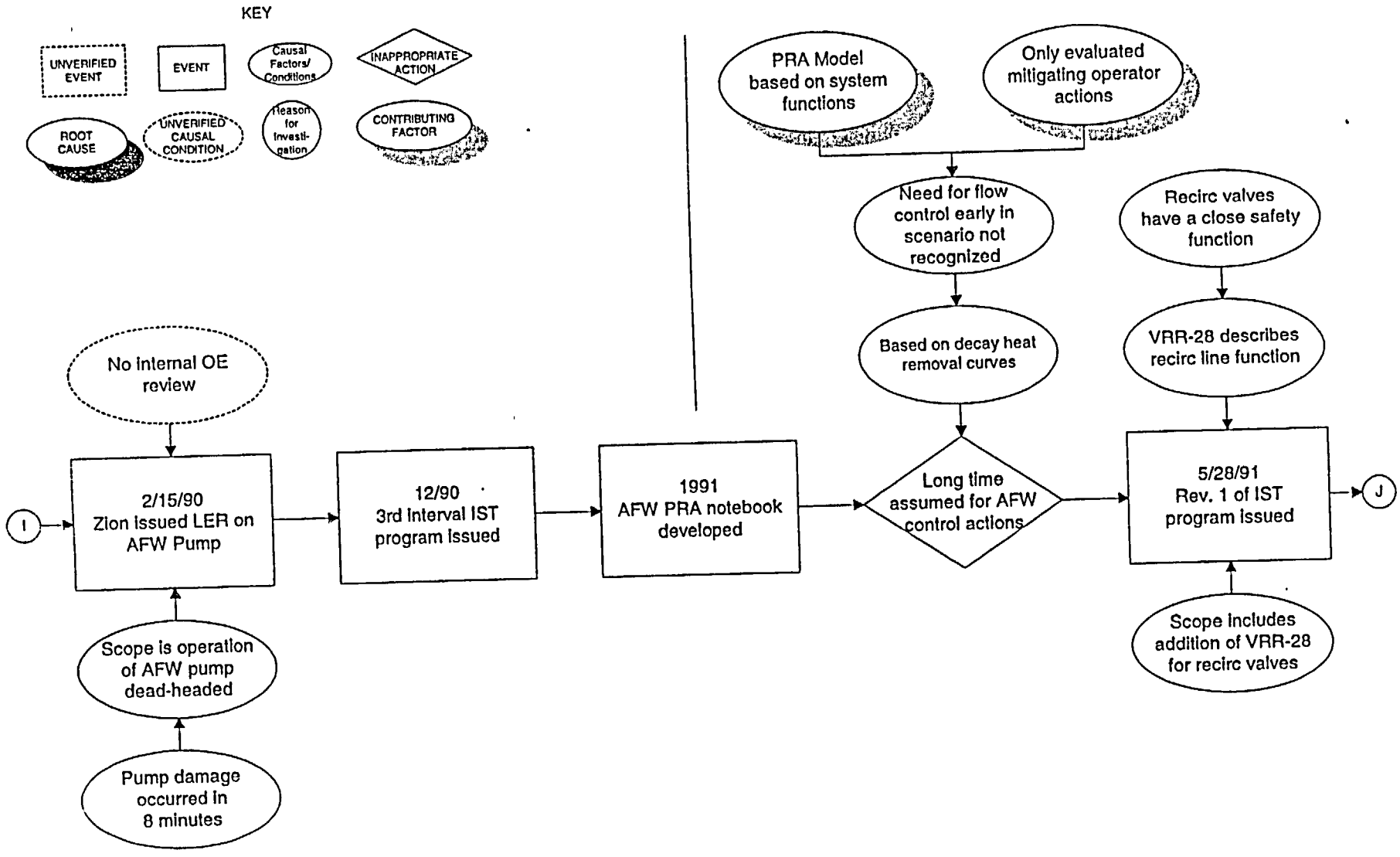

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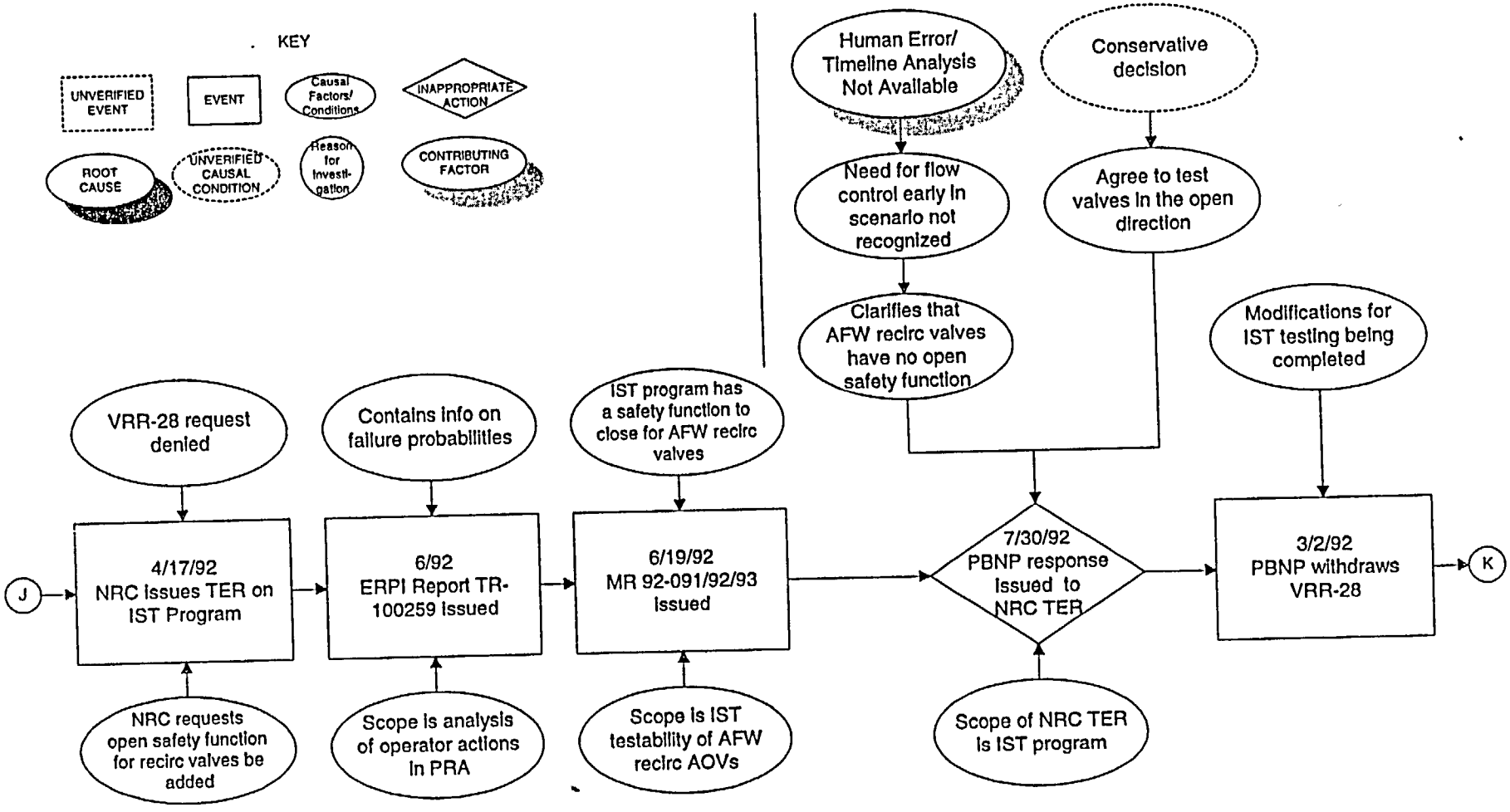
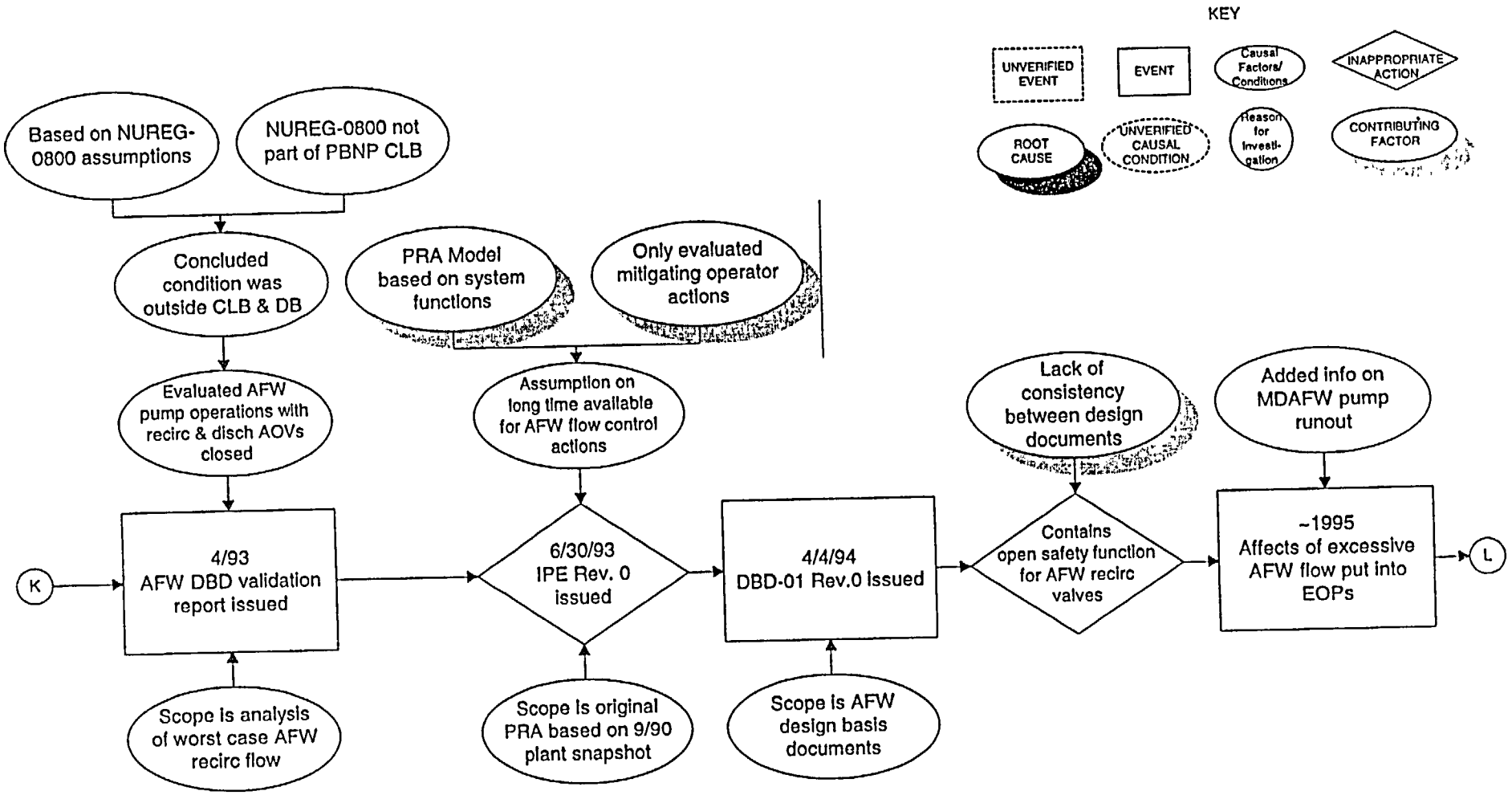
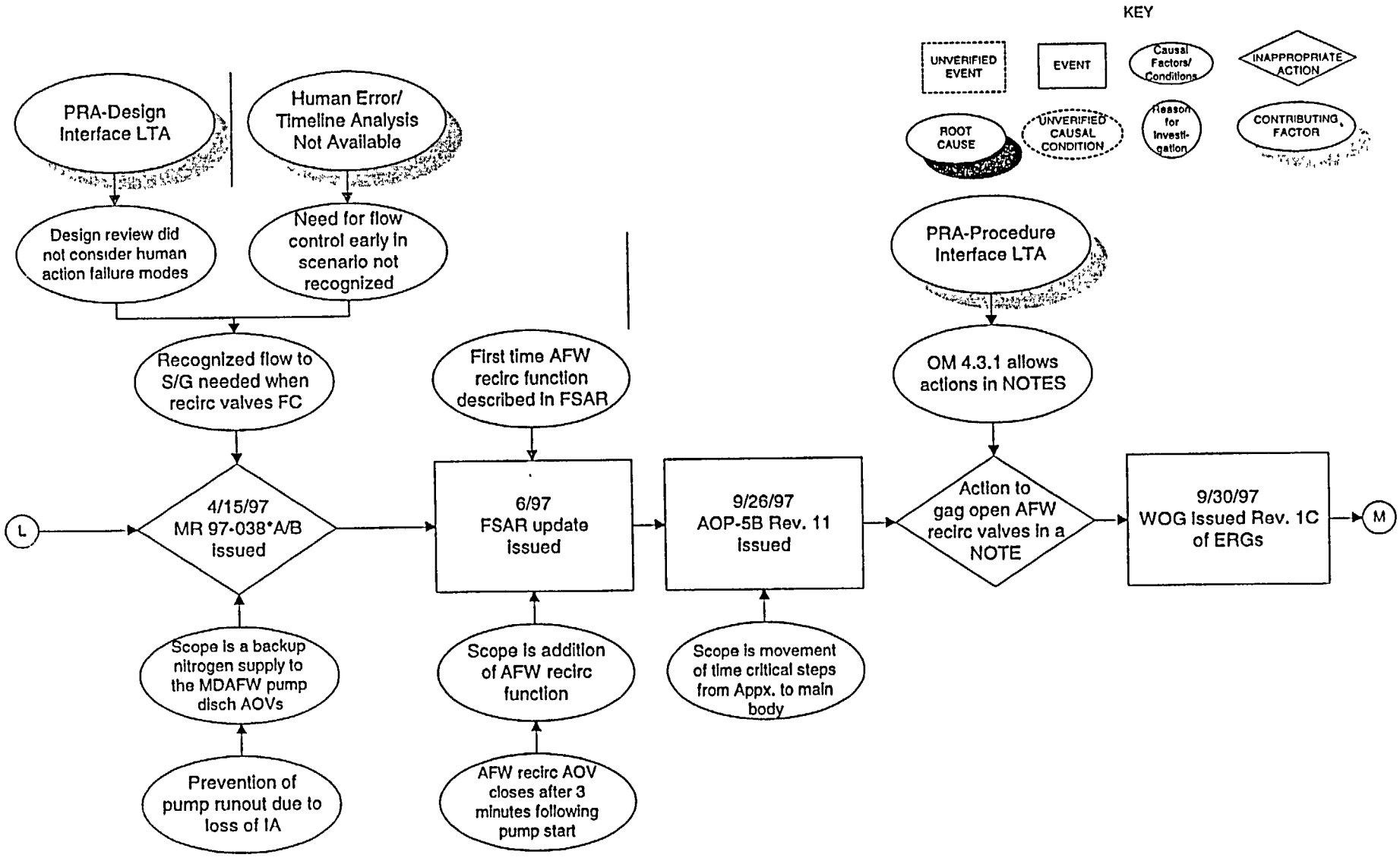


*Increased CDF in AFW PRA Model Due to Procedural Inadequacies Related to Loss of Instrument Air*

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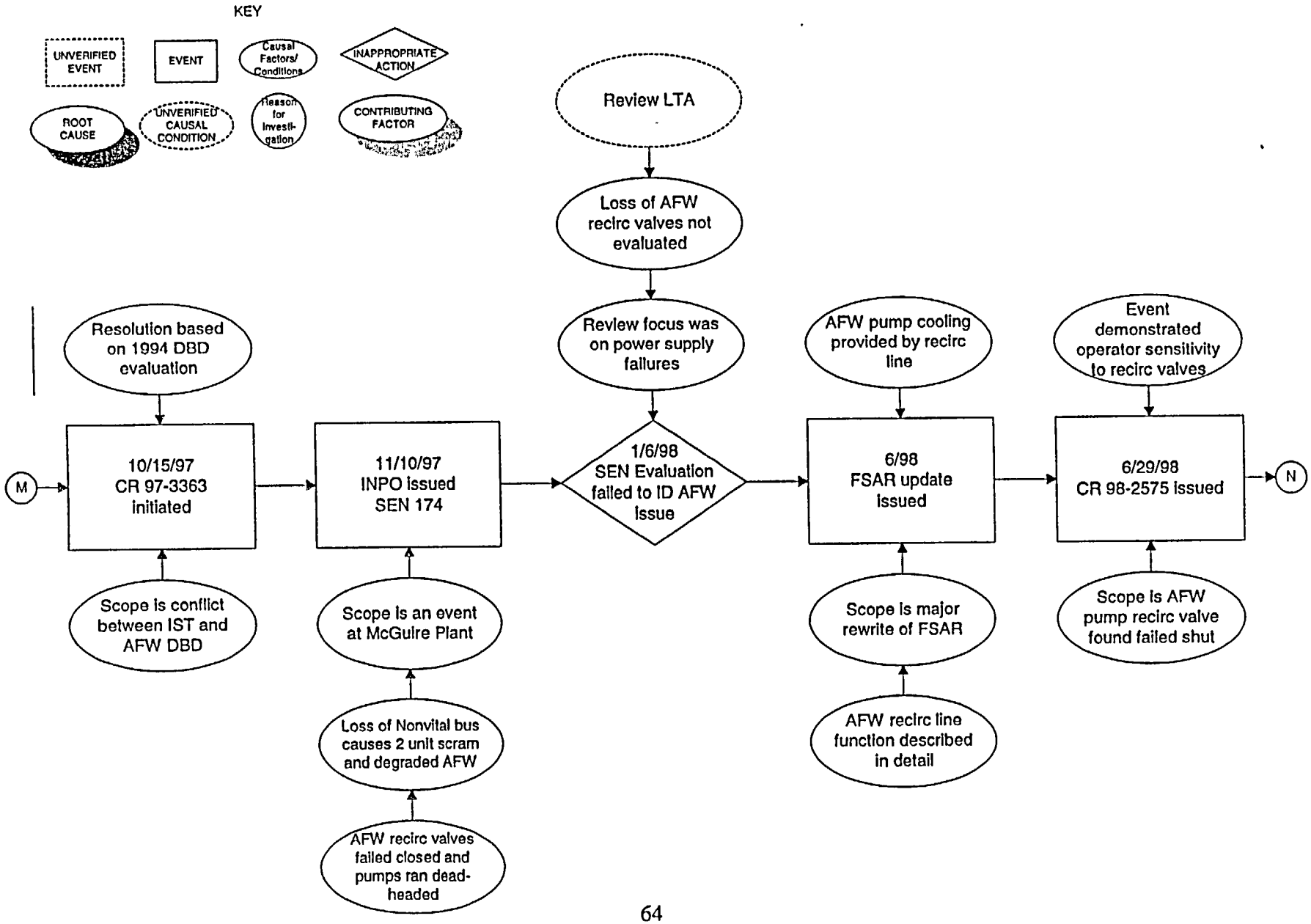
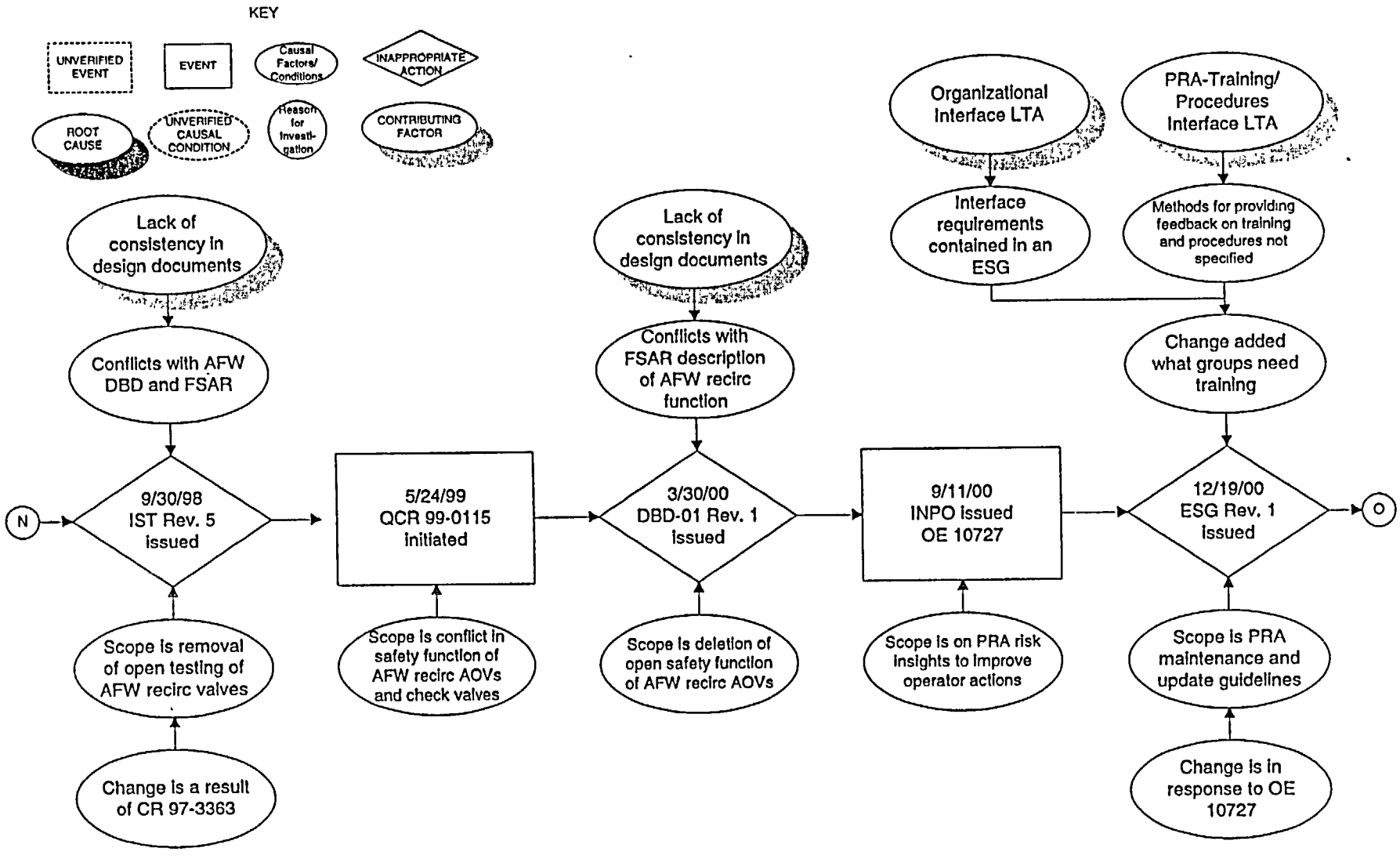
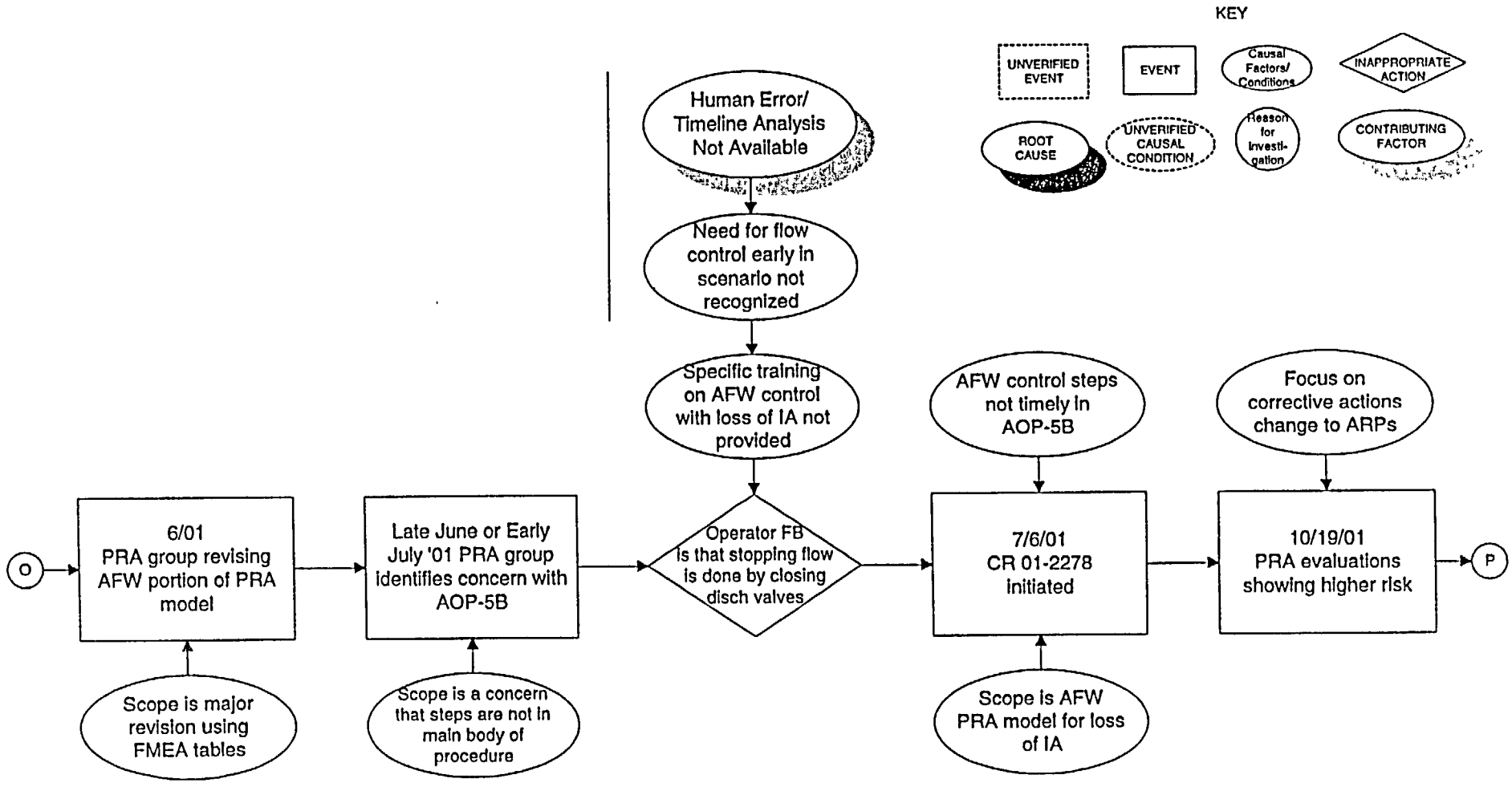
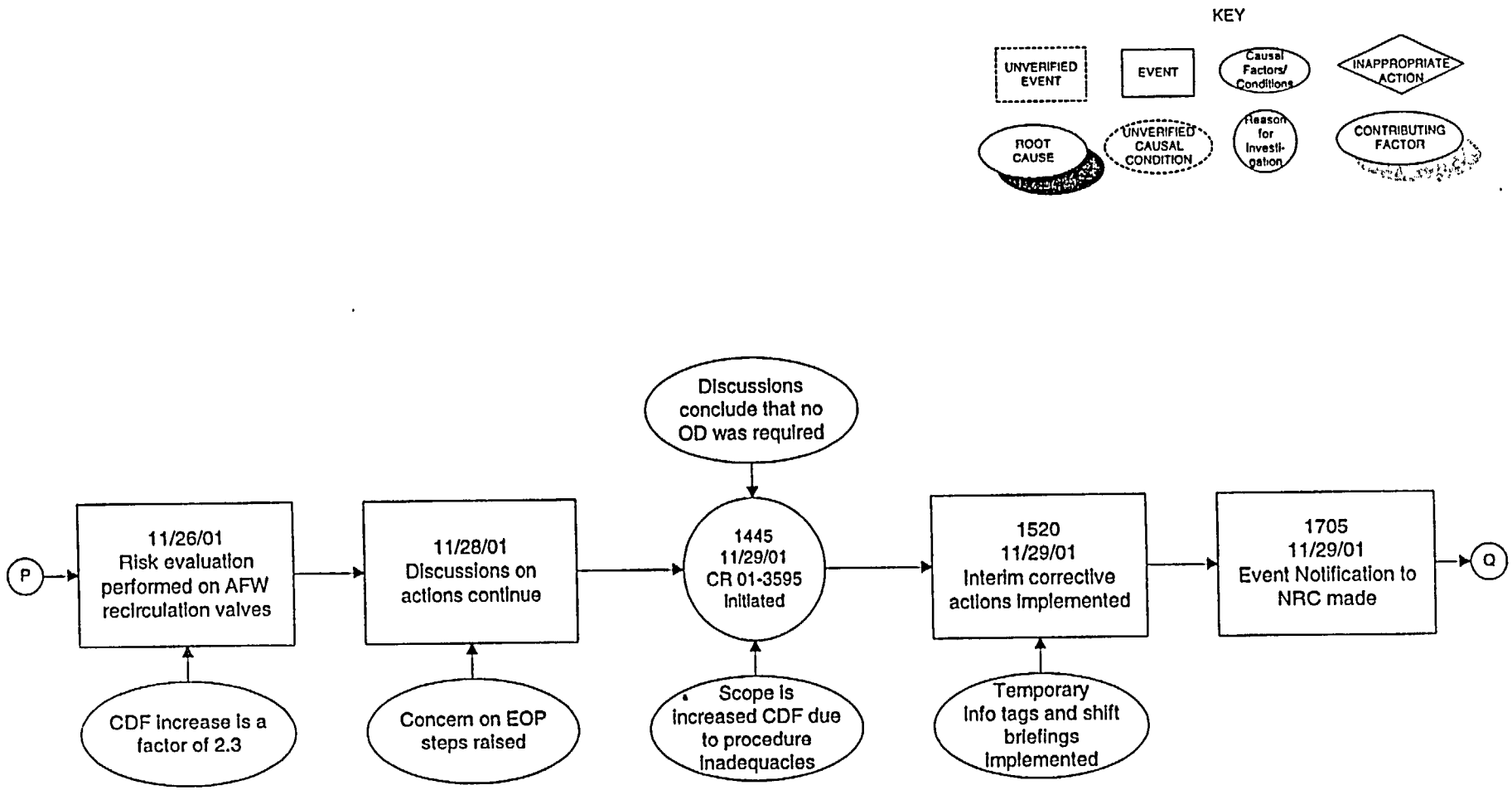
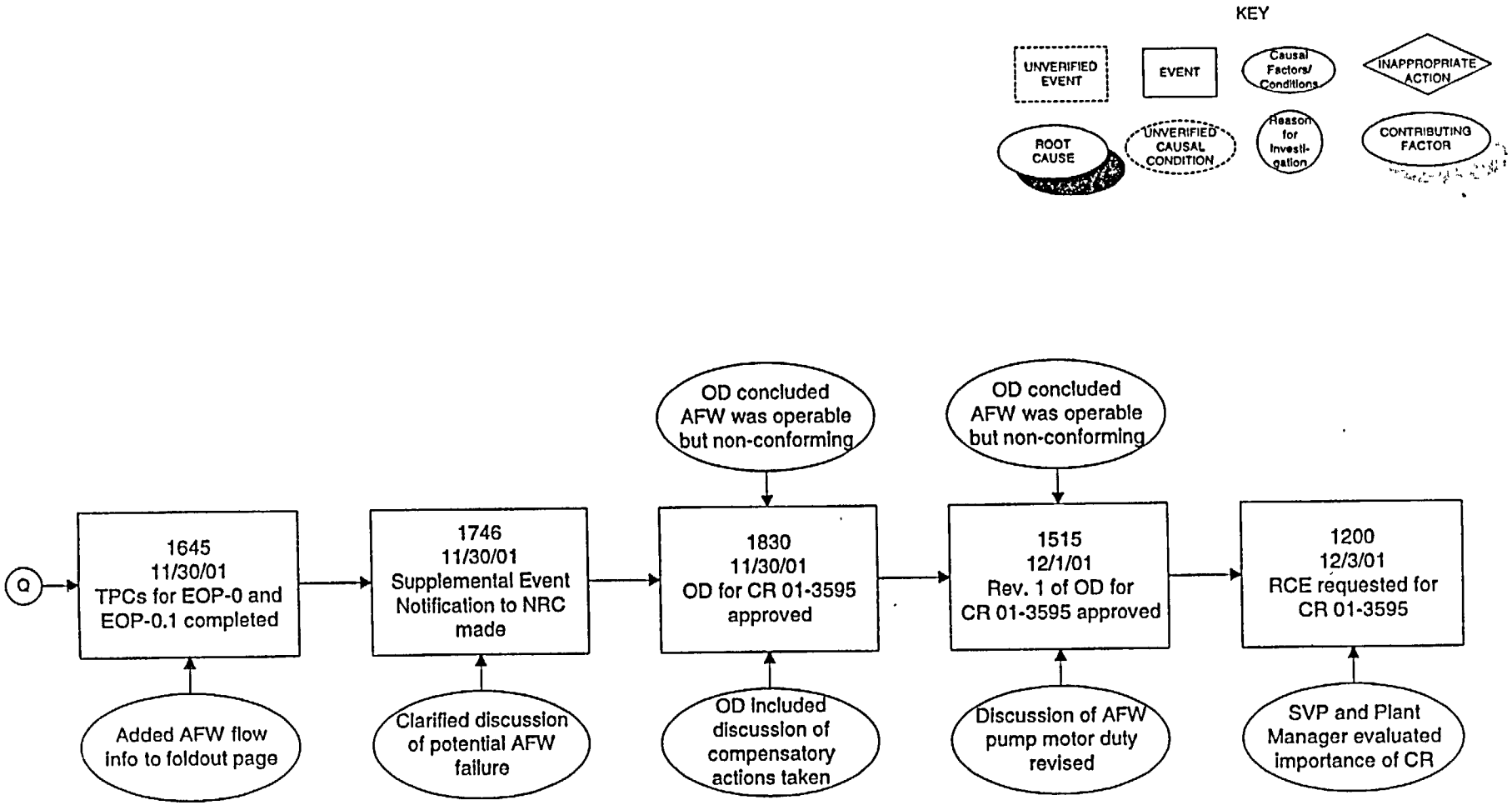
CONTRIBUTING FACTOR
PRA-Design Interface LTA
Human Error/ Timeline Analysis Not Available
Design review did not consider human action failure modes
Need for flow control early in scenario not recognized
Recognized flow to S/G needed when recirc valves FC
First time AFW recirc function described in FSAR
PRA-Procedure Interface LTA
OM 4.3.1 allows actions in NOTES
L
4/15/97 MR 97-038*A/B issued
6/97 FSAR update issued
9/26/97 AOP-5B Rev. 11 issued
Action to gag open AFW recirc valves in a NOTE
9/30/97 WOG issued Rev. 1C of ERGs
M
Scope is a backup nitrogen supply to the MDAFW pump disch AOVs
Scope is addition of AFW recirc function
Scope is movement of time critical steps from Appx. to main body
Prevention of pump runout due to loss of IA
AFW recirc AOV closes after 3 minutes following pump start


KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
Review LTA
Loss of AFW recirc valves not evaluated
Review focus was on power supply failures
1/6/98 SEN Evaluation failed to ID AFW issue
Resolution based on 1994 DBD evaluation
M
10/15/97 CR 97-3363 initiated
Scope is conflict between IST and AFW DBD
11/10/97 INPO issued SEN 174
Scope is an event at McGuire Plant
Loss of Nonvital bus causes 2 unit scram and degraded AFW
AFW recirc valves failed closed and pumps ran dead-headed
AFW pump cooling provided by recirc line
6/98 FSAR update issued
Scope is major rewrite of FSAR
AFW recirc line function described in detail
Event demonstrated operator sensitivity to recirc valves
6/29/98 CR 98-2575 issued
Scope is AFW pump recirc valve found failed shut
N


KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
Lack of consistency in design documents
Conflicts with AFW DBD and FSAR
N
9/30/98 IST Rev. 5 issued
Scope is removal of open testing of AFW recirc valves
Change is a result of CR 97-3363
5/24/99 QCR 99-0115 initiated
Scope is conflict in safety function of AFW recirc AOVs and check valves
Lack of consistency in design documents
Conflicts with FSAR description of AFW recirc function
3/30/00 DBD-01 Rev. 1 issued
Scope is deletion of open safety function of AFW recirc AOVs
9/11/00 INPO issued OE 10727
Scope is on PRA risk insights to improve operator actions
Organizational Interface LTA
Interface requirements contained in an ESG
PRA-Training/ Procedures Interface LTA
Methods for providing feedback on training and procedures not specified
Change added what groups need training
12/19/00 ESG Rev. 1 issued
Scope is PRA maintenance and update guidelines
Change is in response to OE 10727
O


KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
O
6/01 PRA group revising AFW portion of PRA model
Scope is major revision using FMEA tables
Late June or Early July '01 PRA group identifies concern with AOP-5B
Scope is a concern that steps are not in main body of procedure
Human Error/ Timeline Analysis Not Available
Need for flow control early in scenario not recognized
Specific training on AFW control with loss of IA not provided
Operator FB is that stopping flow is done by closing disch valves
AFW control steps not timely in AOP-5B
7/6/01 CR 01-2278 initiated
Scope is AFW PRA model for loss of IA
Focus on corrective actions change to ARPs
10/19/01 PRA evaluations showing higher risk
P


KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
P
11/26/01 Risk evaluation performed on AFW recirculation valves
11/28/01 Discussions on actions continue
Discussions conclude that no OD was required
1445 11/29/01 CR 01-3595 Initiated
1520 11/29/01 Interim corrective actions implemented
1705 11/29/01 Event Notification to NRC made
Q
CDF increase is a factor of 2.3
Concern on EOP steps raised
Scope is increased CDF due to procedure inadequacies
Temporary info tags and shift briefings implemented


KEY
UNVERIFIED EVENT
EVENT
Causal Factors/ Conditions
INAPPROPRIATE ACTION
ROOT CAUSE
UNVERIFIED CAUSAL CONDITION
Reason for Investi-gation
CONTRIBUTING FACTOR
Q
1645 11/30/01 TPCs for EOP-0 and EOP-0.1 completed
Added AFW flow info to foldout page
1746 11/30/01 Supplemental Event Notification to NRC made
Clarified discussion of potential AFW failure
OD concluded AFW was operable but non-conforming
1830 11/30/01 OD for CR 01-3595 approved
OD included discussion of compensatory actions taken
OD concluded AFW was operable but non-conforming
1515 12/1/01 Rev. 1 of OD for CR 01-3595 approved
Discussion of AFW pump motor duty revised
1200 12/3/01 RCE requested for CR 01-3595
SVP and Plant Manager evaluated importance of CR

NMC
Committed to Nuclear Excellence

# INTERNAL CORRESPONDENCE

NPM 2002-0495

**To:** CARB Members

**From:** Richard Flessner

**Date:** September 16, 2002

**Subject:** Addendum to RCE 01-069 Rev.1/ACE000314

**Copy To:** S. J. Nikolai S. A. Pfaff L. J. Peterson File

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The attached addendum to RCE 01-069 Rev.1/ACE000314 is being submitted for CARB review and approval. This addendum is being created to provide a more complete documentation record of items related to RCE 01-069 Rev. 1. The focus of the addendum is primarily on actions taken after the RCE was completed and accepted by CARB. A revision to the RCE is not deemed necessary because the basic conclusions and resulting recommended actions have not changed. Additional discretionary actions have been implemented by NMC and are being included in the addendum for a more complete record.

Attachment

# Addendum to RCE 01-069 Rev.1/ACE000314

This addendum to RCE 01-069 Rev.1 (ACE000314) covers the following items:

1. Inaccuracy in RCE report regarding IST program testing
2. Comments on Independent Review of RCE Report
3. Addition of the Open Safety Function to the AFW recirculation valves
4. Creation of action items to document corrective actions described in RCE report
5. Expansion of Extent of Condition Review
6. Effectiveness Review

Reason for Addendum: This addendum is being created to provide a more complete documentation record of items related to RCE 01-069 Rev. 1. The focus of the addendum is primarily on actions taken after the RCE was completed and accepted by CARB. A revision to the RCE is not deemed necessary because the basic conclusions and resulting recommended actions have not changed. Additional discretionary actions have been implemented by NMC and are being included in the addendum for a more complete record.

## 1. Inaccuracy in RCE report regarding IST program testing

On page 23 of RCE 01-069, Rev. 1, a statement is made regarding the deletion of open testing of the AFW recirculation valves from the IST program as a result of the evaluation made for CR 97-3363. Additional review has determined that testing of the AFW recirculation valves was not deleted, and that time testing data exists for all 4 AFW recirculation valves during the period 1993 to 2002.

## 2. Comments on Independent Review of RCE Report

The independent review of the AFW RCE (CAP002612/CA004074) contained the following final conclusion:

"The following final conclusion is based upon the scope of the investigation as prescribed by the management team in the investigation charter. The RCE represents a high quality, detailed, integrated investigation into the problem statement described in the Team Charter. The report is well constructed and well written and allows a non-involved reader to understand the event and the investigation performed. The root cause is supported by the facts, evidence and failure modes identification. The corrective actions are appropriate for the scope of the investigation and will ensure higher quality EOP documents in the future. Questions regarding the adequacy of the overall scope of the investigation are contained in the main body of the report."

Specific issues discussed in the review are:

- Charter/scope of investigation does not investigate why the design allowed the recirculation valves to fail-closed on loss of instrument air and how this condition went uncorrected until discovered by the PRA review.

  *Comment: The fail-closed position was known and understood in the design and did NOT go uncorrected until discovered by the PRA review. What was not known was the timing of operator actions and the need for specific guidance in the EOPs. The problem was determined to be a procedural issue by PBNP and the NRC; hence the investigation scope was appropriate.*

- No corrective actions exist to ensure that similar components do not have the same failure mode.

  *Comment: Since there was not a problem with the failure mode of the valve, there was no need to evaluate similar components. All operator actions associated with a loss of instrument air condition were evaluated and determined to be appropriate.*

- Root cause may be too narrowly focused.

  *Comment: The RCE evaluated the mismatch between plant design and plant procedures It was determined that the revised procedures could adequately support the plant design. The cited violation is for a procedural problem and not a design issue; hence, the focus was appropriate.*

- Barrier analysis might also be used (in addition to E&CF charting) on the EOP development and validation process.

  Comment: This would be an enhancement. Since the EOPs have been through 3 major revisions by WOG and the current processes for verification and validation are different (and enhanced by corrective actions in the RCE), it was felt that no value would be added by an additional barrier analysis.

- Report does not discuss use of single failure analysis in deriving EOPs.

  *Comment: This comment was based on the misperception that the fail-closed mode of the recirculation valves was not correct. Single failure analysis would be in addition to the designed failure mode of the valve and would not have been applicable.*

- RCE did not address timeliness or effectiveness of CA program in bringing issue to management's attention (initial CR 01-2278 written 7/6/01).

  *Comment: This issue was discussed between the RCE investigator, his Manager and the PRA Group Lead during the RCE evaluation and determined to be appropriate based on the complexity of the issue, the involvement of operations, and risk associated with the issue at that time; therefore, no concern was identified in the final RCE. A statement of there being no problem was not added.*

- Was deletion of testing the recirculation valves (in the open direction) from the IST program a dropped or missed commitment?

  *Comment: Evaluation of this item has determined that time testing of the AFW recirculation valves in the open direction is occurring and has not been deleted.*

- RCE does not discuss how PBNP specific design differences were identified through the original EOP development process.

  *Comment: The report describes the EOP verification process in general terms and the results obtained. The verification was via an approved procedure and checklist. There were more than 2500 discrepancy sheets identified, which is ample evidence that specific plant differences were considered.*

- Is it a safety function for the recirculation valves to open?

  *Comment: The report clearly describes the plant's licensed position that there was no required OPEN safety function for the recirculation valves. The NMC decision to add the OPEN safety function was based on improving equipment reliability and reducing CDF risk.*

- Report does not discuss any findings regarding design configuration control differences.

  *Comment: The report identifies that there were inconsistencies between the FSAR, IST and DBD documents and initiated a corrective action to review the current versions for consistency. This was treated as a broke-fix issue since it was not a significant contributing cause to the event. The evaluator's perception of a design problem gave this issue more importance than warranted.*

- There is no discussion on how the PBNP design compares to other similar plants AFW design.

  *Comment: A review of other plants AFW designs was performed and the PBNP design was found to be fairly unique; since there was no design deficiency, the issue was not discussed in the RCE report.*

- The design change for adding pneumatic back-up supply to the recirculation valves is not identified as a corrective action in the RCE

  *Comment: This corrective action was added to Revision 1 of the RCE.*

### 3. Addition of the Open Safety Function to the AFW recirculation valves

During ongoing reviews of the AFW recirculation issue, NMC determined that there was increased nuclear safety benefit (improved reliability and reduced CDF risk) in the addition of an open safety function to the AFW recirculation valves beyond that credited by the pneumatic back-up supply modifications already installed. Therefore, modification MR 02-029 was initiated to add the open safety function to the AFW recirculation valves. This MR included removal of the internals of the AF-117 check valve to eliminate a common mode failure. The modification was accepted on 9/12/02.

4. **Creation of action items to document corrective actions described in RCE report**

RCE 01-069, Rev. 1 identifies the corrective actions already taken and those being implemented in section VIII of the report, beginning on page 37. T-track references had been provided for the actions being implemented, but not for all of the actions already completed. Subsequently, t-track records have been created to adequately document the completed actions discussed in the report. The following action items have been created:

- Interim Corrective Action #1 – CA026222
- Corrective Action #17 – CA026223
- Corrective Action #18 – CA026224
- Corrective Action #19 – CA026225

Other t-track items related to this event are:

- CA002592 – This item documents the review of the condition from a short-term Maintenance Rule risk monitoring perspective.
- CA002593 – This item documented the OD review of the condition.
- CA002594 –This item tracked issuance of the LER for this event.
- OTH003541 – This item tracked presentation of the completed RCE to CARB.
- CA003983 – This item brought closure documentation back for CARB review once CA003691, CA003692 and CA003693 were completed.
- OTH004389 – This item tracked revision of the RCE to reflect information gained during preparations for the NRC regulatory conference.
- OD Part 1 Rev 2 – This document is attached to the parent CAP001415 and documents the operability determination of the original condition.
- OTH004510 – This item tracks the correction of problems identified with some HEPs from the review performed under CA004388
- CAP012011/CE010138 (KNPP) – These items document KNPP's review of the industry OE notification issued for this event.

5. **Expansion of Extent of Condition Review**

The EOP weakness regarding controlling AFW flow was found during the PRA model update for the AFW system. The PRA model update involved a simultaneous review of plant design, procedures, failure modes and timing of operator actions. However, the update process is not specifically designed to identify procedural errors. Therefore, an alternate approach was developed that combined the elements of the effects of a loss of support component function, the procedures that deal with resolving this function, and the timing of required actions. CAP029344 has been initiated to expand the extent of condition review for the AFW Red Finding using this alternate approach to provide an additional level of assurance that similar issues do not exist in other emergency procedures.

6. **Effectiveness Review**

T-track action item CA003983 was created following the CARB Meeting on 3/5/02 to bring back closure documentation for review at a CARB Meeting once CATPRs 1 and 2 (CA003691 and CA003962), and corrective action #1 (CA003693) were completed. CA003693 is associated with the overall PRA update project, which now has an approved action plan that extends to the end of 2004. It is recommended that the scope of CA003983 be modified to be an effectiveness review of the completed CATPRs as normally performed on RCEs

## STATE CHANGE HISTORY

| Initiate | Assign Work | Assign | Conduct Work | Work Complete | Review & Approval | Approved | Quality Check |
|---|---|---|---|---|---|---|---|
| by RICHARD FLESSNER | 9/3/2002 6 19 55 PM Owner RICHARD FLESSNER | by RICHARD FLESSNER | 9/3/2002 6 22 15 PM Owner RICHARD FLESSNER | by RICHARD FLESSNER | 9/3/2002 6:24:39 PM Owner RICHARD FLESSNER | by RICHARD FLESSNER | 9/3/2002 6.25.37 PM Owner PBNP CAP Admin |

## SECTION 1

| | | | |
|---|---|---|---|
| **Activity Request Id:** | CA026222 | | |
| **Activity Type:** | Corrective Action | **Submit Date:** | 9/3/2002 6:19:55 PM |
| **Site/Unit:** | Point Beach - Common | | |
| **Activity Requested:** | Interim CA#1: Revise EOP-0, EOP-0.1 and ECA-0.0 to address AFW control under loss of instrument air conditions. | | |
| **CATPR:** | N | **Initiator:** | MASTERLARK, JAMES |
| **Initiator Department:** | EPN Engineering Programs Nuclear Safety Analysis PB | **Responsible Group Code:** | EXC Engineering Processes Continuous Improvement PB |
| **Responsible Department:** | Engineering | **Activity Supervisor:** | RICHARD FLESSNER |
| **Activity Performer:** | RICHARD FLESSNER | | |

## SECTION 2

| | | | |
|---|---|---|---|
| **Priority:** | 3 | **Due Date:** | 9/3/2002 |
| **Mode Change Restraint:** | (None) | **Management Exception From PI?:** | N |
| **QA/Nuclear Oversight?:** | N | **Licensing Review?:** | N |
| **NRC Commitment?:** | N | **NRC Commitment Date:** | |

## SECTION 3

**Activity Completed:** 9/3/2002 6:24.39 PM - RICHARD FLESSNER
The following documentation was reviewed and demonstrates completion of this item as stated in the RCE:
EOP-0 Revisions
- TCN 2001-0871 was approved on 11/30/01 for Rev. 35 of Unit 1 EOP-0 to add AFW minimum flow requirements to the Fold Out Page The TCN was incorporated into a permanent change issued as Rev. 36 on 12/14/01.
- TCN 2001-0872 was approved on 11/30/01 for Rev. 36 of Unit 2 EOP-0 to add AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 37 on 12/14/01.
- TCN 2001-0915 was approved on 12/20/01 for Rev. 36 of Unit 1 EOP-0 to add reference to the IA HEADER LOW PRESSURE annunciator for AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 37 on 1/10/02.
- TCN 2001-0914 was approved on 12/20/01 for Rev. 37 of Unit 2 EOP-0 to add reference to

the IA HEADER LOW PRESSURE annunciator for AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 38 on 1/10/02.

EOP-0.1
- TCN 2001-0873 was approved on 11/30/01 for Rev. 24 of Unit 1 EOP-0.1 to add AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 25 on 12/14/01.
- TCN 2001-0874 was approved on 11/30/01 for Rev. 23 of Unit 2 EOP-0.1 to add AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 24 on 12/14/01.
- TCN 2001-0916 was approved on 12/20/01 for Rev. 25 of Unit 1 EOP-0.1 to add reference to the IA HEADER LOW PRESSURE annunciator for AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 26 on 1/10/02.
- TCN 2001-0913 was approved on 12/20/01 for Rev. 24 of Unit 2 EOP-0.1 to add reference to the IA HEADER LOW PRESSURE annunciator for AFW minimum flow requirements to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 25 on 1/10/02.

ECA-0.0
- TCN 2001-0917 was approved on 12/20/01 for Rev. 29 of Unit 1 ECA-0.0 to add AFW minimum flow requirements and reference to the IA HEADER LOW PRESSURE annunciator to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 30 on 1/10/02.
- TCN 2001-0912 was approved on 12/20/01 for Rev. 30 of Unit 2 ECA-0.0 to add AFW minimum flow requirements and reference to the IA HEADER LOW PRESSURE annunciator to the Fold Out Page. The TCN was incorporated into a permanent change issued as Rev. 31 on 1/10/02.

## SECTION 4

| | | | |
|---|---|---|---|
| **QA Supervisor:** | (None) | **Licensing Supervisor:** | (None) |

## SECTION 5

| | | | |
|---|---|---|---|
| **Project:** | CAP Activities & Actions | | |
| **State:** | Quality Check | **Active/Inactive:** | Active |
| **Owner:** | PBNP CAP Admin | **AR Type:** | Parent |
| **Submitter:** | RICHARD FLESSNER | **Assigned Date:** | 9/3/2002 |
| **Last Modified Date:** | 9/3/2002 6:25:37 PM | **Last Modifier:** | RICHARD FLESSNER |
| **Last State Change Date:** | 9/3/2002 6:25.37 PM | **Last State Changer:** | RICHARD FLESSNER |
| **Close Date:** | | | |
| **One Line Description:** | Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW | | |
| **NUTRK ID:** | CR 01-3595 | | |
| **Child Number:** | 0 | | |
| **References:** | CR 01-2278<br>RCE 01-069<br>GOOD CATCH | | |
| **Update:** | This CA is being issued to document a completed corrective action. | | |
| **Import Memo Field:** | | | |

**OLD_ACTION_NUM:**

**Cartridge and Frame:**

## ATTACHMENTS AND PARENT/CHILD LINKS

Linked to ACE000314: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Subtask from CAP001415: Probabilistic Risk Assessment PRA For Auxiliary Feedwater System AFW

Nuclear Power Business Unit

# TEMPORARY CHANGE REVIEW AND APPROVAL

*Note. Refer to NP 1.2.3, Temporary Procedure Changes, for requirements*



## I - INITIATION

Doc Number EOP-0 Current Rev 35 Unit PB1 Temp Change No 2001-0871

Document Title REACTOR TRIP OR SAFETY INJECTION

Existing Effective Temporary Changes

Brief Description ADDED FOP ITEM TO ADDRESS AFW MINIMUM FLOW

(Identify specific changes on Form PBF-0026c, Document Review and Approval Continuation, and include with the package)

☒ Initiate PBF-0026h and include with the change.

Other documents required to be effective concurrently with the temporary change: NONE

Changes pre-screened according to NP 10.3.1? ☒ NO ☐ YES (If Yes, list references and criteria on PBF-0026c)(refer to NP 10.3.1)

Screening completed according to NP 10.3.1? ☐ NA ☒ YES

Safety Evaluation Required? ☒ NO ☐ YES (If Yes, a revision may be processed or final reviews and approvals shall be obtained before implementing)

Determine if the change constitutes a Change Of Intent to the procedure by evaluating the following questions.
(If any answers are YES, a revision may be processed or final reviews and approvals shall be obtained before implementing)

| Will the proposed change: | YES | NO |
|---|---|---|
| 1. Require a change to, affect or invalidate a requirement, commitment, evaluation or description in the Current Licensing Basis (as defined in NP 10.3.1)? | ☐ | ☒ |
| 2. Cause an increase in magnitude, significance or impact such that it should be processed as a revision? | ☐ | ☒ |
| 3. Delete or modify a prerequisite, initial condition, precaution, limitation or other steps that could have safety significance or affect the procedure's margin of safety? | ☐ | ☒ |
| 4. Delete QC hold points, Independent Verification or Concurrent Check steps without the related step(s) that require the performance also being deleted? | ☐ | ☒ |
| 5. Change Tech Spec or other regulatory acceptance criteria other than for re-baselining purposes? | ☐ | ☒ |
| 6. Require a change to the procedure Purpose or change the procedure classification? | ☐ | ☒ |

Initiated By (print/sign) [illegible] Date 11/30/01

## II - INITIAL APPROVAL

This change is correct and complete, can be performed as written, and does not adversely affect personnel or nuclear safety, or Plant operating conditions

Group Supervisor (print/sign) T. VANDERMOSCHI Date 11/30/01
(Cannot be the Initiator)

This change does not adversely affect Plant operating conditions (Safety Related procedures only)

Senior Reactor Operator (print/sign) Rick Merkas Date 11/30/01
(Cannot be the Initiator or Group Supervisor)

## III - PROCEDURE OWNER REVIEW

☒ Permanent ☐ One-time Use ☐ Expiration Date, Event or Condition:

☐ Hold change until procedure completed (final review and approval still required within 14 days of initial approval)

☐ QR/MSS Review NOT Required (Admin/NNSR only) ☒ QR Review Required ☐ MSS Review Required (Reference NP 1.6.5)

Procedure Owner (print/sign) [illegible] Date 11/30/01
This Change and supporting requirements correctly completed and processed

## IV - FINAL REVIEW AND APPROVAL

(Must be completed within 14 days of initial approval) (The Initiator, QR and Approval Authority shall be independent from each other)

QR/MSS (print sign) Terry N. Mount Date 11/30/01
Indicates 50.59/72.48 applicability assessed, any necessary screenings evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed

MSS Meeting No

Approval Authority (print/sign) D. Schoon Date 11-30-01

## V - REVISION INFORMATION FOR PERMANENT CHANGES

Post Typing Review (print sign) T. VANDERMOSCHI Date 12/5/01
Indicates temporary change(s) incorporated exactly as approved and no other changes made to document

Incorporated into Revision Number 36 Effective Date DEC 14 2001

REC'D DEC 17 2001

MICROFILMED

DEC 1 4 20

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL CONTINUATION



Doc Number EOP-0 Revision 35 Unit 1

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number 2001-0871

Description of Changes:

| Step * | Change/Reason |
|---|---|
| FOP | CHANGE: Added AFW minimum flow requirements for the AFW pumps<br>REASON: To prevent damage to the AFW pumps on a loss of instrument air due to the AFW pump mini-recirc valve failing shut with minimum flow through the pump is less than required to cool the pump |

Other Comments

* Note Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to reviewers



References NP 1.1.3 NP 1.2.3

Point Beach Nuclear Plant

# TEMPORARY CHANGE AFFECTED MANUAL LOCATION



Procedure Number EOP-0 Revision 35 Unit PB1

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number 2001-0871

## I - IMMEDIATELY AFTER INITIAL APPROVAL ON PBF-0026e (Non-Intent changes)
(after Final Approval if change of intent involved)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☐ Copy included in work package for field implementation. (WO No. _____ ) | |
| ☒ Copy filed in Control Room temp change binder (Operations only). | 11/30/01 |
| ☒ Original change package provided to GRG to obtain Procedure Owner Review (e.g., Owner review may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 11/30/01 |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |

Performed By (print and sign) Sherri Schwieg / Sherri Schwieg Date 11/30/01

## II - PROCEDURE OWNER REVIEW ON PBF-0026e
(may be performed by OA II, Procedure Writer, etc.)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☒ Copy sent to Document Control Distribution Lead for Master File. (Not required for one-time use change) | 12-3-01 |
| ☐ Copy filed in Group satellite file. (Not required for one-time use changes) | |
| ☐ Copy filed in Group one-time use file. | |
| ☒ Original Temp Change provided to DOS to obtain Final Approvals (e.g., final approval may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 11/30/01 |
| ☒ U1/U2 CR (3) | 11/30/01 |
| ☒ PAB | 12-3-01 |
| ☒ OPs Shop | ↓ |
| ☒ OPs Office | ↓ |
| ☒ Simulator (Trng OA II) | ↓ |
| ☐ | |

Performed By (print and sign) Sherri Schwieg / Date 11/30/01

Title of Proposed Activity: Unit 1 EOP-0 - Rev. 35, Unit 2 EOP-0 - Rev. 36, Unit 1 EOP-0.1, Rev 24, Unit 2 EOP-0 1 - Rev 23

Associated Reference(s) #: CR 01-2278 Action 2

| | | | | |
|---|---|---|---|---|
| Prepared by: | Bob Wartenberg<br>Name ( Print) | Signature | Date | 11/30/01 |
| Reviewed by: | Clayton Graves<br>Name ( Print) | Signature | Date: | 11/30/01 |

**PART I (50.59/72.48) - DESCRIBE THE PROPOSED ACTIVITY AND SEARCH THE PLANT AND ISFSI LICENSING BASIS** (Resource Manual 5.3.1)

**NOTE: The "NMC 10 CFR 50.59 Resource Manual" (Resource Manual) and NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 10 CFR 50.59 and 10 CFR 72.48 screenings.**

I.1 Describe the proposed activity and the scope of the activity being covered by this screening. (The 10 CFR 50.59 / 72.48 review of other portions of the proposed activity may be documented via the applicability and pre-screening process requirements in NP 5.1.8.) Appropriate descriptive material may be attached.

A foldout-page item is being added to Units 1 & 2 procedures EOP-0 and EOP-0.1. The foldout page item, "AFW Minimum Flow Requirements", shall address minimum flow required by the AFW pumps in the case of a failed closed mini-recirc valve on any running AFW pumps.

I.2 Search the PBNP Current Licensing Basis (CLB) as follows: Final Safety Analysis Report (FSAR), FSAR Change Requests (FCRs) with assigned numbers, the Fire Protection Evaluation Report (FPER), the CLB (Regulatory) Commitment Database, the Technical Specifications (both Custom and Improved), the Technical Specifications Bases, and the Technical Requirements Manual. Search the ISFSI licensing basis as follows: VSC-24 Safety Analysis Report, the VSC-24 Certificate of Compliance, the CLB (Regulatory) Commitment Database, and the VSC-24 10 CFR 72.212 Site Evaluation Report. Describe the pertinent design function(s), performance requirements, and methods of evaluation for both the plant and for the cask/ISFSI as appropriate. Identify where the pertinent information is described in the above documents (by document section number and title). (Resource Manual 5 3.1 and NEI 96-07, App B, B 2)

FSAR 10.2, Auxiliary Feedwater System

I.3 Does the proposed activity involve a change to any Custom or Improved Technical Specification (ITS)? Changes to Technical Specifications require a License Amendment Request (Resource Manual Section 5.3.1.2).

Technical Specification Change : ☐ Yes ☒ No

If a Technical Specification change is required, explain what the change should be and why it is required

I.4 Does the proposed activity involve a change to the terms, conditions or specifications incorporated in any VSC-24 cask Certificate of Compliance (CoC)? Changes to a VSC-24 cask Certificate of Compliance require a CoC amendment request

☐ Yes ☒ No

If a storage cask Certificate of Compliance change is required, explain what the change should be and why it is required

---------- 10 CFR 50.59 SCREENING ----------

PART II (50.59) - DETERMINE IF THE CHANGE INVOLVES A *DESIGN FUNCTION* (Resource Manual 5.3.2)

Compare the proposed activity to the relevant CLB descriptions, and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☒ | ☐ | Does the proposed activity involve Safety Analyses or structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☒[1] | Does the proposed activity involve SSCs whose failure could initiate a transient (e.g., reactor trip, loss of feedwater, etc.) or accident, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve CLB-described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications? |
| ☐ | ☒ | Does the activity involve a *method of evaluation* described in the FSAR? |
| ☐ | ☒ | Is the activity a *test or experiment*? (i.e., a non-passive activity which gathers data) |
| ☐ | ☒ | Does the activity exceed or potentially affect a *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If THIS questions is answered YES, a 10 CFR 50.59 Evaluation is required.) |

If the answers to ALL of these questions are NO, mark Part III as not applicable, document the 10 CFR 50.59 screening in the conclusion section (Part IV), then proceed directly to Part V - 10 CFR 72.48 Pre-screening Questions.

If any of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved.

FSAR 10.2 states each AFW pump has an AOV controlled recirc line back to the CST to ensure minimum flow to dissipate heat. This change ensures the minimum AFW flow requirements will be maintained on any running AFW pump in the case of a failed shut AFW mini-recirc flow control valve.

PART III (50.59) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (Resource Manual 5.3.3)

If ALL the questions in Part II are answered NO, then Part III is ☐ NOT APPLICABLE.

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 50.59 Evaluation is required; EXCEPT where noted in Part III.3.

III.1 CHANGES TO THE FACILITY OR PROCEDURES

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☒ | Does the activity adversely affect the *design function* of an SSC credited in safety analyses? |
| ☐ | ☒ | Does the activity adversely affect the method of performing or controlling the *design function* of an SSC credited in the safety analyses? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion as necessary)

This change ensures that minimum recirc flow requirements as stated in FSAR 10.2 are not violated.

III.2 CHANGES TO A METHOD OF EVALUATION

(If the activity does not involve a method of evaluation, these questions are ☒ NOT APPLICABLE )

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in the CLB? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in the CLB? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary).

III.3 TESTS OR EXPERIMENTS

If the activity is not a test or experiment, the questions in III.3.a and III.3.b are ☒ NOT APPLICABLE.

a Answer these two questions first

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the CLB? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the facility? |

If the answer to BOTH questions in V.3 a is NO, continue to III.3.b. If the answer to EITHER question is YES, then describe the basis

b. Answer these additional questions ONLY for tests or experiments which do NOT meet the criteria given in III.3 a above. If the answer to either question in III.3.a is YES, then these three questions are ☐ NOT APPLICABLE.

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the CLB? |
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is inconsistent with the analyses or descriptions in the CLB? |
| ☐ | ☐ | Does the activity place the facility in a condition not previously evaluated or that could affect the capability of an SSC to perform its intended functions? |

If any answer in III 3 b is YES, a 10 CFR 50 59 Evaluation is required. If the answers in III.3 b are ALL NO, describe the basis for the conclusion (attach additional discussion as necessary).

**Part IV - 10 CFR 50.59 SCREENING CONCLUSION** (Resource Manual 5.3.4).

ıeck all that apply:

A 10 CFR 50.59 Evaluation is ☐ required or ☒ NOT required.

A Point Beach FSAR change is ☐ required or ☒ NOT required. If an FSAR change is required, then initiate an FSAR Change Request (FCR) per NP 5.2.6.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A Technical Specification Bases change is ☐ required or ☒ NOT required. If a change to the Technical Specification Bases is required, then initiate a Technical Specification Bases change per NP 5.2.15.

A Technical Requirements Manual change is ☐ required or ☒ NOT required. If a change to the Technical Requirements Manual is required, then initiate a Technical Requirements Manual change per NP 5.2.15.

---------- **10 CFR 72.48 SCREENING** ----------

**NOTE: NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 72.48 screenings.**

**PART V (72.48) - 10 CFR 72.48 INITIAL SCREENING QUESTIONS**

Part V determines if a full 10 CFR 72.48 screening is required to be completed (Parts VI and VII) for the proposed activity.

| YES | NO | QUESTION |
|---|---|---|
| | ☒ | Does the proposed activity involve IN ANY MANNER the dry fuel storage cask(s), the cask transfer/transport equipment, any ISFSI facility SSC(s), or any ISFSI facility monitoring as follows: Multi-Assembly Sealed Basket (MSB), MSB Transfer Cask (MTC), MTC Lifting Yoke, Ventilated Concrete Cask (VCC), Ventilated Storage Cask (VSC), VSC Transporter (VCST), ISFSI Storage Pad Facility, ISFSI Storage Pad Data/Communication Links, or PPCS/ISFSI Continuous Temperature Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) installed in the plant specifically added to support cask loading/unloading activities, as follows: Cask Dewatering System (CDW), Cask Reflood System (CRF), or Hydrogen Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) needed for plant operation which are also used to support cask loading/unloading activities, as follows: Spent Fuel Pool (SFP), SFP Cooling and Filtration (SF), Primary Auxiliary Building Ventilation System (VNPAB), Drumming Area Ventilation System (VNDRM), RE-105 (SFP Low Range Monitor), RE-135 (SFP High Range Monitor), RE-221 (Drumming Area Vent Gas Monitor), RE-325 (Drumming Area Exhaust Low-Range Gas Monitor), PAB Crane, SFP Platform Bridge, Truck Access Area, or Decon Area? |
| ☐ | ☒ | Does the proposed activity involve a change to Point Beach CLB design criteria for external events such as earthquakes, tornadoes, high winds, flooding, etc.? |
| ☐ | ☒ | Does the activity involve plant heavy load requirements or procedures for areas of the plant used to support cask loading/unloading activities? |
| ☐ | ☒ | Does the activity involve any potential for fire or explosion where casks are loaded, unloaded, transported or stored? |

If ANY of the Part V questions are answered YES, then a full 10 CFR 72.48 screening is required and answers to the questions in Part VI and Part VII are to be provided. If ALL the questions in Part V are answered NO, then check Parts VI and VII as not applicable. Complete Part VIII to document the conclusion that no 10 CFR 72.48 evaluation is required

## PART VI (72.48) - DETERMINE IF THE CHANGE INVOLVES A ISFSI LICENSING BASIS *DESIGN FUNCTION*

(ALL the questions in Part V are NO, then Part VI is ☒ NOT APPLICABLE )

Compare the proposed activity to the relevant portions of the ISFSI licensing basis and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI Safety Analyses or plant/cask/ISFSI structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs whose function is relied upon for prevention of a radioactive release, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, CoC conditions, or orders? |
| ☐ | ☐ | Does the activity involve a *method of evaluation* described in the ISFSI licensing basis? |
| ☐ | ☐ | Is the activity a *test or experiment*? (i.e., a non-passive activity which gathers data) |
| ☐ | ☐ | Does the activity exceed or potentially affect a cask *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If THIS questions is answered YES, a 10 CFR 72.48 Evaluation is required.) |

If the answers to ALL of these questions are NO, mark Parts VII as not applicable, and document the 10 CFR 72.48 screening in the conclusion section (Part VIII).

If any of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved.

## PART VII (72.48) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (NEI 96-07, Appendix B, Section B.4.2.1)

(If ALL the questions in Part V or Part VI are answered NO, then Part VII is ☒ NOT APPLICABLE )

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 72.48 Evaluation is required; EXCEPT where noted in Part VII.3.

VII.1 Changes to the Facility or Procedures

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity adversely affect the *design function* of a plant, cask, or ISFSI SSC credited in safety analyses? |
| ☐ | ☐ | Does the activity adversely affect the method of performing or controlling the *design function* of a plant, cask, or ISFSI SSC credited in the safety analyses? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary)

VII.2 Changes to a Method of Evaluation

(If the activity does not involve a method of evaluation, these questions are ☐ NOT APPLICABLE.)

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in a cask SAR? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in a cask SAR? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary).

VII.3 Tests or Experiments

(If the activity is not a test or experiment, the questions in VII.3.a and VII.3.b are ☐ NOT APPLICABLE.)

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the cask ISFSI licensing basis? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the cask(s) or ISFSI facility? |

If the answer to both questions is NO, continue to VII.3.b. If the answer to EITHER question is YES, then briefly describe the basis.

b. Answer these additional questions ONLY for tests or experiments which do not meet the criteria given in VII.3.a above. If the answer to either question in VII.3 a is YES, then these three questions are ☐ NOT APPLICABLE:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity utilize or control a plant, cask or ISFSI facility SSC in a manner that is inconsistent with the analyses or descriptions in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity place the cask or ISFSI facility in a condition not previously evaluated or that could affect the capability of a plant, cask, or ISFSI SSC to perform its intended functions? |

If any answer in VII.3.b is YES, a 10 CFR 72.48 Evaluation is required If the answers are all NO, describe the basis for the conclusion (attach additional discussion as necessary).

## PART VIII - DOCUMENT THE CONCLUSION OF THE 10 CFR 72.48 SCREENING

Check all that apply:

A 10 CFR 72.48 Evaluation is ☐ required or ☒ NOT required. Obtain a screening number and provide the original to Records Management regardless of the conclusion of the 50.59 or 72.48 screening.

A VSC-24 cask Safety Analysis Report change is ☐ required or ☒ NOT required If a VSC-24 cask SAR change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A change to the VSC-24 10 CFR 72.212 Site Evaluation Report is ☐ required or ☒ NOT required. If a VSC-24 10 CFR 72 212 Site Evaluation Report change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

A. PURPOSE

1. This procedure provides directions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, assess plant conditions, and direct the operator to the appropriate recovery procedure.

2. This procedure is applicable for all plant conditions where RCS hot leg temperature is greater than or equal to 350°F with accumulators in service, and assumes the RHR system is not in service for decay heat removal and all SI system components are available.

B. SYMPTOMS OR ENTRY CONDITIONS

1. The following are symptoms that require a reactor trip, if one has not occurred:

| REACTOR TRIP SIGNAL | SETPOINT |
|---|---|
| ΔT Overtemperature | Variable |
| ΔT Overpower | Variable |
| RCP Breaker Trip Low Voltage | STPT 21.1 |
| RCP Breaker Trip Low Frequency | STPT 21.1 |
| RCS Loop Low Flow | 93 % |
| S/G Low-Low Level | 25% |
| S/G Low Level with Flow Mismatch | 30% of span |
| PZR Pressure Low | 1925 psig |
| PZR Pressure High | 2365 psig |
| PZR Level High | 80% |
| NIS Power High Range, High Level | 107% |
| NIS Power Low Range, High Level | 20% |
| NIS Intermediate Range | Current equal to 25% |
| NIS source range | $5 \times 10^5$ counts/sec |
| Manual Reactor Trip | N/A |
| Turbine Trip | N/A |
| Safety Injection | N/A |

2. The following are symptoms of a reactor trip:

   o Any reactor trip annunciator - LIT
   o Rapid drop in neutron level indicated by nuclear instrumentation
   o All rod bottom lights - ON
   o Reactor trip and bypass breakers - OPEN

3. The following are symptoms that require a reactor trip and safety injection, if one has not occurred:

| SAFETY INJECTION SIGNAL | SETPOINT |
|---|---|
| PZR Low Pressure | 1735 psig |
| Steam Line Low Pressure | 530 psig |
| Containment High Pressure | 5 psig |
| Manual Safety Injection | N/A |

4. The following are symptoms of a reactor trip and safety injection:

   a. Safeguards pumps and associated cooling water pumps- RUNNING

      • SI pumps
      • RHR pumps
      • Component cooling water pumps
      • Service water pumps

   b. SI-Spray Active Status Panel white lights - ON

   c. Containment Isolation Panels "A" and "B" white lights - ON

5. This procedure is entered from the following procedures if SI actuates:

   o EOP-0.2 UNIT 1, NATURAL CIRCULATION COOLDOWN, FOLDOUT
   o EOP-0.3 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), FOLDOUT
   o EOP-0.4 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS), FOLDOUT

6. This procedure is entered from the following procedure when PZR pressure is less than 1735 PSIG:

   • EOP-0.1 UNIT 1, REACTOR TRIP RESPONSE, Step 15

7. This procedure is entered from the following procedure if PZR level cannot be maintained:

   • CSP-I.2 UNIT 1, RESPONSE TO LOW PRESSURIZER LEVEL, Step 7

8. This procedure is entered from the following procedures if RCS subcooling or PZR level cannot be maintained:

   o EOP-0.1 UNIT 1, REACTOR TRIP RESPONSE, FOLDOUT
   o EOP-0.2 UNIT 1, NATURAL CIRCULATION COOLDOWN, FOLDOUT
   o EOP-0.3 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), FOLDOUT
   o EOP-0.4 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS), FOLDOUT

9. This procedure is entered from the following procedures when power is restored to a 480 Vac safequards bus prior to placing ECCS components in pull-out:

   o ECA-0.0 UNIT 1, LOSS OF ALL AC POWER, Step 16
   o ECA-0.0 UNIT 1, LOSS OF ALL AC POWER, Step 26

10. This procedure is entered from other plant procedures when a reactor trip or safety injection has occurred.

C. REFERENCES

1. Technical Specifications for Point Beach Nuclear Plant

2. Final Safety Analysis Report for Point Beach Nuclear Plant

3. As-built plant drawings

4. Generic Technical Guidelines developed by the Westinghouse Owners Group (WOG). This consists of the following documents:

   a. Low pressure version of the WOG Optimal Recovery Guidelines, Status Trees, and Functional Restoration Guidelines

   b. Background documents for each low pressure version Optimal Recovery Guideline, Status Tree, and Functional Restoration Guideline

   c. WOG Emergency Response Guideline Executive Volume

   d. WOG Emergency Response Guideline Maintenance Program Summary

5. Calculation 97-0126, Service Water System LOCA - Recirculation Phase

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

NOTE

Steps 1 through 4 are immediate action steps.

| | | |
|---|---|---|
| 1 | Verify Reactor Trip:<br><br>• Check reactor trip breakers and bypass breakers - OPEN<br><br>• 1-52/RTA<br>• 1-52/RTB<br>• 1-52/BYA<br>• 1-52/BYB<br><br>• Check all rod bottom lights - LIT<br><br>• Check all rod position indicators - ON BOTTOM<br><br>• Check neutron flux - LOWERING<br><br>• 1N-35<br>• 1N-36 | Perform the following:<br><br>a. Manually trip reactor.<br><br>b. IF reactor will NOT trip, THEN perform the following:<br><br>1) Deenergize rod drive motor generators by deenergizing 1B-01 and 1B-02.<br><br>• 1B52-04B or 1A52-02<br>• 1B52-05B or 1A52-15<br><br>2) WHEN the reactor has tripped, THEN close the following breakers:<br><br>• 1A52-02 for 1B-01<br>• 1B52-04B for 1B-01<br>• 1A52-15 for 1B-02<br>• 1B52-05B for 1B-02<br><br>3) IF reactor power is greater than or equal to 5% OR intermediate range power is rising, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to CSP-S.1 UNIT 1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.<br><br>4) As time permits, reenergize stripped MCCs.<br><br>5) Dispatch operator to locally open reactor trip breakers and bypass breakers in rod control room. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 2 | Verify Turbine Trip: | |
| | a. Check turbine stop valves - BOTH SHUT:<br>o SL and SR - SHUT<br><u>OR</u><br>o Annunciator 1C03 1E1 4-3, TURBINE STOP VALVES TWO CLOSED - LIT<br><u>OR</u><br>o Turbine Valves Closed bistable lights - LIT | a. Shutdown turbine as follows:<br>1) Depress turbine trip pushbutton.<br>2) <u>IF</u> turbine will <u>NOT</u> trip, <u>THEN</u> perform the following:<br>a) Manually run back turbine.<br>b) Stop both EH oil pumps and place in pull-out.<br>c) <u>IF</u> turbine still has <u>NOT</u> tripped, <u>THEN</u> shut main steam isolation valves.<br>• 1MS-2018 for S/G A<br>• 1MS-2017 for S/G B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 3 | Verify Safeguards Buses Energized: | |
| | a. Check 4160 Vac safeguards buses - AT LEAST ONE ENERGIZED<br><br>o 1A-05, train A<br>o 1A-06, train B | a. Try to restore power to at least one bus:<br><br>1) Close any supply breaker.<br><br>o 1A52-57 for 1A-05<br>o 1A52-54 for 1A-06<br>o 1A52-77 for 1A-06<br><br>2) IF breakers will NOT close, THEN fast start and load any emergency diesel generator.<br><br>3) IF power can NOT be restored, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions for information only per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to ECA-0.0 UNIT 1, LOSS OF ALL AC POWER. |
| | b. Check 480 Vac safeguards buses - AT LEAST ONE ENERGIZED<br><br>o 1B-03, train A<br>o 1B-04, train B | b. Try to restore power to at least one bus:<br><br>1) Close any supply breaker.<br><br>o 1A52-58 for 1B-03<br>o 1B52-16B for 1B-03<br>o 1A52-84 for 1B-04<br>o 1B52-17B for 1B-04<br><br>2) IF power can NOT be restored, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions for information only per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to ECA-0.0 UNIT 1, LOSS OF ALL AC POWER. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 4 | Check If SI Is Actuated: | |
| | a. Check SI annunciators - ANY LIT<br>o (1C04-1B 4-2), MANUAL SAFETY INJECTION<br>OR<br>o (1C04-1B 4-3), CONTAINMENT PRESSURE HIGH<br>OR<br>o (1C04-1B 4-4), PRESSURIZER LOW PRESSURE SI<br>OR<br>o (1C04-1B 4-5), STEAM LINE A PRESSURE LOW-LOW<br>OR<br>o (1C04-1B 4-6), STEAM LINE B PRESSURE LOW-LOW | a. Determine appropriate recovery actions:<br>1) Check if SI is required:<br>o Containment pressure - GREATER THAN 5 PSIG<br>OR<br>o Steam line A pressure - LESS THAN 530 PSIG<br>OR<br>o Steam line B pressure - LESS THAN 530 PSIG<br>OR<br>o PZR pressure - LESS THAN 1735 PSIG<br>OR<br>o PZR level - LESS THAN 10%<br>OR<br>o RCS subcooling - LESS THAN 35°F<br>2) IF SI is required, THEN perform the following:<br>a) Manually actuate both trains of SI and Containment Isolation.<br>b) OBSERVE NOTE PRIOR TO STEP 5 and go to Step 5.<br>3) IF SI is NOT required, THEN perform the following:<br>a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br>b) Go to EOP-0.1 UNIT 1, REACTOR TRIP RESPONSE. |
| | b. Check SI - BOTH TRAINS ACTUATED<br>• SI pumps - BOTH RUNNING<br>• RHR pumps - BOTH RUNNING | b. Manually actuate both trains of SI and Containment Isolation. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

NOTE

Foldout page shall be monitored throughout the remainder of this procedure.

5 Verify Automatic Actions Per ATTACHMENT A, AUTOMATIC ACTION VERIFICATION, While Continuing With This Procedure

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

CAUTION

If motor-driven auxiliary feedwater pump flow is greater than 240 gpm, its motor breaker may trip due to over current.

NOTE

If both units require AFW flow, at least one AFW pump must be aligned to each unit.

6 Verify Secondary Heat Sink Available:

a. Check level in at least one S/G - GREATER THAN [51%] 29%

a. Establish AFW flow as follows:

1) Manually start pumps and align valves as necessary to establish AFW flow greater than or equal to 200 gpm.

2) IF AFW flow greater than or equal to 200 gpm can NOT be established, THEN perform the following:

a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.

b) Go to CSP-H.1 UNIT 1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

b. Control pumps and align valves as necessary to maintain S/G level between [51%] 29% and 65%

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 7 | Verify RCP Seal Cooling:<br><br>o Check labyrinth seal ΔP - GREATER THAN 20 INCHES<br><br>OR<br><br>o Check component cooling to RCP thermal barrier - NORMAL | IF seal cooling to any RCP is lost, THEN reestablish seal cooling:<br><br>a. Stop affected RCP(s).<br><br>o 1P-1A, loop A<br>o 1P-1B, loop B<br><br>b. Start pumps and align valves as necessary to reestablish component cooling water flow to all RCP thermal barriers.<br><br>c. IF all charging pumps are stopped, THEN reestablish seal injection flow:<br><br>1) Ensure adequate power is available to run one charging pump. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for KW ratings.<br><br>2) Start one charging pump at minimum speed for seal injection.<br><br>o 1P-2A, train A<br>o 1P-2B, train A<br>o 1P-2C, train B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 8 | Verify RCS Temperature Control:<br><br>a. Check RCS wide range cold leg temperatures:<br><br>• LESS THAN OR EQUAL TO 547°F<br><br>AND<br><br>• STABLE | Perform the following:<br><br>1. IF RCS cold leg temperature less than 547°F AND RCS temperatures are trending lower, THEN stabilize RCS temperature as follows:<br><br>a) Stop dumping steam.<br><br>b) IF cooldown continues, THEN control feed flow:<br><br>1) Reduce total feed flow.<br><br>2) Maintain total feed flow greater than or equal to 200 gpm until level greater than [51%] 29% in at least one S/G.<br><br>c) IF cooldown can NOT be stopped by controlling feed flow, THEN isolate steam lines:<br><br>1) Shut both main steam isolation valves.<br><br>• 1MS-2018 for S/G A<br>• 1MS-2017 for S/G B<br><br>2) Ensure main steam isolation bypass valves - BOTH SHUT<br><br>• 1MS-234 for S/G A<br>• 1MS-236 for S/G B<br><br>2. IF RCS cold leg temperature greater than 547°F OR RCS temperature trending higher, THEN stabilize RCS temperature at or below 547°F as follows:<br><br>o Dump steam to condenser.<br><br>OR<br><br>o Dump steam using atmospheric steam dumps. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 9 | Check PZR PORVs - BOTH SHUT<br><br>• 1RC-430<br>• 1RC-431C | IF PZR pressure less than 2335 psig, THEN stop PORV flow:<br><br>a. Manually shut affected PORVs.<br><br>b. IF any PORV can NOT be shut, THEN isolate that PORV:<br><br>1) Manually shut associated block valve.<br><br>o 1RC-515 for 1RC-431C<br>o 1RC-516 for 1RC-430<br><br>2) IF any open PORV can NOT be isolated, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to EOP-1 UNIT 1, LOSS OF REACTOR OR SECONDARY COOLANT. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 10 | Verify PZR Spray Valves Shut: | |
| | a. Check normal spray valves - BOTH SHUT<br>• 1RC-431A, loop A<br>• 1RC-431B, loop B | a. IF PZR pressure less than 2260 psig, THEN stop spray flow:<br>1) Manually shut both spray valves.<br>2) IF any spray valve can NOT be shut, THEN place manual override switch to close for failed spray valve(s).<br>o 1RC-431A-S for 1RC-431A<br>o 1RC-431B-S for 1RC-431B<br>3) IF any spray valve can NOT be shut using manual override, THEN stop RCP supplying failed spray valve(s).<br>o For 1RC-431A, stop RCP A<br>o For 1RC-431B, stop RCP B |
| | b. Check auxiliary spray valve - SHUT<br>• 1CV-296 | b. Stop auxiliary spray flow:<br>1) Manually shut auxiliary spray valve.<br>2) IF auxiliary spray valve can NOT be shut, THEN minimize charging and shut charging line flow control valve.<br>• 1HC-142 |
| 11 | Check If RCPs Should Remain Running: | |
| | a. Check RCPs - ANY RUNNING | a. Go to Step 12. |
| | b. Check RCS subcooling based on core exit thermocouples - GREATER THAN OR EQUAL TO [60°F] 30°F | b. IF at least one SI pump is running AND SI pump capable of delivering flow, THEN stop both RCPs.<br>• 1P-1A, loop A<br>• 1P-1B, loop B |
| 12 | Start Monitoring Critical Safety Functions Per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 13 | Verify Containment Sump Recirculation Not Required: | |
| | a. Check RWST level - GREATER THAN OR EQUAL TO 60% | a. Go to EOP-1.3 UNIT 1, TRANSFER TO CONTAINMENT SUMP RECIRCULATION. |
| | b. Check RCS pressure - GREATER THAN [425 PSIG] 200 PSIG | b. IF RHR flow is greater than 450 gpm, THEN go to EOP-1.3 UNIT 1, TRANSFER TO CONTAINMENT SUMP RECIRCULATION. |
| 14 | Check If Secondary System Is Intact:<br>• No S/G pressure trending lower in an uncontrolled manner<br>AND<br>• No S/G completely depressurized | IF any S/G is faulted, THEN go to EOP-2 UNIT 1, FAULTED STEAM GENERATOR ISOLATION. |
| 15 | Check If S/G Tubes Are Intact:<br>• Check secondary system radiation levels - NORMAL<br>a. Condenser air ejector<br>• 1RE-215<br>• RE-225<br>b. S/G blowdown<br>• 1RE-219<br>• 1RE-222<br>c. Main steam line<br>• 1RE-231 for S/G A<br>• 1RE-232 for S/G B<br>• Request local surveys of main steam lines<br>• Request Chemistry to prepare for periodic activity samples of both S/Gs | IF conditions indicate a S/G tube rupture, THEN go to EOP-3 UNIT 1, STEAM GENERATOR TUBE RUPTURE. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 16 | Check If RCS Is Intact Inside Containment:<br><br>a. Check containment radiation levels - NORMAL<br><br>1) Containment<br><br>• 1RE-102, train A<br>• 1RE-107, train A<br><br>2) Containment high range<br><br>• 1RE-126, train A<br>• 1RE-127, train A<br>• 1RE-128, train A<br><br>b. On 1C20, check containment sump "A" level - NORMAL<br><br>• 1LI-958, train A<br>• 1LI-959, train A<br><br>c. Check containment pressure - NORMAL | Go to EOP-1 UNIT 1, LOSS OF REACTOR OR SECONDARY COOLANT. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 17 | Check If SI Should Be Terminated: | |
| | a. Check RCS subcooling based on core exit thermocouples - GREATER THAN 35°F | a. Go to Step 18. |
| | b. Verify secondary heat sink:<br>o Level in at least one S/G - GREATER THAN 29%<br>OR<br>o Total feed flow to S/Gs - GREATER THAN OR EQUAL TO 200 GPM | b. IF neither condition satisfied, THEN go to Step 18. |
| | c. Check RCS pressure:<br>• Pressure - GREATER THAN 1600 PSIG<br>AND<br>• Pressure - STABLE OR TRENDING HIGHER | c. Go to Step 18. |
| | d. Check PZR level - GREATER THAN 10% | d. Raise PZR level:<br>1) Raise charging flow.<br>2) Go to Step 18. |
| | e. Go to EOP-1.1 UNIT 1, SI TERMINATION | |
| 18 | Stabilize S/G Levels: | |
| | a. Check S/G levels - GREATER THAN 29% | a. Maintain total feed flow greater than or equal to 200 gpm until level in at least one S/G is greater than 29%. |
| | b. Control feed flow to maintain S/G levels between 29% and 65% | b. IF level in any S/G continues to rise in an uncontrolled manner, THEN go to EOP-3 UNIT 1, STEAM GENERATOR TUBE RUPTURE. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 19 | Check If S/G Tubes Are Intact:<br><br>• Check secondary system radiation levels - NORMAL<br><br>a. Condenser air ejector<br><br>• 1RE-215<br>• RE-225<br><br>b. S/G blowdown<br><br>• 1RE-219<br>• 1RE-222<br><br>c. Main steam line<br><br>• 1RE-231 for S/G A<br>• 1RE-232 for S/G B<br><br>• Request local surveys of main steam lines<br><br>• Request Chemistry to prepare for periodic activity samples of both S/Gs | IF conditions indicate a S/G tube rupture, THEN go to EOP-3 UNIT 1, STEAM GENERATOR TUBE RUPTURE. |

CAUTION

If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

20 Reset SI

21 Reset Containment Isolation

22 Reset 1B-03 And 1B-04 Non-Safeguards Equipment Lockouts

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

CAUTION

Placing loads on energized AC safeguards buses in excess of the power source's capacity could result in loss of the power source. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for KW ratings.

| | | |
|---|---|---|
| * 23 | Check 4160 Vac Safeguards Buses - BOTH ENERGIZED BY OFFSITE POWER<br><br>• 1A-05, train A<br>• 1A-06, train B | Monitor EDG loading per AOP-22 UNIT 1, EDG LOAD MANAGEMENT, while continuing with this procedure. |
| 24 | Reestablish Instrument Air To Containment: | |
| | a. Start second instrument air compressor<br><br>o K-2A<br>o K-2B | |
| | b. Check instrument air header pressure - GREATER THAN 80 PSIG | b. Start service air compressors as necessary to establish instrument air header pressure greater than 80 psig.<br><br>o K-3A<br>o K-3B |
| | c. Open one and then open the other instrument air containment isolation valve<br><br>• 1IA-3047<br>• 1IA-3048 | c. <u>IF</u> no valve can be opened, <u>THEN</u> gag open one valve as follows:<br><br>1) Manually hold valve switch in open position.<br><br>2) Locally gag open valve. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 25 | Check If RCS Is Intact Outside Containment:<br><br>a. Request local radiation surveys in auxiliary building<br><br>b. Check auxiliary building radiation levels - NORMAL<br><br>• PPCS RMS screen, Page 104<br>• {New PPCS RMS GRID screen}<br>• Local surveys<br><br>c. Check auxiliary building sump levels - NORMAL<br><br>• C01A 1-11, AUXILIARY BUILDING -19 FT SUMP LEVEL HI - NOT LIT<br>• C01A 2-11, AUXILIARY BUILDING NORTH SUMP LEVEL HI - NOT LIT<br>• C01A 3-11, AUXILIARY BUILDING SOUTH SUMP LEVEL HI - NOT LIT | Perform the following:<br><br>1. Evaluate cause of abnormal conditions.<br><br>2. <u>IF</u> cause is a loss of RCS inventory outside containment, <u>THEN</u> go to <u>ECA-1.2 UNIT 1, LOCA OUTSIDE CONTAINMENT</u>. |
| 26 | Check PZR Relief Tank Conditions - NORMAL<br><br>• Pressure<br><br>• Temperature<br><br>• Level | Evaluate cause of abnormal conditions. |
| 27 | Check If RHR Pumps Should Be Stopped:<br><br>a. Check RCS pressure - GREATER THAN 200 PSIG<br><br>b. Check RCS pressure - STABLE OR TRENDING HIGHER<br><br>c. Stop both RHR pumps<br><br>• 1P-10A, train A<br>• 1P-10B, train B | a. Go to <u>EOP-1 UNIT 1, LOSS OF REACTOR OR SECONDARY COOLANT</u>.<br><br>b. Go to <u>Step 28</u>. |

**********
* d. Maintain RCS pressure greater than 200 psig

d. <u>IF</u> RCS pressure lowers in an uncontrolled manner to less than 200 psig, <u>THEN</u> restart RHR pumps to supply water to RCS. *
**********

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 28 | Verify Charging Flow: | |
| | a. Ensure RCS Loop A Cold Leg Normal Charging Isolation Valve - OPEN<br><br>• 1CV-1298 | |
| | b. Check charging pumps - AT LEAST ONE RUNNING<br><br>o 1P-2A, train A<br>o 1P-2B, train A<br>o 1P-2C, train B | b. Perform the following:<br><br>1) <u>IF</u> component cooling water flow to any RCP thermal barrier is lost, <u>THEN</u> locally shut affected RCP(s) seal injection throttle valve before starting charging pumps.<br><br>o 1CV-300A, RCP A<br>o 1CV-300B, RCP B<br><br>2) Start charging pumps as necessary to establish at least one running. |
| | c. Start additional charging pumps and adjust speed on running charging pumps as necessary to establish desired charging flow | |
| | d. Adjust charging line flow controller as necessary to maintain labyrinth seal ΔP greater than 20 inches<br><br>• 1HC-142 | |
| 29 | Check If Diesels Should Be Stopped: | |
| | a. Check 4160 Vac safeguards buses - ENERGIZED BY OFFSITE POWER<br><br>• 1A-05, train A<br>• 1A-06, train B | a. Restore offsite power to 4160 Vac safeguards buses. |
| | b. Stop all unloaded EDGs:<br><br>o OP-11A G-01, EMERGENCY DIESEL GENERATOR G-01<br>o OP-11A G-02, EMERGENCY DIESEL GENERATOR G-02<br>o OP-11B, EMERGENCY DIESEL GENERATOR G-03 (G-04) | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 30 | Ensure Miscellaneous Electrical Loads Are Energized: | |
| | a. Ensure MCCs - ENERGIZED<br>• 1B-31, 1B52-14C, train A<br>• B-43, 1B52-21C, train B | |
| | b. Check battery chargers supplying DC buses - ENERGIZED<br>o D-07<br>o D-09<br>o D-108<br>o D-109 | b. Restore battery chargers:<br>1) Close affected battery charger supply contactor.<br>2) <u>IF</u> contactor does <u>NOT</u> close <u>OR</u> battery charger will <u>NOT</u> operate, <u>THEN</u> restore battery chargers per AOP-0.0, VITAL DC SYSTEM MALFUNCTION, while continuing with this procedure. |
| | c. Ensure cavity cooling fan - ONE RUNNING<br>o 1W-4A, train A<br>o 1W-4B, train A | |
| | d. Check cable spreading room ventilation operating:<br>1) Check cable spreading room recirc fans - ONE RUNNING<br>o W-13A1<br>o W-13A2<br>2) Check CSR chilled water recirc pumps - ONE RUNNING<br>o P-111A<br>o P-111B | d. Restore cable spreading room ventilation per OI-90, CONTROL, COMPUTER, AND CABLE SPREADING ROOM VENTILATION SYSTEMS. |
| | e. Start additional loads as necessary to meet current plant conditions. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT | |
| 31 | Return To <u>Step 8</u> | |

-END-

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 1 of 10)
AUTOMATIC ACTION VERIFICATION

A1 Verify Feedwater Isolation:

| | |
|---|---|
| a. Check main feed lines isolated:<br><br>1) Feedwater regulating control valves - BOTH SHUT<br><br>• 1CS-466 for S/G A<br>• 1CS-476 for S/G B<br><br>2) Feedwater regulating bypass valves - BOTH SHUT<br><br>• 1CS-480 for S/G A<br>• 1CS-481 for S/G B | a. <u>IF</u> any main feedline can <u>NOT</u> be isolated, <u>THEN</u> perform the following:<br><br>a) Trip main feed pumps.<br><br>• 1P-28A<br>• 1P-28B<br><br>b) Place condensate pumps in pull-out.<br><br>• 1P-25A<br>• 1P-25B<br><br>c) Stop heater drain tank pumps.<br><br>• 1P-27A<br>• 1P-27B<br>• 1P-27C |
| b. Check main feed pumps - BOTH TRIPPED<br><br>• 1P-28A<br>• 1P-28B | b. Trip main feed pumps. |
| c. Check MFP discharge MOVs - BOTH SHUT<br><br>• 1CS-2190, train A<br>• 1CS-2189, train B | c. Manually shut valves. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 2 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A2 | Verify Containment Isolation: | |
| | a. Check containment isolation panels "A" and "B" - ALL LIGHTS LIT | a. Perform the following:<br>1) Manually actuate Containment Isolation.<br>2) <u>IF</u> any valve open <u>AND</u> flow path <u>NOT</u> required, <u>THEN</u> shut valve(s). Refer to ATTACHMENT B. |
| | b. Check other valves - SHUT<br>• RS-SA-9 , Unit 1 steam supply to rad waste system<br>• Any valve which may be open under administrative control | b. Manually shut valve(s). |
| A3 | Verify AFW Actuation: | |
| | a. Check motor-driven AFW pumps - BOTH RUNNING<br>• P-38A, train A<br>• P-38B, train B | a. Establish AFW flow as follows:<br>1) Ensure steam supply valves to turbine-driven AFW pump - BOTH OPEN<br>• 1MS-2020, train A<br>• 1MS-2019, train B<br>2) <u>WHEN</u> SI sequence complete, <u>THEN</u> manually start motor-driven AFW pumps. |
| | b. Check S/G levels - BOTH LESS THAN [51%] 25% | b. Go to <u>Step A4</u>. |
| | c. Ensure steam supply valves to turbine-driven AFW pump - BOTH OPEN<br>• 1MS-2020, train A<br>• 1MS-2019, train B | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 3 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A4 | Check SI Pumps - BOTH RUNNING<br><br>• 1P-15A, train A<br>• 1P-15B, train B | WHEN SI sequence complete, THEN establish SI flow as follows:<br><br>a. Manually start SI pumps.<br><br>b. IF any SI pump can NOT be started, THEN isolate system boundary as follows:<br><br>1) Place affected SI pump in pull-out.<br><br>2) Ensure affected SI pump suction valve shut.<br><br>o 1SI-896A, train A<br>o 1SI-896B, train B |
| A5 | Check RHR Pumps - BOTH RUNNING<br><br>• 1P-10A, train A<br>• 1P-10B, train B | WHEN SI sequence complete, THEN establish RHR flow as follows:<br><br>a. Manually start RHR pumps.<br><br>b. IF any RHR pump can NOT be started, THEN isolate system boundary as follows:<br><br>1) Place affected RHR pump in pull-out.<br><br>2) Ensure affected RHR pump suction valve shut.<br><br>o 1SI-856A, train A<br>o 1SI-856B, train B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 4 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A6 | Check Component Cooling Water Pumps - ONLY ONE RUNNING<br><br>o 1P-11A, train A<br>o 1P-11B, train B | Establish one component cooling water pump running as follows:<br><br>a. <u>IF</u> no component cooling water pump running, <u>THEN</u> perform the following:<br><br>1) Stop all RCP(s).<br><br>• 1P-1A, loop A<br>• 1P-1B, loop B<br><br>2) Manually start one component cooling water pump by placing control switch to stop and then auto-after-stop.<br><br>3) Match flags for running and stopped pumps.<br><br>b. <u>IF</u> both component cooling water pumps running, <u>THEN</u> place one pump in standby. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 5 of 10)
AUTOMATIC ACTION VERIFICATION

A7 Verify Service Water System Alignment:

a. Check service water pumps - SIX RUNNING

- P-32A, train A
- P-32B, train A
- P-32F, train A
- P-32C, train B
- P-32D, train B
- P-32E, train B

a. WHEN SI sequence complete, THEN manually start pumps.

b. Check service water isolation valves - SHUT

- At least one spent fuel pool heat exchanger A isolation MOV
  - o SW-2927A, inlet MOV
  - o SW-2930A, discharge MOV
- At least one spent fuel pool heat exchanger B isolation MOV
  - o SW-2927B, inlet MOV
  - o SW-2930B, discharge MOV
- At least one auxiliary building A/C condenser isolation MOV
  - o SW-2816, train A
  - o SW-4479, train B
- At least one water treatment system inlet MOV
  - o SW-4478, train A
  - o SW-2817, train B

b. Perform the following:

1) Manually shut valve(s).

2) IF any isolation valve will NOT shut, THEN locally shut valve or associated manual isolation valve.

c. Locally at blowdown evap panel C-180, check at least one radwaste service water valve shut

- o SW-LW-61, train A
- o SW-LW-62, train B

c. Perform the following:

1) Locally shut valve(s).

2) IF any valve will NOT shut, THEN locally shut associated manual isolation valve.

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 6 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A8 | Verify Containment Accident Cooling Units Running: | |
| | a. Check containment accident recirculation fans - ALL RUNNING<br>• 1W-1A1, train A<br>• 1W-1B1, train A<br>• 1W-1C1, train B<br>• 1W-1D1, train B | a. <u>WHEN</u> SI sequence complete, <u>THEN</u> manually start fans. |
| | b. Check containment ventilation cooler outlet emergency FCVs - BOTH OPEN<br>• 1SW-2907, train A<br>• 1SW-2908, train B | b. Manually open containment ventilation cooler outlet emergency FCVs. |
| | c. Check annunciator C01B 2-3, UNIT 1 CONTAINMENT RECIRC COOLERS WATER FLOW LOW - CLEAR | c. Perform the following:<br>1) Ensure non-affected unit's service water isolation valves - BOTH SHUT<br>• 2SW-2907 , train A<br>• 2SW-2908 , train B<br>2) Isolate service water to non-safety loads as necessary to clear annunciator. |
| A9 | Check Control Room Fans Armed: | |
| | a. Check Control Room Charcoal Filter Fan W-14A - WHITE LIGHT OFF | a. At MCC 1B-32, depress Control Circuit Arming pushbutton for Control Room charcoal filter fan W-14A.<br>• 1B52-329B |
| | b. Check Control Room Recirc Fan W-13B2 - WHITE LIGHT OFF | b. At MCC 1B-42, depress Control Circuit Arming pushbutton for Control Room recirc fan W-13B2.<br>• 1B52-428M |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 7 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A10 | Check Control Room Ventilation - IN AN ACCIDENT MODE<br><br>• Control Room recirc fans - AT LEAST ONE RUNNING<br><br>o W-13B1<br>o W-13B2<br><br>• Control Room damper solenoid valve - PURPLE LIGHT LIT | Align Control Room ventilation per OI-90, CONTROL, COMPUTER, AND CABLE SPREADING ROOM VENTILATION SYSTEMS. |
| A11 | Check If Main Steam Lines Can Remain Open: | |
| | a. Check MSIVs - ANY OPEN | a. Go to Step A12. |
| | b. Check containment pressure - LESS THAN OR EQUAL TO 15 PSIG | b. Isolate both steam lines as follows:<br><br>1) Shut both main steam isolation valves.<br><br>• 1MS-2018 for S/G A<br>• 1MS-2017 for S/G B<br><br>2) Go to Step A12. |
| | c. Check high-high steam flow bistable lights - NOT LIT | c. Ensure main steam isolation valve on affected main steam line(s) shut.<br><br>o 1MS-2018 for S/G A<br>o 1MS-2017 for S/G B |
| | d. Check high steam flow bistable lights - NOT LIT | d. IF RCS average temperature is less than 543°F, THEN ensure main steam isolation valve on affected main steam line(s) shut.<br><br>o 1MS-2018 for S/G A<br>o 1MS-2017 for S/G B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 8 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A12 | Verify Proper SI Valve Alignment:<br><br>a. Check Unit 1 SI Active status panel - ALL LIGHTS LIT<br><br>b. Check Unit 1 SI - Spray Ready status panel - NO LIGHTS LIT | Manually align pumps and valves as necessary to establish proper SI alignment. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 9 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A13 | Verify Containment Spray Not Required:<br><br>a. Check containment pressure recorder - HAS REMAINED LESS THAN 25 PSIG<br><br>• 1PR-968<br>• 1PR-969 | Establish containment spray as follows:<br><br>1. Check containment spray actuated:<br><br>• Annunciator {C01 B 2-6}, CONTAINMENT SPRAY - LIT<br><br>2. <u>IF</u> containment spray has <u>NOT</u> actuated, <u>THEN</u> manually actuate containment spray.<br><br>3. Verify the following equipment status:<br><br>a) Ensure containment spray pump discharge MOVs - ALL OPEN<br><br>• 1SI-860A for 1P-14A<br>• 1SI-860B for 1P-14A<br>• 1SI-860C for 1P-14B<br>• 1SI-860D for 1P-14B<br><br>b) Ensure containment spray pumps - AT LEAST ONE RUNNING<br><br>o 1P-14A, train A<br>o 1P-14B, train B<br><br>c) Shutdown one train of containment spray as follows:<br><br>1) Place one containment spray pump in pull-out.<br><br>o 1P-14A, train A<br>o 1P-14B, train B<br><br>2) Ensure suction on idle spray pump shut.<br><br>o 1SI-870A for 1P-14A<br>o 1SI-870B for 1P-14B<br><br>d) <u>WHEN</u> containment spray has been actuated for greater than two minutes, <u>THEN</u> ensure at least one spray additive eductor suction valve open.<br><br>o 1SI-836A, train A<br>o 1SI-836B, train B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 10 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A14 | Verify SI Flow: | |
| | a. Check RCS wide range pressure - LESS THAN 1400 PSIG | a. Return to procedure and step in effect. |
| | b. Check SI pumps - FLOW INDICATED<br>• 1FI-925, train A<br>• 1FI-924, train B | b. Manually start pumps and align valves as necessary to establish SI pump flow. |
| | c. Check RCS wide range pressure - LESS THAN [425 PSIG] 200 PSIG | c. Return to procedure and step in effect. |
| | d. Check RHR pumps - FLOW INDICATED<br>• 1FI-626, train A<br>• 1FI-928, train B | d. Manually start pumps and align valves as necessary to establish RHR pump flow. |

-END-

ATTACHMENT B
(Page 1 of 2)
CONTAINMENT ISOLATION VALVES

| PANEL A | | |
|---|---|---|
| COMPONENT | DESCRIPTION | TRAIN |
| 1CV-1296 | Auxiliary charging line | A |
| 1RC-538 | Pressurizer relief tank to gas analyzer | A |
| 1WG-1788 | Reactor coolant drain tank to gas analyzer | A |
| 1WL-1698 | Reactor coolant drain tank to -19 ft sump | A |
| 1WL-1003A | Reactor coolant drain tank pump suction | A |
| 1WL-1003B | Reactor coolant drain tank pump suction | A |
| 1RC-508 | Reactor makeup water to containment | A or B |
| 1RC-539 | Pressurizer relief tank to gas analyzer | B |
| 1WG-1789 | Reactor coolant drain tank to gas analyzer | B |
| 1SI-846 | Accumulator nitrogen supply | A or B |
| 1WL-1721 | Reactor coolant drain tank pumps suction | B |
| 1VNPSE-3244 | Containment purge supply | A |
| 1VNPSE-3212 | Containment purge exhaust | A |
| 1WL-1723 | Sump A drain | A |
| 1SC-951 | Pressurizer steam sample | A |
| 1SC-953 | Pressurizer liquid sample | A |
| 1VNPSE-3245 | Containment purge supply | B |
| 1VNPSE-3213 | Containment purge exhaust | B |
| 1WL-1728 | Sump A drain | B |
| 1SC-966A | Pressurizer steam sample | A or B |
| 1SC-966B | Pressurizer liquid sample | A or B |

ATTACHMENT B
(Page 2 of 2)
CONTAINMENT ISOLATION VALVES

| PANEL B | | |
|---|---|---|
| COMPONENT | DESCRIPTION | TRAIN |
| 1CC-769 | Component cooling water outlet from excess letdown heat exchanger | A or B |
| 1CV-313 | Reactor coolant pump seal return | A |
| 1CV-371 | Letdown line | A |
| 1MS-5958 | Steam generator blowdown | A or B |
| 1MS-5959 | Steam generator blowdown | A or B |
| 1WG-1786 | Reactor coolant drain tank vent | A |
| 1CV-313A | Reactor coolant pump seal return | B |
| 1CV-371A | Letdown line | B |
| 1WG-1787 | Reactor coolant drain tank vent | B |
| 1RM-3200C | RE-211/212 supply | A |
| 1RM-3200A | RE-211/212 return | A or B |
| 1MS-2083 | Steam generator A sample | A or B |
| 1MS-2084 | Steam generator B sample | A or B |
| 1SC-955 | Reactor coolant hot leg sample | A |
| 1IA-3047 | Instrument air line | A or B |
| 1RM-3200B | RE-211/212 supply | B |
| 1SC-966C | Reactor coolant hot leg sample | A or B |
| 1IA-3048 | Instrument air line | A or B |

-END-

FOLDOUT PAGE FOR EOP-0 UNIT 1

1. RCP TRIP CRITERIA

IF all conditions listed below occur, THEN trip both RCPs:

- RCS subcooling - LESS THAN [60°F] 30°F
- SI pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW
- Operator controlled cooldown - NOT IN PROGRESS

2. FAULTED S/G ISOLATION CRITERIA

IF any S/G pressure trending lower in an uncontrolled manner OR any S/G completely depressurized, THEN the following may be performed:

a. Isolate feed flow to faulted S/G.

b. Maintain total feed flow greater than or equal to 200 gpm until narrow range level in at least one S/G is greater than [51%] 29%.

3. RUPTURED S/G ISOLATION CRITERIA

IF any S/G level rises in an uncontrolled manner OR any S/G has abnormal radiation, AND narrow range level in affected S/G(s) is greater than [51%] 29%, THEN feed flow may be isolated to affected S/G(s).

4. AFW SUPPLY SWITCHOVER CRITERIA

IF CST level lowers to less than 8 feet, THEN switch to alternate AFW suction supply per AOP-23 UNIT 1, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY.

5. ADVERSE CONTAINMENT CONDITIONS

IF any condition listed below occurs, THEN adverse containment setpoint values in brackets, [], shall be used:

o Containment pressure - GREATER THAN 10 PSIG

OR

o Containment radiation level - GREATER THAN OR EQUAL TO $10^5$ R/HR

OR

o Integrated dose to containment - GREATER THAN $10^6$ R

6. AFW MINIMUM FLOW REQUIREMENTS

IF any AFW pump mini-recirc valve fails shut, THEN maintain minimum flow or stop the affected AFW pump as necessary to control S/G levels.

o P-38A minimum flow - GREATER THAN 50 GPM

o P-38B minimum flow - GREATER THAN 50 GPM

o P-29 minimum flow - GREATER THAN 75 GPM

Nuclear Power Business Unit

# TEMPORARY CHANGE REVIEW AND APPROVAL

*Note: Refer to NP 1.2.3, Temporary Procedure Changes, for requirements.*



## I - INITIATION

Doc Number EOP-0 Current Rev 36 Unit PB1 Temp Change No. TCN2001-0915

Document Title REACTOR TRIP OR SAFETY INJECTION

Existing Effective Temporary Changes

Brief Description MODIFY FOP FOR MINIMUM AFW FLOW TO INCLUDE LOW IA HDR PRESSURE

(Identify specific changes on Form PBF-0026c, Document Review and Approval Continuation, and include with the package)

☒ Initiate PBF-0026h and include with the change.

Other documents required to be effective concurrently with the temporary change:

Changes pre-screened according to NP 10.3.1? ☒ NO ☐ YES (If Yes, list references and criteria on PBF-0026c)(refer to NP 10.3.1)

Screening completed according to NP 10.3.1? ☐ NA ☒ YES

Safety Evaluation Required? ☒ NO ☐ YES (If Yes a revision may be processed or final reviews and approvals shall be obtained before implementing)

Determine if the change constitutes a Change Of Intent to the procedure by evaluating the following questions
(If any answers are YES, a revision may be processed or final reviews and approvals shall be obtained before implementing)

| Will the proposed change: | YES | NO |
|---|---|---|
| 1. Require a change to, affect or invalidate a requirement, commitment, evaluation or description in the Current Licensing Basis (as defined in NP 10.3.1)? | ☐ | ☒ |
| 2. Cause an increase in magnitude, significance or impact such that it should be processed as a revision? | ☐ | ☒ |
| 3. Delete or modify a prerequisite, initial condition, precaution, limitation or other steps that could have safety significance or affect the procedure's margin of safety? | ☐ | ☒ |
| 4. Delete QC hold points, Independent Verification or Concurrent Check steps without the related step(s) that require the performance also being deleted? | ☐ | ☒ |
| 5. Change Tech Spec or other regulatory acceptance criteria other than for re-baselining purposes? | ☐ | ☒ |
| 6. Require a change to the procedure Purpose or change the procedure classification? | ☐ | ☒ |

Initiated By (print/sign) C. R. Carroll / [signature] Date 12/20/01

## II - INITIAL APPROVAL

This change is correct and complete, can be performed as written, and does not adversely affect personnel or nuclear safety, or Plant operating conditions

Group Supervisor (print/sign) R. SANDERBOSTON / [signature] Date 12/20/01

(Cannot be the Initiator)

This change does not adversely affect Plant operating conditions. (Safety Related procedures only)

Senior Reactor Operator (print/sign) M Schug / M Schug Date 12/20/01

(Cannot be the Initiator or Group Supervisor)

## III - PROCEDURE OWNER REVIEW

☒ Permanent ☐ One-time Use ☐ Expiration Date, Event or Condition:

☐ Hold change until procedure completed (final review and approval still required within 14 days of initial approval)

☒ QR/MSS Review NOT Required (Admin NNSR only) ☒ QR Review Required ☐ MSS Review Required (Reference NP 1.6.5)

12/20/01 Procedure Owner (print/sign) BOB WANTENBERG / [signature] Date 12/20/01

This Change and supporting requirements correctly completed and processed

## IV - FINAL REVIEW AND APPROVAL

(Must be completed within 14 days of initial approval) (The Initiator, QR and Approval Authority shall be independent from each other)

QR/MSS (print sign) BOB WANTENBERG / [signature] Date 12/20/01

Indicates 50.59/72.48 applicability assessed, any necessary screening/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed.

MSS Meeting No.

Approval Authority (print/sign) D. SCHOON / [signature] Date 12-20-01

## V - REVISION INFORMATION FOR PERMANENT CHANGES

Post Typing Review (print/sign) C. R. Carroll / [signature] Date 1/4/01

Indicates temporary change(s) incorporated exactly as approved and no other changes made to document

Incorporated into Revision Number 37 Effective Date JAN 1 2001

REC'D JAN 11 2002

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL CONTINUATION



Doc Number EOP-0 Revision 36 Unit 1

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number TCN 2001-0915

Description of Changes:

| Step * | Change/Reason |
|---|---|
| FOLDOUT PAGE | CHANGE: ADDED REFERENCE TO ANNUNCIATOR C01 A 1-9, IA HEADER LOW PRESSURE FOR THE AFW MINIMUM FLOW REQUIREMENTS.<br>REASON: TO ENSURE MINIMUM FLOW IS MAINTAINED THROUGH THE AFW PUMPS DURING OPERATION. |

Other Comments

* Note: Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to readers.

Point Beach Nuclear Plant

# TEMPORARY CHANGE AFFECTED MANUAL LOCATION



Procedure Number EOP-0 Revision 36 Unit PB1

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number TCN 2001-0915

## I - IMMEDIATELY AFTER INITIAL APPROVAL ON PBF-0026e (Non-Intent changes)

(after Final Approval if change of intent involved)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☐ Copy included in work package for field implementation. (WO No. _____ ) | |
| ☒ Copy filed in Control Room temp change binder (Operations only). | 12-20-01 |
| ☒ Original change package provided to RJW to obtain Procedure Owner Review (e.g., Owner review may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 12-20-01 |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |

Performed By (print and sign) Carol Schroeder / Carol Schroeder Date 12-20-01

## II - PROCEDURE OWNER REVIEW ON PBF-0026e

(may be performed by OA II, Procedure Writer, etc.)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☒ Copy sent to Document Control Distribution Lead for Master File (Not required for one-time use change) | 12-20-01 |
| ☐ Copy filed in Group satellite file. (Not required for one-time use changes.) | |
| ☐ Copy filed in Group one-time use file. | |
| ☒ Original Temp Change provided to DDS to obtain Final Approvals (e.g., final approval may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 12-20-01 |
| ☒ U1/U2 | 12-20-01 |
| ☒ PAB | ✓ |
| ☒ Ops Shop | ✓ |
| ☒ Ops Office | ✓ |
| ☒ Simulator (Thru OAII) | ✓ |
| ☐ | |

Performed By (print and sign) Carol Schroeder / Carol Schroeder Date 12-20-01

LOP ARP Actions For AFW Min-Recirc Flow Requirements

| | | | | |
|---|---|---|---|---|
| Preparer | Bob Warnenberg (Name (Print)) | (Signature) | Date | 1/22/01 |
| Reviewer | E. Clayton Graves (Name (Print)) | (Signature) | Date | 1/22/01 |

PART I (50.59/72.48) - DESCRIBE THE PROPOSED ACTIVITY AND SEARCH THE PLANT AND ISFSI LICENSING BASIS (Resource Manual 5.3.1)

NOTE: The "NMC 10 CFR 50.59 Resource Manual" (Resource Manual) and NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 10 CFR 50.59 and 10 CFR 72.48 screenings.

1.1 Describe the proposed activity and the scope of the activity being covered by this screening. (The 10 CFR 50.59 / 72.48 review of other portions of the proposed activity may be documented via the applicability and pre-screening process requirements in NP 5.1.8.) Appropriate descriptive material may be attached.

EOP-0 and EOP-0.1 for both units revised the AFW Minimum Flow Requirements foldout-page criteria to include the instrument air header pressure low annunciator in the alarm state as additional criteria for monitoring AFW min-recirc flow requirements.

AFW Minimum Flow Requirements foldout-page item was added to ECA-0.0. The foldout-page item is identical to the AFW minimum flow requirements used in EOP-0 and EOP-0.1.

Step 3.1 was added to ARP COI A 1-9. The step states to monitor and maintain AFW min-recirc flow requirements should the AFW pump min-recirc valve fail shut.

1.2 Search the PBNP Current Licensing Basis (CLB) as follows: Final Safety Analysis Report (FSAR), FSAR Change Requests (FCRs) with assigned numbers, the Fire Protection Evaluation Report (FPER), the CLB (Regulatory) Commitment Database, the Technical Specifications (both Custom and Improved), the Technical Specifications Bases, and the Technical Requirements Manual. Search the ISFSI licensing basis as follows: VSC-24 Safety Analysis Report, the VSC-24 Certificate of Compliance, the CLB (Regulatory) Commitment Database, and the VSC-24 10 CFR 72.212 Site Evaluation Report. Describe the permanent design functions, performance requirements, and methods of [illegible] for the [illegible] (ISFSI) as appropriate. Identify where the permanent information is described [illegible] document section number and title). (Resource Manual 5.3.1 and NEI 96-07, App. B, B.2.)

FSAR [illegible] Feedwater System

[illegible] activity involves [illegible] Custom or Improved Technical Specifications requires [illegible] Request. Resource Manual [illegible]

[illegible] Application Change [illegible]

[illegible]

Does the proposed activity involve a change to the terms, conditions or specifications incorporated in any VSC-24 cask Certificate of Compliance (CoC)? Changes to a VSC-24 cask Certificate of Compliance require a CoC amendment request.

☐ Yes ☒ No

If a storage cask Certificate of Compliance change is required, explain what the change should be and why it is required.

---------- 10 CFR 50.59 SCREENING ----------

PART II (50.59) - DETERMINE IF THE CHANGE INVOLVES A *DESIGN FUNCTION* (Resource Manual 5.3.2)

Compare the proposed activity to the relevant CLB descriptions, and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☒ | ☐ | Does the proposed activity involve Safety Analyses or structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs whose failure could initiate a transient (e.g., reactor trip, loss of feedwater, etc.) or accident, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve CLB-described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications? |
| ☐ | ☒ | Does the activity involve a *method of evaluation* described in the FSAR? |
| ☐ | ☒ | Is the activity a *test or experiment*? (i.e., a non-passive activity which gathers data) |
| ☐ | ☒ | Does the activity exceed or potentially affect a *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If THIS questions is answered YES, a 10 CFR 50.59 Evaluation is required.) |

If the answers to ALL of these questions are NO, mark Part III as not applicable, document the 10 CFR 50.59 screening in the conclusion section (Part IV), then proceed directly to Part V - 10 CFR 72.48 Pre-screening Questions.

If any of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved.

FSAR 10.2 states each AFW pump has an AOV controlled recirc line back to the CST to ensure minimum flow to dissipate heat. This change ensures the minimum AFW flow requirements will be maintained on any running AFW pump in the case of a failed shut AFW mini-recirc flow control valve

PART III (50.59) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (Resource Manual 5.3.3)

If ALL the questions in Part II are answered NO, then Part III is ☐ NOT APPLICABLE.

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 50.59 Evaluation is required, EXCEPT where noted in Part III.3.

III.1 CHANGES TO THE FACILITY OR PROCEDURES

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☒ | Does the activity adversely affect the *design function* of an SSC credited in safety analyses? |

☐ ☒ Does the activity adversely affect the method of performing or controlling the *design function* of an SSC credited in the safety analyses?

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion as necessary):

This change ensures that minimum recirc flow requirements as stated in FSAR 16.2 are not violated.

III.2 CHANGES TO A METHOD OF EVALUATION

(If the activity does not involve a method of evaluation, these questions are ☒ NOT APPLICABLE.)

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in the CLB? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in the CLB? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary).

III.3 TESTS OR EXPERIMENTS

If the activity is not a test or experiment, the questions in III.3.a and III.3.b are ☒ NOT APPLICABLE.

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the CLB? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the facility? |

If the answer to BOTH questions in V.3.a is NO, continue to III.3.b. If the answer to EITHER question is YES, then describe the basis.

b. Answer these additional questions ONLY for tests or experiments which do NOT meet the criteria given in III.3.a above. If the answer to either question in III.3.a is YES, then these three questions are ☐ NOT APPLICABLE.

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the CLB? |
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is inconsistent with the analyses or descriptions in the CLB? |
| ☐ | ☐ | Does the activity place the facility in a condition not previously evaluated or that could affect the capability of an SSC to perform its intended functions? |

If any answer in III.3.b is YES, a 10 CFR 50.59 Evaluation is required. If the answers in III.3.b are ALL NO, describe the basis for the conclusion (attach additional discussion as necessary):

IV - 10 CFR 50.59 SCREENING CONCLUSION (Resource Manual 5.3.4).

Check all that apply:

A 10 CFR 50.59 Evaluation is ☐ required or ☒ NOT required.

A Point Beach FSAR change is ☐ required or ☒ NOT required. If an FSAR change is required, then initiate an FSAR Change Request (FCR) per NP 5.2.6.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A Technical Specification Bases change is ☐ required or ☒ NOT required. If a change to the Technical Specification Bases is required, then initiate a Technical Specification Bases change per NP 5.2.15.

A Technical Requirements Manual change is ☐ required or ☒ NOT required If a change to the Technical Requirements Manual is required, then initiate a Technical Requirements Manual change per NP 5.2.15.

---------- 10 CFR 72.48 SCREENING ----------

**NOTE: NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 72.48 screenings.**

**PART V (72.48) - 10 CFR 72.48 INITIAL SCREENING QUESTIONS**

Part V determines if a full 10 CFR 72.48 screening is required to be completed (Parts VI and VII) for the proposed activity.

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER the dry fuel storage cask(s), the cask transfer/transport equipment, any ISFSI facility SSC(s), or any ISFSI facility monitoring as follows: Multi-Assembly Sealed Basket (MSB), MSB Transfer Cask (MTC), MTC Lifting Yoke, Ventilated Concrete Cask (VCC), Ventilated Storage Cask (VSC), VSC Transporter (VCST), ISFSI Storage Pad Facility, ISFSI Storage Pad Data/Communication Links, or PPCS/ISFSI Continuous Temperature Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) installed in the plant specifically added to support cask loading/unloading activities, as follows: Cask Dewatering System (CDW), Cask Reflood System (CRF), or Hydrogen Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) needed for plant operation which are also used to support cask loading/unloading activities, as follows: Spent Fuel Pool (SFP), SFP Cooling and Filtration (SF), Primary Auxiliary Building Ventilation System (VNPAB), Drumming Area Ventilation System (VNDRM), RE-105 (SFP Low Range Monitor), RE-135 (SFP High Range Monitor), RE-221 (Drumming Area Vent Gas Monitor), RE-325 (Drumming Area Exhaust Low-Range Gas Monitor), PAB Crane, SFP Platform Bridge, Truck Access Area, or Decon Area? |
| ☐ | ☒ | Does the proposed activity involve a change to Point Beach CLB design criteria for external events such as earthquakes, tornadoes, high winds, flooding, etc.? |
| ☐ | ☒ | Does the activity involve plant heavy load requirements or procedures for areas of the plant used to support cask loading/unloading activities? |
| ☐ | ☒ | Does the activity involve any potential for fire or explosion where casks are loaded, unloaded, transported or stored? |

If ANY of the Part V questions are answered YES, then a full 10 CFR 72.48 screening is required and answers to the questions in Part VI and Part VII are to be provided. If ALL the questions in Part V are answered NO, then check Parts VI and VII as not applicable. Complete Part VIII to document the conclusion that no 10 CFR 72.48 evaluation is required.

## PART VI (72.48) - DETERMINE IF THE CHANGE INVOLVES A ISFSI LICENSING BASIS *DESIGN FUNCTION*

(If ALL the questions in Part V are NO, then Part VI is ☒ NOT APPLICABLE.)

Compare the proposed activity to the relevant portions of the ISFSI licensing basis and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI Safety Analyses or plant/cask/ISFSI structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs whose function is relied upon for prevention of a radioactive release, **OR** whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, CoC conditions, or orders? |
| ☐ | ☐ | Does the activity involve a *method of evaluation* described in the ISFSI licensing basis? |
| ☐ | ☐ | Is the activity a *test or experiment*? (i.e., a non-passive activity which gathers data) |
| ☐ | ☐ | Does the activity exceed or potentially affect a cask *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If **THIS** questions is answered **YES**, a 10 CFR 72.48 Evaluation is required.) |

If the answers to **ALL** of these questions are **NO**, mark Parts VII as not applicable, and document the 10 CFR 72.48 screening in the conclusion section (Part VIII).

If any of the above questions are marked **YES**, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved.

## PART VII (72.48) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (NEI 96-07, Appendix B, Section B.4.2.1)

(If **ALL** the questions in Part V or Part VI are answered **NO**, then Part VII is ☒ NOT APPLICABLE.)

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any **YES** answer means that a 10 CFR 72.48 Evaluation is required; **EXCEPT** where noted in Part VII.3.

VII.1 Changes to the Facility or Procedures

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity adversely affect the *design function* of a plant, cask, or ISFSI SSC credited in safety analyses? |
| ☐ | ☐ | Does the activity adversely affect the method of performing or controlling the *design function* of a plant, cask, or ISFSI SSC credited in the safety analyses? |

If any answer is **YES**, a 10 CFR 72.48 Evaluation is required. If both answers are **NO**, describe the basis for the conclusion (attach additional discussion, as necessary):

VII.2 Changes to a Method of Evaluation

(If the activity does not involve a method of evaluation, these questions are ☐ NOT APPLICABLE.)

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in a cask SAR? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in a cask SAR? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary):

VII.3 Tests or Experiments

(If the activity is not a test or experiment, the questions in VII.3.a and VII.3.b are ☐ NOT APPLICABLE.)

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the cask ISFSI licensing basis? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the cask(s) or ISFSI facility? |

If the answer to both questions is NO, continue to VII.3.b. If the answer to EITHER question is YES, then briefly describe the basis.

b. Answer these additional questions ONLY for tests or experiments which do not meet the criteria given in VII.3.a above. If the answer to either question in VII.3.a is YES, then these three questions are ☐ NOT APPLICABLE:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity utilize or control a plant, cask or ISFSI facility SSC in a manner that is inconsistent with the analyses or descriptions in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity place the cask or ISFSI facility in a condition not previously evaluated or that could affect the capability of a plant, cask, or ISFSI SSC to perform its intended functions? |

If any answer in VII.3.b is YES, a 10 CFR 72.48 Evaluation is required. If the answers are all NO, describe the basis for the conclusion (attach additional discussion as necessary):

PART VIII - DOCUMENT THE CONCLUSION OF THE 10 CFR 72.48 SCREENING

Check all that apply:

A 10 CFR 72.48 Evaluation is ☐ required or ☒ NOT required. Obtain a screening number and provide the original to Records Management regardless of the conclusion of the 50.59 or 72.48 screening.

A VSC-24 cask Safety Analysis Report change is ☐ required or ☒ NOT required. If a VSC-24 cask SAR change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A change to the VSC-24 10 CFR 72.212 Site Evaluation Report is ☐ required or ☒ NOT required. If a VSC-24 10 CFR 72.212 Site Evaluation Report change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

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INITIATING DEVICES AND SETPOINTS
Annunciator C01D Window 1-2 Reflash

INSTRUMENT AIR HEADER PRESSURE LOW

TERMINAL STRIP LOCATION
UXE-10

1.0 AUTOMATIC ACTIONS

None

2.0 POSSIBLE CAUSES

2.1 Alarm on Window 1-2 at rear of panel C01

3.0 OPERATOR ACTIONS

3.1 Perform the following:

3.1.1 **IF** any AFW pump mini-recirc valve falls shut in conjunction with this alarm, **THEN** monitor and maintain minimum AFW flow **OR** stop the affected AFW pump as necessary to control S/G levels.

3.1.2 Respond per ARP C01 D 1-2, INSTRUMENT AIR HEADER PRESS LO.

4.0 SUPPLEMENTAL ACTIONS

4.1 Refer to AOP 5B, Loss of Instrument Air.

5.0 REFERENCES

5.1 Westinghouse drawing 499B666, Sheet 1667; Service/Instrument Air Alarms Main Control Board C01

5.2 ARP C01 D 1-2, INSTRUMENT AIR HEADER PRESS LO

5.3 AOP 5B, Loss of Instrument Air

A. PURPOSE

1. This procedure provides directions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, assess plant conditions, and direct the operator to the appropriate recovery procedure.

2. This procedure is applicable for all plant conditions where RCS hot leg temperature is greater than or equal to 350°F with accumulators in service, and assumes the RHR system is not in service for decay heat removal and all SI system components are available.

B. SYMPTOMS OR ENTRY CONDITIONS

1. The following are symptoms that require a reactor trip, if one has not occurred:

| REACTOR TRIP SIGNAL | SETPOINT |
|---|---|
| ΔT Overtemperature | Variable |
| ΔT Overpower | Variable |
| RCP Breaker Trip Low Voltage | STPT 21.1 |
| RCP Breaker Trip Low Frequency | STPT 21.1 |
| RCS Loop Low Flow | 93 % |
| S/G Low-Low Level | 25% |
| S/G Low Level with Flow Mismatch | 30% of span |
| PZR Pressure Low | 1925 psig |
| PZR Pressure High | 2365 psig |
| PZR Level High | 80% |
| NIS Power High Range, High Level | 107% |
| NIS Power Low Range, High Level | 20% |
| NIS Intermediate Range | Current equal to 25% |
| NIS source range | $5 \times 10^5$ counts/sec |
| Manual Reactor Trip | N/A |
| Turbine Trip | N/A |
| Safety Injection | N/A |

2. The following are symptoms of a reactor trip:

   o Any reactor trip annunciator - LIT
   o Rapid drop in neutron level indicated by nuclear instrumentation
   o All rod bottom lights - ON
   o Reactor trip and bypass breakers - OPEN

3. The following are symptoms that require a reactor trip and safety injection, if one has not occurred:

| SAFETY INJECTION SIGNAL | SETPOINT |
|---|---|
| PZR Low Pressure | 1735 psig |
| Steam Line Low Pressure | 530 psig |
| Containment High Pressure | 5 psig |
| Manual Safety Injection | N/A |

4. The following are symptoms of a reactor trip and safety injection:

   a. Safeguards pumps and associated cooling water pumps- RUNNING

      • SI pumps
      • RHR pumps
      • Component cooling water pumps
      • Service water pumps

   b. SI-Spray Active Status Panel white lights - ON

   c. Containment Isolation Panels "A" and "B" white lights - ON

5. This procedure is entered from the following procedures if SI actuates:

   o EOP-0.2 UNIT 1, NATURAL CIRCULATION COOLDOWN, FOLDOUT
   o EOP-0.3 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), FOLDOUT
   o EOP-0.4 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS), FOLDOUT

6. This procedure is entered from the following procedure when PZR pressure is less than 1735 PSIG:

   • EOP-0.1 UNIT 1, REACTOR TRIP RESPONSE, Step 15

7. This procedure is entered from the following procedure if PZR level cannot be maintained:

   • CSP-I.2 UNIT 1, RESPONSE TO LOW PRESSURIZER LEVEL, Step 7

8. This procedure is entered from the following procedures if RCS subcooling or PZR level cannot be maintained:

   o EOP-0.1 UNIT 1, REACTOR TRIP RESPONSE, FOLDOUT
   o EOP-0.2 UNIT 1, NATURAL CIRCULATION COOLDOWN, FOLDOUT
   o EOP-0.3 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS), FOLDOUT
   o EOP-0.4 UNIT 1, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS), FOLDOUT

9. This procedure is entered from the following procedures when power is restored to a 480 Vac safequards bus prior to placing ECCS components in pull-out:

   o ECA-0.0 UNIT 1, LOSS OF ALL AC POWER, Step 16
   o ECA-0.0 UNIT 1, LOSS OF ALL AC POWER, Step 26

10. This procedure is entered from other plant procedures when a reactor trip or safety injection has occurred.

C. REFERENCES

1. Technical Specifications for Point Beach Nuclear Plant

2. Final Safety Analysis Report for Point Beach Nuclear Plant

3. As-built plant drawings

4. Generic Technical Guidelines developed by the Westinghouse Owners Group (WOG). This consists of the following documents:

   a. Low pressure version of the WOG Optimal Recovery Guidelines, Status Trees, and Functional Restoration Guidelines

   b. Background documents for each low pressure version Optimal Recovery Guideline, Status Tree, and Functional Restoration Guideline

   c. WOG Emergency Response Guideline Executive Volume

   d. WOG Emergency Response Guideline Maintenance Program Summary

5. Calculation 97-0126, Service Water System LOCA - Recirculation Phase

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

NOTE

Steps 1 through 4 are immediate action steps.

| | | |
|---|---|---|
| 1 | Verify Reactor Trip:<br><br>• Check reactor trip breakers and bypass breakers - OPEN<br><br>  • 1-52/RTA<br>  • 1-52/RTB<br>  • 1-52/BYA<br>  • 1-52/BYB<br><br>• Check all rod bottom lights - LIT<br><br>• Check all rod position indicators - ON BOTTOM<br><br>• Check neutron flux - LOWERING<br><br>  • 1N-35<br>  • 1N-36 | Perform the following:<br><br>a. Manually trip reactor.<br><br>b. IF reactor will NOT trip, THEN perform the following:<br><br>  1) Deenergize rod drive motor generators by deenergizing 1B-01 and 1B-02.<br><br>  • 1B52-04B or 1A52-02<br>  • 1B52-05B or 1A52-15<br><br>  2) WHEN the reactor has tripped, THEN close the following breakers:<br><br>  • 1A52-02 for 1B-01<br>  • 1B52-04B for 1B-01<br>  • 1A52-15 for 1B-02<br>  • 1B52-05B for 1B-02<br><br>  3) IF reactor power is greater than or equal to 5% OR intermediate range power is rising, THEN perform the following:<br><br>  a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>  b) Go to CSP-S.1 UNIT 1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.<br><br>  4) As time permits, reenergize stripped MCCs.<br><br>  5) Dispatch operator to locally open reactor trip breakers and bypass breakers in rod control room. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 2 | Verify Turbine Trip: | |
| | a. Check turbine stop valves - BOTH SHUT:<br>o SL and SR - SHUT<br><u>OR</u><br>o Annunciator 1C03 1E1 4-3, TURBINE STOP VALVES TWO CLOSED - LIT<br><u>OR</u><br>o Turbine Valves Closed bistable lights - LIT | a. Shutdown turbine as follows:<br>1) Depress turbine trip pushbutton.<br>2) <u>IF</u> turbine will <u>NOT</u> trip, <u>THEN</u> perform the following:<br>a) Manually run back turbine.<br>b) Stop both EH oil pumps and place in pull-out.<br>c) <u>IF</u> turbine still has <u>NOT</u> tripped, <u>THEN</u> shut main steam isolation valves.<br>• 1MS-2018 for S/G A<br>• 1MS-2017 for S/G B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 3 | Verify Safeguards Buses Energized: | |
| | a. Check 4160 Vac safeguards buses - AT LEAST ONE ENERGIZED<br><br>o 1A-05, train A<br>o 1A-06, train B | a. Try to restore power to at least one bus:<br><br>1) Close any supply breaker.<br><br>o 1A52-57 for 1A-05<br>o 1A52-54 for 1A-06<br>o 1A52-77 for 1A-06<br><br>2) IF breakers will NOT close, THEN fast start and load any emergency diesel generator.<br><br>3) IF power can NOT be restored, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions for information only per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to ECA-0.0 UNIT 1, LOSS OF ALL AC POWER. |
| | b. Check 480 Vac safeguards buses - AT LEAST ONE ENERGIZED<br><br>o 1B-03, train A<br>o 1B-04, train B | b. Try to restore power to at least one bus:<br><br>1) Close any supply breaker.<br><br>o 1A52-58 for 1B-03<br>o 1B52-16B for 1B-03<br>o 1A52-84 for 1B-04<br>o 1B52-17B for 1B-04<br><br>2) IF power can NOT be restored, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions for information only per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to ECA-0.0 UNIT 1, LOSS OF ALL AC POWER. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 4 | Check If SI Is Actuated: | |
| | a. Check SI annunciators - ANY LIT<br>o {1C04-1B 4-2}, MANUAL SAFETY INJECTION<br>OR<br>o {1C04-1B 4-3}, CONTAINMENT PRESSURE HIGH<br>OR<br>o {1C04-1B 4-4}, PRESSURIZER LOW PRESSURE SI<br>OR<br>o {1C04-1B 4-5}, STEAM LINE A PRESSURE LOW-LOW<br>OR<br>o {1C04-1B 4-6}, STEAM LINE B PRESSURE LOW-LOW | a. Determine appropriate recovery actions:<br>1) Check if SI is required:<br>o Containment pressure - GREATER THAN 5 PSIG<br>OR<br>o Steam line A pressure - LESS THAN 530 PSIG<br>OR<br>o Steam line B pressure - LESS THAN 530 PSIG<br>OR<br>o PZR pressure - LESS THAN 1735 PSIG<br>OR<br>o PZR level - LESS THAN 10%<br>OR<br>o RCS subcooling - LESS THAN 35°F<br>2) IF SI is required, THEN perform the following:<br>a) Manually actuate both trains of SI and Containment Isolation.<br>b) OBSERVE NOTE PRIOR TO STEP 5 and go to Step 5.<br>3) IF SI is NOT required, THEN perform the following:<br>a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br>b) Go to EOP-0.1 UNIT 1, REACTOR TRIP RESPONSE. |
| | b. Check SI - BOTH TRAINS ACTUATED<br>• SI pumps - BOTH RUNNING<br>• RHR pumps - BOTH RUNNING | b. Manually actuate both trains of SI and Containment Isolation. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

NOTE

Foldout page shall be monitored throughout the remainder of this procedure.

5 Verify Automatic Actions Per ATTACHMENT A, AUTOMATIC ACTION VERIFICATION, While Continuing With This Procedure

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

CAUTION

If motor-driven auxiliary feedwater pump flow is greater than 240 gpm, its motor breaker may trip due to over current.

NOTE

If both units require AFW flow, at least one AFW pump must be aligned to each unit.

| | | |
|---|---|---|
| 6 | Verify Secondary Heat Sink Available: | |
| | a. Check level in at least one S/G - GREATER THAN [51%] 29% | a. Establish AFW flow as follows:<br>1) Manually start pumps and align valves as necessary to establish AFW flow greater than or equal to 200 gpm.<br>2) IF AFW flow greater than or equal to 200 gpm can NOT be established, THEN perform the following:<br>a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br>b) Go to CSP-H.1 UNIT 1, RESPONSE TO LOSS OF SECONDARY HEAT SINK. |
| | b. Control pumps and align valves as necessary to maintain S/G level between [51%] 29% and 65% | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 7 | Verify RCP Seal Cooling:<br><br>o Check labyrinth seal ΔP - GREATER THAN 20 INCHES<br><br>OR<br><br>o Check component cooling to RCP thermal barrier - NORMAL | IF seal cooling to any RCP is lost, THEN reestablish seal cooling:<br><br>a. Stop affected RCP(s).<br><br>o 1P-1A, loop A<br>o 1P-1B, loop B<br><br>b. Start pumps and align valves as necessary to reestablish component cooling water flow to all RCP thermal barriers.<br><br>c. IF all charging pumps are stopped, THEN reestablish seal injection flow:<br><br>1) Ensure adequate power is available to run one charging pump. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for KW ratings.<br><br>2) Start one charging pump at minimum speed for seal injection.<br><br>o 1P-2A, train A<br>o 1P-2B, train A<br>o 1P-2C, train B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 8 | Verify RCS Temperature Control:<br><br>a. Check RCS wide range cold leg temperatures:<br><br>• LESS THAN OR EQUAL TO 547°F<br><br>AND<br><br>• STABLE | Perform the following:<br><br>1. IF RCS cold leg temperature less than 547°F AND RCS temperatures are trending lower, THEN stabilize RCS temperature as follows:<br><br>a) Stop dumping steam.<br><br>b) IF cooldown continues, THEN control feed flow:<br><br>1) Reduce total feed flow.<br><br>2) Maintain total feed flow greater than or equal to 200 gpm until level greater than [51%] 29% in at least one S/G.<br><br>c) IF cooldown can NOT be stopped by controlling feed flow, THEN isolate steam lines:<br><br>1) Shut both main steam isolation valves.<br><br>• 1MS-2018 for S/G A<br>• 1MS-2017 for S/G B<br><br>2) Ensure main steam isolation bypass valves - BOTH SHUT<br><br>• 1MS-234 for S/G A<br>• 1MS-236 for S/G B<br><br>2. IF RCS cold leg temperature greater than 547°F OR RCS temperature trending higher, THEN stabilize RCS temperature at or below 547°F as follows:<br><br>o Dump steam to condenser.<br><br>OR<br><br>o Dump steam using atmospheric steam dumps. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 9 | Check PZR PORVs - BOTH SHUT<br><br>• 1RC-430<br>• 1RC-431C | IF PZR pressure less than 2335 psig, THEN stop PORV flow:<br><br>a. Manually shut affected PORVs.<br><br>b. IF any PORV can NOT be shut, THEN isolate that PORV:<br><br>1) Manually shut associated block valve.<br><br>o 1RC-515 for 1RC-431C<br>o 1RC-516 for 1RC-430<br><br>2) IF any open PORV can NOT be isolated, THEN perform the following:<br><br>a) Start monitoring Critical Safety Functions per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES.<br><br>b) Go to EOP-1 UNIT 1, LOSS OF REACTOR OR SECONDARY COOLANT. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 10 | Verify PZR Spray Valves Shut: | |
| | a. Check normal spray valves - BOTH SHUT<br>• 1RC-431A, loop A<br>• 1RC-431B, loop B | a. IF PZR pressure less than 2260 psig, THEN stop spray flow:<br>1) Manually shut both spray valves.<br>2) IF any spray valve can NOT be shut, THEN place manual override switch to close for failed spray valve(s).<br>o 1RC-431A-S for 1RC-431A<br>o 1RC-431B-S for 1RC-431B<br>3) IF any spray valve can NOT be shut using manual override, THEN stop RCP supplying failed spray valve(s).<br>o For 1RC-431A, stop RCP A<br>o For 1RC-431B, stop RCP B |
| | b. Check auxiliary spray valve - SHUT<br>• 1CV-296 | b. Stop auxiliary spray flow:<br>1) Manually shut auxiliary spray valve.<br>2) IF auxiliary spray valve can NOT be shut, THEN minimize charging and shut charging line flow control valve.<br>• 1HC-142 |
| 11 | Check If RCPs Should Remain Running: | |
| | a. Check RCPs - ANY RUNNING | a. Go to Step 12. |
| | b. Check RCS subcooling based on core exit thermocouples - GREATER THAN OR EQUAL TO [60°F] 30°F | b. IF at least one SI pump is running AND SI pump capable of delivering flow, THEN stop both RCPs.<br>• 1P-1A, loop A<br>• 1P-1B, loop B |
| 12 | Start Monitoring Critical Safety Functions Per CSP-ST.0 UNIT 1, CRITICAL SAFETY FUNCTION STATUS TREES | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 13 | Verify Containment Sump Recirculation Not Required: | |
| | a. Check RWST level - GREATER THAN OR EQUAL TO 60% | a. Go to EOP-1.3 UNIT 1, TRANSFER TO CONTAINMENT SUMP RECIRCULATION. |
| | b. Check RCS pressure - GREATER THAN [425 PSIG] 200 PSIG | b. IF RHR flow is greater than 450 gpm, THEN go to EOP-1.3 UNIT 1, TRANSFER TO CONTAINMENT SUMP RECIRCULATION. |
| 14 | Check If Secondary System Is Intact:<br>• No S/G pressure trending lower in an uncontrolled manner<br>AND<br>• No S/G completely depressurized | IF any S/G is faulted, THEN go to EOP-2 UNIT 1, FAULTED STEAM GENERATOR ISOLATION. |
| 15 | Check If S/G Tubes Are Intact:<br>• Check secondary system radiation levels - NORMAL<br>a. Condenser air ejector<br>• 1RE-215<br>• RE-225<br>b. S/G blowdown<br>• 1RE-219<br>• 1RE-222<br>c. Main steam line<br>• 1RE-231 for S/G A<br>• 1RE-232 for S/G B<br>• Request local surveys of main steam lines<br>• Request Chemistry to prepare for periodic activity samples of both S/Gs | IF conditions indicate a S/G tube rupture, THEN go to EOP-3 UNIT 1, STEAM GENERATOR TUBE RUPTURE. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 16 | Check If RCS Is Intact Inside Containment:<br><br>a. Check containment radiation levels - NORMAL<br><br>1) Containment<br><br>• 1RE-102, train A<br>• 1RE-107, train A<br><br>2) Containment high range<br><br>• 1RE-126, train A<br>• 1RE-127, train A<br>• 1RE-128, train A<br><br>b. On 1C20, check containment sump "A" level - NORMAL<br><br>• 1LI-958, train A<br>• 1LI-959, train A<br><br>c. Check containment pressure - NORMAL | Go to EOP-1 UNIT 1, LOSS OF REACTOR OR SECONDARY COOLANT. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 17 | Check If SI Should Be Terminated: | |
| | a. Check RCS subcooling based on core exit thermocouples - GREATER THAN 35°F | a. Go to Step 18. |
| | b. Verify secondary heat sink:<br>o Level in at least one S/G - GREATER THAN 29%<br>OR<br>o Total feed flow to S/Gs - GREATER THAN OR EQUAL TO 200 GPM | b. IF neither condition satisfied, THEN go to Step 18. |
| | c. Check RCS pressure:<br>• Pressure - GREATER THAN 1600 PSIG<br>AND<br>• Pressure - STABLE OR TRENDING HIGHER | c. Go to Step 18. |
| | d. Check PZR level - GREATER THAN 10% | d. Raise PZR level:<br>1) Raise charging flow.<br>2) Go to Step 18. |
| | e. Go to EOP-1.1 UNIT 1, SI TERMINATION | |
| 18 | Stabilize S/G Levels: | |
| | a. Check S/G levels - GREATER THAN 29% | a. Maintain total feed flow greater than or equal to 200 gpm until level in at least one S/G is greater than 29%. |
| | b. Control feed flow to maintain S/G levels between 29% and 65% | b. IF level in any S/G continues to rise in an uncontrolled manner, THEN go to EOP-3 UNIT 1, STEAM GENERATOR TUBE RUPTURE. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 19 | Check If S/G Tubes Are Intact:<br><br>• Check secondary system radiation levels - NORMAL<br><br>a. Condenser air ejector<br><br>• 1RE-215<br>• RE-225<br><br>b. S/G blowdown<br><br>• 1RE-219<br>• 1RE-222<br><br>c. Main steam line<br><br>• 1RE-231 for S/G A<br>• 1RE-232 for S/G B<br><br>• Request local surveys of main steam lines<br><br>• Request Chemistry to prepare for periodic activity samples of both S/Gs | IF conditions indicate a S/G tube rupture, THEN go to EOP-3 UNIT 1, STEAM GENERATOR TUBE RUPTURE. |

CAUTION

If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

20 Reset SI

21 Reset Containment Isolation

22 Reset 1B-03 And 1B-04 Non-Safeguards Equipment Lockouts

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

CAUTION

Placing loads on energized AC safeguards buses in excess of the power source's capacity could result in loss of the power source. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT, for KW ratings.

| | | |
|---|---|---|
| 23 | Check 4160 Vac Safeguards Buses - BOTH ENERGIZED BY OFFSITE POWER<br><br>• 1A-05, train A<br>• 1A-06, train B | Monitor EDG loading per AOP-22 UNIT 1, EDG LOAD MANAGEMENT, while continuing with this procedure. |
| 24 | Reestablish Instrument Air To Containment: | |
| | a. Start second instrument air compressor<br><br>o K-2A<br>o K-2B | |
| | b. Check instrument air header pressure - GREATER THAN 80 PSIG | b. Start service air compressors as necessary to establish instrument air header pressure greater than 80 psig.<br><br>o K-3A<br>o K-3B |
| | c. Open one and then open the other instrument air containment isolation valve<br><br>• 1IA-3047<br>• 1IA-3048 | c. <u>IF</u> no valve can be opened, <u>THEN</u> gag open one valve as follows:<br><br>1) Manually hold valve switch in open position.<br><br>2) Locally gag open valve. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 25 | Check If RCS Is Intact Outside Containment:<br><br>a. Request local radiation surveys in auxiliary building<br><br>b. Check auxiliary building radiation levels - NORMAL<br><br>• PPCS RMS screen, Page 104<br>• (New PPCS RMS GRID screen)<br>• Local surveys<br><br>c. Check auxiliary building sump levels - NORMAL<br><br>• C01A 1-11, AUXILIARY BUILDING -19 FT SUMP LEVEL HI - NOT LIT<br>• C01A 2-11, AUXILIARY BUILDING NORTH SUMP LEVEL HI - NOT LIT<br>• C01A 3-11, AUXILIARY BUILDING SOUTH SUMP LEVEL HI - NOT LIT | Perform the following:<br><br>1. Evaluate cause of abnormal conditions.<br><br>2. IF cause is a loss of RCS inventory outside containment, THEN go to ECA-1.2 UNIT 1, LOCA OUTSIDE CONTAINMENT. |
| 26 | Check PZR Relief Tank Conditions - NORMAL<br><br>• Pressure<br><br>• Temperature<br><br>• Level | Evaluate cause of abnormal conditions. |
| 27 | Check If RHR Pumps Should Be Stopped:<br><br>a. Check RCS pressure - GREATER THAN 200 PSIG<br><br>b. Check RCS pressure - STABLE OR TRENDING HIGHER<br><br>c. Stop both RHR pumps<br><br>• 1P-10A, train A<br>• 1P-10B, train B | a. Go to EOP-1 UNIT 1, LOSS OF REACTOR OR SECONDARY COOLANT.<br><br>b. Go to Step 28. |

**********
* d. Maintain RCS pressure greater than 200 psig

d. IF RCS pressure lowers in an uncontrolled manner to less than 200 psig, THEN restart RHR pumps to supply water to RCS. *
**********

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 28 | Verify Charging Flow: | |
| | a. Ensure RCS Loop A Cold Leg Normal Charging Isolation Valve - OPEN<br><br>• 1CV-1298 | |
| | b. Check charging pumps - AT LEAST ONE RUNNING<br><br>o 1P-2A, train A<br>o 1P-2B, train A<br>o 1P-2C, train B | b. Perform the following:<br><br>1) <u>IF</u> component cooling water flow to any RCP thermal barrier is lost, <u>THEN</u> locally shut affected RCP(s) seal injection throttle valve before starting charging pumps.<br><br>o 1CV-300A, RCP A<br>o 1CV-300B, RCP B<br><br>2) Start charging pumps as necessary to establish at least one running. |
| | c. Start additional charging pumps and adjust speed on running charging pumps as necessary to establish desired charging flow | |
| | d. Adjust charging line flow controller as necessary to maintain labyrinth seal ΔP greater than 20 inches<br><br>• 1HC-142 | |
| 29 | Check If Diesels Should Be Stopped: | |
| | a. Check 4160 Vac safeguards buses - ENERGIZED BY OFFSITE POWER<br><br>• 1A-05, train A<br>• 1A-06, train B | a. Restore offsite power to 4160 Vac safeguards buses. |
| | b. Stop all unloaded EDGs:<br><br>o OP-11A G-01, EMERGENCY DIESEL GENERATOR G-01<br>o OP-11A G-02, EMERGENCY DIESEL GENERATOR G-02<br>o OP-11B, EMERGENCY DIESEL GENERATOR G-03 (G-04) | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|
| 30 | Ensure Miscellaneous Electrical Loads Are Energized: | |
| | a. Ensure MCCs - ENERGIZED<br>• 1B-31, 1B52-14C, train A<br>• B-43, 1B52-21C, train B | |
| | b. Check battery chargers supplying DC buses - ENERGIZED<br>o D-07<br>o D-09<br>o D-108<br>o D-109 | b. Restore battery chargers:<br>1) Close affected battery charger supply contactor.<br>2) <u>IF</u> contactor does <u>NOT</u> close <u>OR</u> battery charger will <u>NOT</u> operate, <u>THEN</u> restore battery chargers per AOP-0.0, VITAL DC SYSTEM MALFUNCTION, while continuing with this procedure. |
| | c. Ensure cavity cooling fan - ONE RUNNING<br>o 1W-4A, train A<br>o 1W-4B, train A | |
| | d. Check cable spreading room ventilation operating:<br>1) Check cable spreading room recirc fans - ONE RUNNING<br>o W-13A1<br>o W-13A2<br>2) Check CSR chilled water recirc pumps - ONE RUNNING<br>o P-111A<br>o P-111B | d. Restore cable spreading room ventilation per OI-90, CONTROL, COMPUTER, AND CABLE SPREADING ROOM VENTILATION SYSTEMS. |
| | e. Start additional loads as necessary to meet current plant conditions. Refer to AOP-22 UNIT 1, EDG LOAD MANAGEMENT | |
| 31 | Return To <u>Step 8</u> | |

-END-

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 1 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A1 | Verify Feedwater Isolation: | |
| | a. Check main feed lines isolated:<br><br>1) Feedwater regulating control valves - BOTH SHUT<br><br>• 1CS-466 for S/G A<br>• 1CS-476 for S/G B<br><br>2) Feedwater regulating bypass valves - BOTH SHUT<br><br>• 1CS-480 for S/G A<br>• 1CS-481 for S/G B | a. <u>IF</u> any main feedline can <u>NOT</u> be isolated, <u>THEN</u> perform the following:<br><br>a) Trip main feed pumps.<br><br>• 1P-28A<br>• 1P-28B<br><br>b) Place condensate pumps in pull-out.<br><br>• 1P-25A<br>• 1P-25B<br><br>c) Stop heater drain tank pumps.<br><br>• 1P-27A<br>• 1P-27B<br>• 1P-27C |
| | b. Check main feed pumps - BOTH TRIPPED<br><br>• 1P-28A<br>• 1P-28B | b. Trip main feed pumps. |
| | c. Check MFP discharge MOVs - BOTH SHUT<br><br>• 1CS-2190, train A<br>• 1CS-2189, train B | c. Manually shut valves. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 2 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A2 | Verify Containment Isolation: | |
| | a. Check containment isolation panels "A" and "B" - ALL LIGHTS LIT | a. Perform the following:<br>1) Manually actuate Containment Isolation.<br>2) IF any valve open AND flow path NOT required, THEN shut valve(s). Refer to ATTACHMENT B. |
| | b. Check other valves - SHUT<br>• RS-SA-9 , Unit 1 steam supply to rad waste system<br>• Any valve which may be open under administrative control | b. Manually shut valve(s). |
| A3 | Verify AFW Actuation: | |
| | a. Check motor-driven AFW pumps - BOTH RUNNING<br>• P-38A, train A<br>• P-38B, train B | a. Establish AFW flow as follows:<br>1) Ensure steam supply valves to turbine-driven AFW pump - BOTH OPEN<br>• 1MS-2020, train A<br>• 1MS-2019, train B<br>2) WHEN SI sequence complete, THEN manually start motor-driven AFW pumps. |
| | b. Check S/G levels - BOTH LESS THAN [51%] 25% | b. Go to Step A4. |
| | c. Ensure steam supply valves to turbine-driven AFW pump - BOTH OPEN<br>• 1MS-2020, train A<br>• 1MS-2019, train B | |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 3 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A4 | Check SI Pumps - BOTH RUNNING<br><br>• 1P-15A, train A<br>• 1P-15B, train B | WHEN SI sequence complete, THEN establish SI flow as follows:<br><br>a. Manually start SI pumps.<br><br>b. IF any SI pump can NOT be started, THEN isolate system boundary as follows:<br><br>1) Place affected SI pump in pull-out.<br><br>2) Ensure affected SI pump suction valve shut.<br><br>o 1SI-896A, train A<br>o 1SI-896B, train B |
| A5 | Check RHR Pumps - BOTH RUNNING<br><br>• 1P-10A, train A<br>• 1P-10B, train B | WHEN SI sequence complete, THEN establish RHR flow as follows:<br><br>a. Manually start RHR pumps.<br><br>b. IF any RHR pump can NOT be started, THEN isolate system boundary as follows:<br><br>1) Place affected RHR pump in pull-out.<br><br>2) Ensure affected RHR pump suction valve shut.<br><br>o 1SI-856A, train A<br>o 1SI-856B, train B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 4 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A6 | Check Component Cooling Water Pumps - ONLY ONE RUNNING<br><br>o 1P-11A, train A<br>o 1P-11B, train B | Establish one component cooling water pump running as follows:<br><br>a. <u>IF</u> no component cooling water pump running, <u>THEN</u> perform the following:<br><br>1) Stop all RCP(s).<br><br>• 1P-1A, loop A<br>• 1P-1B, loop B<br><br>2) Manually start one component cooling water pump by placing control switch to stop and then auto-after-stop.<br><br>3) Match flags for running and stopped pumps.<br><br>b. <u>IF</u> both component cooling water pumps running, <u>THEN</u> place one pump in standby. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 5 of 10)
AUTOMATIC ACTION VERIFICATION

A7 Verify Service Water System Alignment:

a. Check service water pumps - SIX RUNNING

- P-32A, train A
- P-32B, train A
- P-32F, train A
- P-32C, train B
- P-32D, train B
- P-32E, train B

a. WHEN SI sequence complete, THEN manually start pumps.

b. Check service water isolation valves - SHUT

- At least one spent fuel pool heat exchanger A isolation MOV
  - o SW-2927A, inlet MOV
  - o SW-2930A, discharge MOV
- At least one spent fuel pool heat exchanger B isolation MOV
  - o SW-2927B, inlet MOV
  - o SW-2930B, discharge MOV
- At least one auxiliary building A/C condenser isolation MOV
  - o SW-2816, train A
  - o SW-4479, train B
- At least one water treatment system inlet MOV
  - o SW-4478, train A
  - o SW-2817, train B

b. Perform the following:

1) Manually shut valve(s).

2) IF any isolation valve will NOT shut, THEN locally shut valve or associated manual isolation valve.

c. Locally at blowdown evap panel C-180, check at least one radwaste service water valve shut

- o SW-LW-61, train A
- o SW-LW-62, train B

c. Perform the following:

1) Locally shut valve(s).

2) IF any valve will NOT shut, THEN locally shut associated manual isolation valve.

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 6 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A8 | Verify Containment Accident Cooling Units Running: | |
| | a. Check containment accident recirculation fans - ALL RUNNING<br>• 1W-1A1, train A<br>• 1W-1B1, train A<br>• 1W-1C1, train B<br>• 1W-1D1, train B | a. <u>WHEN</u> SI sequence complete, <u>THEN</u> manually start fans. |
| | b. Check containment ventilation cooler outlet emergency FCVs - BOTH OPEN<br>• 1SW-2907, train A<br>• 1SW-2908, train B | b. Manually open containment ventilation cooler outlet emergency FCVs. |
| | c. Check annunciator C01B 2-3, UNIT 1 CONTAINMENT RECIRC COOLERS WATER FLOW LOW - CLEAR | c. Perform the following:<br>1) Ensure non-affected unit's service water isolation valves - BOTH SHUT<br>• 2SW-2907 , train A<br>• 2SW-2908 , train B<br>2) Isolate service water to non-safety loads as necessary to clear annunciator. |
| A9 | Check Control Room Fans Armed: | |
| | a. Check Control Room Charcoal Filter Fan W-14A - WHITE LIGHT OFF | a. At MCC 1B-32, depress Control Circuit Arming pushbutton for Control Room charcoal filter fan W-14A.<br>• 1B52-329B |
| | b. Check Control Room Recirc Fan W-13B2 - WHITE LIGHT OFF | b. At MCC 1B-42, depress Control Circuit Arming pushbutton for Control Room recirc fan W-13B2.<br>• 1B52-428M |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 7 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A10 | Check Control Room Ventilation - IN AN ACCIDENT MODE<br><br>• Control Room recirc fans - AT LEAST ONE RUNNING<br><br>o W-13B1<br>o W-13B2<br><br>• Control Room damper solenoid valve - PURPLE LIGHT LIT | Align Control Room ventilation per OI-90, CONTROL, COMPUTER, AND CABLE SPREADING ROOM VENTILATION SYSTEMS. |
| A11 | Check If Main Steam Lines Can Remain Open: | |
| | a. Check MSIVs - ANY OPEN | a. Go to Step A12. |
| | b. Check containment pressure - LESS THAN OR EQUAL TO 15 PSIG | b. Isolate both steam lines as follows:<br><br>1) Shut both main steam isolation valves.<br><br>• 1MS-2018 for S/G A<br>• 1MS-2017 for S/G B<br><br>2) Go to Step A12. |
| | c. Check high-high steam flow bistable lights - NOT LIT | c. Ensure main steam isolation valve on affected main steam line(s) shut.<br><br>o 1MS-2018 for S/G A<br>o 1MS-2017 for S/G B |
| | d. Check high steam flow bistable lights - NOT LIT | d. IF RCS average temperature is less than 543°F, THEN ensure main steam isolation valve on affected main steam line(s) shut.<br><br>o 1MS-2018 for S/G A<br>o 1MS-2017 for S/G B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 8 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A12 | Verify Proper SI Valve Alignment:<br><br>a. Check Unit 1 SI Active status panel - ALL LIGHTS LIT<br><br>b. Check Unit 1 SI - Spray Ready status panel - NO LIGHTS LIT | Manually align pumps and valves as necessary to establish proper SI alignment. |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 9 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A13 | Verify Containment Spray Not Required:<br><br>a. Check containment pressure recorder - HAS REMAINED LESS THAN 25 PSIG<br><br>• 1PR-968<br>• 1PR-969 | Establish containment spray as follows:<br><br>1. Check containment spray actuated:<br><br>• Annunciator {C01 B 2-6}, CONTAINMENT SPRAY - LIT<br><br>2. <u>IF</u> containment spray has <u>NOT</u> actuated, <u>THEN</u> manually actuate containment spray.<br><br>3. Verify the following equipment status:<br><br>a) Ensure containment spray pump discharge MOVs - ALL OPEN<br><br>• 1SI-860A for 1P-14A<br>• 1SI-860B for 1P-14A<br>• 1SI-860C for 1P-14B<br>• 1SI-860D for 1P-14B<br><br>b) Ensure containment spray pumps - AT LEAST ONE RUNNING<br><br>o 1P-14A, train A<br>o 1P-14B, train B<br><br>c) Shutdown one train of containment spray as follows:<br><br>1) Place one containment spray pump in pull-out.<br><br>o 1P-14A, train A<br>o 1P-14B, train B<br><br>2) Ensure suction on idle spray pump shut.<br><br>o 1SI-870A for 1P-14A<br>o 1SI-870B for 1P-14B<br><br>d) <u>WHEN</u> containment spray has been actuated for greater than two minutes, <u>THEN</u> ensure at least one spray additive eductor suction valve open.<br><br>o 1SI-836A, train A<br>o 1SI-836B, train B |

| STEP | ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED |
|---|---|---|

ATTACHMENT A
(Page 10 of 10)
AUTOMATIC ACTION VERIFICATION

| | | |
|---|---|---|
| A14 | Verify SI Flow: | |
| | a. Check RCS wide range pressure - LESS THAN 1400 PSIG | a. Return to procedure and step in effect. |
| | b. Check SI pumps - FLOW INDICATED<br>• 1FI-925, train A<br>• 1FI-924, train B | b. Manually start pumps and align valves as necessary to establish SI pump flow. |
| | c. Check RCS wide range pressure - LESS THAN [425 PSIG] 200 PSIG | c. Return to procedure and step in effect. |
| | d. Check RHR pumps - FLOW INDICATED<br>• 1FI-626, train A<br>• 1FI-928, train B | d. Manually start pumps and align valves as necessary to establish RHR pump flow. |

-END-

ATTACHMENT B
(Page 1 of 2)
CONTAINMENT ISOLATION VALVES

| PANEL A | | |
|---|---|---|
| COMPONENT | DESCRIPTION | TRAIN |
| 1CV-1296 | Auxiliary charging line | A |
| 1RC-538 | Pressurizer relief tank to gas analyzer | A |
| 1WG-1788 | Reactor coolant drain tank to gas analyzer | A |
| 1WL-1698 | Reactor coolant drain tank to -19 ft sump | A |
| 1WL-1003A | Reactor coolant drain tank pump suction | A |
| 1WL-1003B | Reactor coolant drain tank pump suction | A |
| 1RC-508 | Reactor makeup water to containment | A or B |
| 1RC-539 | Pressurizer relief tank to gas analyzer | B |
| 1WG-1789 | Reactor coolant drain tank to gas analyzer | B |
| 1SI-846 | Accumulator nitrogen supply | A or B |
| 1WL-1721 | Reactor coolant drain tank pumps suction | B |
| 1VNPSE-3244 | Containment purge supply | A |
| 1VNPSE-3212 | Containment purge exhaust | A |
| 1WL-1723 | Sump A drain | A |
| 1SC-951 | Pressurizer steam sample | A |
| 1SC-953 | Pressurizer liquid sample | A |
| 1VNPSE-3245 | Containment purge supply | B |
| 1VNPSE-3213 | Containment purge exhaust | B |
| 1WL-1728 | Sump A drain | B |
| 1SC-966A | Pressurizer steam sample | A or B |
| 1SC-966B | Pressurizer liquid sample | A or B |

ATTACHMENT B
(Page 2 of 2)
CONTAINMENT ISOLATION VALVES

| PANEL B | | |
|---|---|---|
| COMPONENT | DESCRIPTION | TRAIN |
| 1CC-769 | Component cooling water outlet from excess letdown heat exchanger | A or B |
| 1CV-313 | Reactor coolant pump seal return | A |
| 1CV-371 | Letdown line | A |
| 1MS-5958 | Steam generator blowdown | A or B |
| 1MS-5959 | Steam generator blowdown | A or B |
| 1WG-1786 | Reactor coolant drain tank vent | A |
| 1CV-313A | Reactor coolant pump seal return | B |
| 1CV-371A | Letdown line | B |
| 1WG-1787 | Reactor coolant drain tank vent | B |
| 1RM-3200C | RE-211/212 supply | A |
| 1RM-3200A | RE-211/212 return | A or B |
| 1MS-2083 | Steam generator A sample | A or B |
| 1MS-2084 | Steam generator B sample | A or B |
| 1SC-955 | Reactor coolant hot leg sample | A |
| 1IA-3047 | Instrument air line | A or B |
| 1RM-3200B | RE-211/212 supply | B |
| 1SC-966C | Reactor coolant hot leg sample | A or B |
| 1IA-3048 | Instrument air line | A or B |

-END-

FOLDOUT PAGE FOR EOP-0 UNIT 1

1. RCP TRIP CRITERIA
IF all conditions listed below occur, THEN trip both RCPs:
- RCS subcooling - LESS THAN [60°F] 30°F
- SI pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW
- Operator controlled cooldown - NOT IN PROGRESS

2. FAULTED S/G ISOLATION CRITERIA
IF any S/G pressure trending lower in an uncontrolled manner OR any S/G completely depressurized, THEN the following may be performed:
a. Isolate feed flow to faulted S/G.
b. Maintain total feed flow greater than or equal to 200 gpm until narrow range level in at least one S/G is greater than [51%] 29%.

3. RUPTURED S/G ISOLATION CRITERIA
IF any S/G level rises in an uncontrolled manner OR any S/G has abnormal radiation, AND narrow range level in affected S/G(s) is greater than [51%] 29%, THEN feed flow may be isolated to affected S/G(s).

4. AFW SUPPLY SWITCHOVER CRITERIA
IF CST level lowers to less than 8 feet, THEN switch to alternate AFW suction supply per AOP-23 UNIT 1, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY.

5. ADVERSE CONTAINMENT CONDITIONS
IF any condition listed below occurs, THEN adverse containment setpoint values in brackets, [], shall be used:
o Containment pressure - GREATER THAN 10 PSIG
OR
o Containment radiation level - GREATER THAN OR EQUAL TO $10^5$ R/HR
OR
o Integrated dose to containment - GREATER THAN $10^6$ R

6. AFW MINIMUM FLOW REQUIREMENTS
IF any AFW pump mini-recirc valve fails shut OR annunciator C01 A 1-9, INSTRUMENT AIR HEADER PRESSURE LOW in alarm, THEN monitor and maintain minimum AFW flow or stop the affected AFW pump as necessary to control S/G levels.
o P-38A minimum flow - GREATER THAN 50 GPM
o P-38B minimum flow - GREATER THAN 50 GPM
o P-29 minimum flow - GREATER THAN 75 GPM

SMB 12-7-01

NIM

Nuclear Power Business Unit

# TEMPORARY CHANGE REVIEW AND APPROVAL

*Note Refer to NP 1 2 3, Temporary Procedure Changes, for requirements*



## I - INITIATION

Doc Number EOP-0 Current Rev 36 Unit PB2 Temp Change No. 2001-0872

Document Title REACTOR TRIP OR SAFETY INJECTION

Existing Effective Temporary Changes

Brief Description ADDED FOP ITEM TO ADDRESS AFW MINIMUM FLOW

(Identify specific changes on Form PBF-0026c, Document Review and Approval Continuation, and include with the package)

☒ Initiate PBF-0026h and include with the change.

Other documents required to be effective concurrently with the temporary change. NONE

Changes pre-screened according to NP 10.3.1? ☒ NO ☐ YES (If Yes, list references and criteria on PBF-0026c/refer to NP 10 3 1)

Screening completed according to NP 10.3.1? ☐ NA ☒ YES

Safety Evaluation Required? ☒ NO ☐ YES (If Yes a revision may be processed or final reviews and approvals shall be obtained before implementing)

Determine if the change constitutes a Change Of Intent to the procedure by evaluating the following questions
(If any answers are YES, a revision may be processed or final reviews and approvals shall be obtained before implementing)

Will the proposed change:

| | YES | NO |
|---|---|---|
| 1. Require a change to, affect or invalidate a requirement, commitment, evaluation or description in the Current Licensing Basis (as defined in NP 10 3.1)? | ☐ | ☒ |
| 2. Cause an increase in magnitude, significance or impact such that it should be processed as a revision? | ☐ | ☒ |
| 3. Delete or modify a prerequisite, initial condition, precaution, limitation or other steps that could have safety significance or affect the procedure's margin of safety? | ☐ | ☒ |
| 4. Delete QC hold points, Independent Verification or Concurrent Check steps without the related step(s) that require the performance also being deleted? | ☐ | ☒ |
| 5. Change Tech Spec or other regulatory acceptance criteria other than for re-baselining purposes? | ☐ | ☒ |
| 6. Require a change to the procedure Purpose or change the procedure classification? | ☐ | ☒ |

Initiated By (print/sign) BUB WARTENBERG / [signature] Date 11/30/01

## II - INITIAL APPROVAL

This change is correct and complete, can be performed as written, and does not adversely affect personnel or nuclear safety, or Plant operating conditions.

Group Supervisor (print/sign) TERRANCE BOCCHI / [signature] Date 11/30/01
(Cannot be the Initiator)

This change does not adversely affect Plant operating conditions. (Safety Related procedures only)

Senior Reactor Operator (print/sign) Rick Merkes / [signature] Date 11/30/01
(Cannot be the Initiator or Group Supervisor)

## III - PROCEDURE OWNER REVIEW

☒ Permanent ☐ One-time Use ☐ Expiration Date, Event or Condition:

☐ Hold change until procedure completed (final review and approval still required within 14 days of initial approval)

☐ QR/MSS Review NOT Required (Admin/NNSR only) ☒ QR Review Required ☐ MSS Review Required (Reference NP 1 6 5)

Procedure Owner (print/sign) C. R. Gardner / [signature] Date 11/30/01
This Change and supporting requirements correctly completed and processed

## IV - FINAL REVIEW AND APPROVAL

(Must be completed within 14 days of initial approval) (The Initiator, QR and Approval Authority shall be independent from each other)

QR/MSS (print/sign) Terry N. Mount / [signature] Date 11/30/01
Indicates 50 59/72.48 applicability assessed, any necessary screenings/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed.

MSS Meeting No

Approval Authority (print/sign) D. SCHOON / [signature] Date 11-30-01

## V - REVISION INFORMATION FOR PERMANENT CHANGES

Post Typing Review (print/sign) TERRANCE BOCCHI / [signature] Date 12/5/01
Indicates temporary change(s) incorporated exactly as approved and no other changes made to document

Incorporated into Revision Number 37 Effective Date DEC 14 2001

MICROFILMED

REC'D DEC 17 2001

FEB 11 2002

DEC 1 4 2001

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL CONTINUATION



Doc Number EOP-0 Revision 36 Unit 2

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number 2001-0872

Description of Changes:

| Step * | Change/Reason |
|---|---|
| FOP | CHANGE: Added AFW minimum flow requirements for the AFW pumps<br>REASON: To prevent damage to the AFW pumps on a loss of instrument air due to the AFW pump mini-recirc valve failing shut with minimum flow through the pump is less than required to cool the pump. |

Other Comments

* Note: Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to reviewers



References NP 1.3 NP 1.2.3

Sing + 12-7-01

Point Beach Nuclear Plant

# TEMPORARY CHANGE AFFECTED MANUAL LOCATION



Procedure Number EOP-0 Revision 36 Unit PB2

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number 2001-0872

## I - IMMEDIATELY AFTER INITIAL APPROVAL ON PBF-0026e (Non-Intent changes)
(after Final Approval if change of intent involved)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☐ Copy included in work package for field implementation. (WO No _____ ) | |
| ☒ Copy filed in Control Room temp change binder (Operations only). | 11/30/01 |
| ☒ Original change package provided to GRC to obtain Procedure Owner Review (e.g., Owner review may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 11/30/01 |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |

Performed By (print and sign) Sherri Schweigl / Sherri Schweigl Date 11/30/01

## II - PROCEDURE OWNER REVIEW ON PBF-0026e
(may be performed by OA II, Procedure Writer, etc.)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☒ Copy sent to Document Control Distribution Lead for Master File. (Not required for one-time use change) | 12-3-01 |
| ☐ Copy filed in Group satellite file. (Not required for one-time use changes) | |
| ☐ Copy filed in Group one-time use file. | |
| ☒ Original Temp Change provided to DOS to obtain Final Approvals (e.g., final approval may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 11/30/01 |
| ☒ U1/U2 CR (3) | 11/30/01 |
| ☒ PAB | 12-3-01 |
| ☒ CPS Shop | ↓ |
| ☒ OPS Office | ↓ |
| ☒ Simulator (Trg OA II) | ↓ |
| ☐ | |

Performed By (print and sign) Sherri Schweigl / Sherri Schweigl Date 11/30/01

of Proposed Activity: Unit 1 EOP-0 - Rev. 35, Unit 2 EOP-0 - Rev. 36, Unit 1 EOP-0.1, Rev 24 Unit 2 EOP-0.1 - Rev 23

Associated Reference(s) #: CR 01-2278 Action 2

| | | | | |
|---|---|---|---|---|
| Prepared by: | Bob Wartenberg<br>Name (Print) | Signature | Date: | 11/30/01 |
| Reviewed by | Clayton Graves<br>Name (Print) | Signature | Date. | 11/30/01 |

PART I (50.59/72.48) - DESCRIBE THE PROPOSED ACTIVITY AND SEARCH THE PLANT AND ISFSI LICENSING BASIS (Resource Manual 5.3.1)

NOTE: The "NMC 10 CFR 50.59 Resource Manual" (Resource Manual) and NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 10 CFR 50.59 and 10 CFR 72.48 screenings.

I.1 Describe the proposed activity and the scope of the activity being covered by this screening. (The 10 CFR 50.59 / 72.48 review of other portions of the proposed activity may be documented via the applicability and pre-screening process requirements in NP 5.1.8.) Appropriate descriptive material may be attached.

A foldout-page item is being added to Units 1 & 2 procedures EOP-0 and EOP-0.1. The foldout page item, "AFW Minimum Flow Requirements", shall address minimum flow required by the AFW pumps in the case of a failed closed mini-recirc valve on any running AFW pumps.

I.2 Search the PBNP Current Licensing Basis (CLB) as follows: Final Safety Analysis Report (FSAR), FSAR Change Requests (FCRs) with assigned numbers, the Fire Protection Evaluation Report (FPER), the CLB (Regulatory) Commitment Database, the Technical Specifications (both Custom and Improved), the Technical Specifications Bases, and the Technical Requirements Manual. Search the ISFSI licensing basis as follows: VSC-24 Safety Analysis Report, the VSC-24 Certificate of Compliance, the CLB (Regulatory) Commitment Database, and the VSC-24 10 CFR 72.212 Site Evaluation Report. Describe the pertinent design function(s), performance requirements, and methods of evaluation for both the plant and for the cask/ISFSI as appropriate. Identify where the pertinent information is described in the above documents (by document section number and title). (Resource Manual 5.3.1 and NEI 96-07, App. B, B.2)

FSAR 10.2, Auxiliary Feedwater System

I.3 Does the proposed activity involve a change to any Custom or Improved Technical Specification (ITS)? Changes to Technical Specifications require a License Amendment Request (Resource Manual Section 5.3.1.2).

Technical Specification Change: ☐ Yes ☒ No

If a Technical Specification change is required, explain what the change should be and why it is required.

I.4 Does the proposed activity involve a change to the terms, conditions or specifications incorporated in any VSC-24 cask Certificate of Compliance (CoC)? Changes to a VSC-24 cask Certificate of Compliance require a CoC amendment request

☐ Yes ☒ No

If a storage cask Certificate of Compliance change is required, explain what the change should be and why it is required.

---------- 10 CFR 50.59 SCREENING ----------

PART II (50.59) - DETERMINE IF THE CHANGE INVOLVES A *DESIGN FUNCTION* (Resource Manual 5 3 2)

Compare the proposed activity to the relevant CLB descriptions, and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☒ | ☐ | Does the proposed activity involve Safety Analyses or structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs whose failure could initiate a transient (e g , reactor trip, loss of feedwater, etc.) or accident, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve CLB-described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications? |
| ☐ | ☒ | Does the activity involve a *method of evaluation* described in the FSAR? |
| ☐ | ☒ | Is the activity a *test or experiment*? (i e., a non-passive activity which gathers data) |
| ☐ | ☒ | Does the activity exceed or potentially affect a *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If THIS questions is answered YES, a 10 CFR 50.59 Evaluation is required.) |

If the answers to ALL of these questions are NO, mark Part III as not applicable, document the 10 CFR 50.59 screening in the clusion section (Part IV), then proceed directly to Part V - 10 CFR 72.48 Pre-screening Questions

If any of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved

FSAR 10.2 states each AFW pump has an AOV controlled recirc line back to the CST to ensure minimum flow to dissipate heat. This change ensures the minimum AFW flow requirements will be maintained on any running AFW pump in the case of a failed shut AFW mini-recirc flow control valve.

PART III (50.59) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (Resource Manual 5.3.3)

If ALL the questions in Part II are answered NO, then Part III is ☐ NOT APPLICABLE.

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 50 59 Evaluation is required; EXCEPT where noted in Part III.3.

III.1 CHANGES TO THE FACILITY OR PROCEDURES

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☒ | Does the activity adversely affect the *design function* of an SSC credited in safety analyses? |
| ☐ | ☒ | Does the activity adversely affect the method of performing or controlling the *design function* of an SSC credited in the safety analyses? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion as necessary)

This change ensures that minimum recirc flow requirements as stated in FSAR 10 2 are not violated.

2 CHANGES TO A METHOD OF EVALUATION

(If the activity does not involve a method of evaluation, these questions are ☒ NOT APPLICABLE )

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in the CLB? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in the CLB? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary).

III 3 TESTS OR EXPERIMENTS

If the activity is not a test or experiment, the questions in III.3.a and III 3 b are ☒ NOT APPLICABLE.

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the CLB? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the facility? |

If the answer to BOTH questions in V.3.a is NO, continue to III.3.b. If the answer to EITHER question is YES, then describe the basis.

b. Answer these additional questions ONLY for tests or experiments which do NOT meet the criteria given in III 3.a above If the answer to either question in III.3.a is YES, then these three questions are ☐ NOT APPLICABLE.

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the CLB? |
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is inconsistent with the analyses or descriptions in the CLB? |
| ☐ | ☐ | Does the activity place the facility in a condition not previously evaluated or that could affect the capability of an SSC to perform its intended functions? |

If any answer in III.3.b is YES, a 10 CFR 50.59 Evaluation is required. If the answers in III.3.b are ALL NO, describe the basis for the conclusion (attach additional discussion as necessary).

Part IV - 10 CFR 50.59 SCREENING CONCLUSION (Resource Manual 5.. 4).

heck all that apply:

A 10 CFR 50 59 Evaluation is ☐ required or ☒ NOT required.

A Point Beach FSAR change is ☐ required or ☒ NOT required. If an FSAR change is required, then initiate an FSAR Change Request (FCR) per NP 5.2 6.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A Technical Specification Bases change is ☐ required or ☒ NOT required. If a change to the Technical Specification Bases is required, then initiate a Technical Specification Bases change per NP 5.2.15.

A Technical Requirements Manual change is ☐ required or ☒ NOT required. If a change to the Technical Requirements Manual is required, then initiate a Technical Requirements Manual change per NP 5.2 15.

---------- 10 CFR 72.48 SCREENING ----------

NOTE: NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 72.48 screenings.

PART V (72.48) - 10 CFR 72.48 INITIAL SCREENING QUESTIONS

Part V determines if a full 10 CFR 72.48 screening is required to be completed (Parts VI and VII) for the proposed activity.

| YES | NO | QUESTION |
|---|---|---|
| | ☒ | Does the proposed activity involve IN ANY MANNER the dry fuel storage cask(s), the cask transfer/transport equipment, any ISFSI facility SSC(s), or any ISFSI facility monitoring as follows: Multi-Assembly Sealed Basket (MSB), MSB Transfer Cask (MTC), MTC Lifting Yoke, Ventilated Concrete Cask (VCC), Ventilated Storage Cask (VSC), VSC Transporter (VCST), ISFSI Storage Pad Facility, ISFSI Storage Pad Data/Communication Links, or PPCS/ISFSI Continuous Temperature Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) installed in the plant specifically added to support cask loading/unloading activities, as follows: Cask Dewatering System (CDW), Cask Reflood System (CRF), or Hydrogen Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) needed for plant operation which are also used to support cask loading/unloading activities, as follows: Spent Fuel Pool (SFP), SFP Cooling and Filtration (SF), Primary Auxiliary Building Ventilation System (VNPAB), Drumming Area Ventilation System (VNDRM). RE-105 (SFP Low Range Monitor), RE-135 (SFP High Range Monitor), RE-221 (Drumming Area Vent Gas Monitor), RE-325 (Drumming Area Exhaust Low-Range Gas Monitor), PAB Crane, SFP Platform Bridge, Truck Access Area, or Decon Area? |
| ☐ | ☒ | Does the proposed activity involve a change to Point Beach CLB design criteria for external events such as earthquakes, tornadoes, high winds, flooding, etc.? |
| ☐ | ☒ | Does the activity involve plant heavy load requirements or procedures for areas of the plant used to support cask loading/unloading activities? |
| ☐ | ☒ | Does the activity involve any potential for fire or explosion where casks are loaded, unloaded, transported or stored? |

If ANY of the Part V questions are answered YES, then a full 10 CFR 72.48 screening is required and answers to the questions in Part VI and Part VII are to be provided. If ALL the questions in Part V are answered NO, then check Parts VI and VII as not applicable. Complete Part VIII to document the conclusion that no 10 CFR 72.48 evaluation is required.

PART VI (72.48) - DETERMINE IF THE CHANGE INVOLVES A ISFSI LICENSING BASIS *DESIGN FUNCTION*

(If ALL the questions in Part V are NO, then Part VI is ☒ NOT APPLICABLE )

Compare the proposed activity to the relevant portions of the ISFSI licensing basis and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI Safety Analyses or plant/cask/ISFSI structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs whose function is relied upon for prevention of a radioactive release, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, CoC conditions, or orders? |
| ☐ | ☐ | Does the activity involve a *method of evaluation* described in the ISFSI licensing basis? |
| ☐ | ☐ | Is the activity a *test or experiment*? (i e , a non-passive activity which gathers data) |
| ☐ | ☐ | Does the activity exceed or potentially affect a cask *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If THIS questions is answered YES, a 10 CFR 72.48 Evaluation is required ) |

If the answers to ALL of these questions are NO, mark Parts VII as not applicable, and document the 10 CFR 72.48 screening in the conclusion section (Part VIII)

If any of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved.

PART VII (72.48) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (NEI 96-07, Appendix B, Section B.4.2.1)

(If ALL the questions in Part V or Part VI are answered NO, then Part VII is ☒ NOT APPLICABLE )

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 72.48 Evaluation is required; EXCEPT where noted in Part VII.3.

VII.1 Changes to the Facility or Procedures

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity adversely affect the *design function* of a plant, cask, or ISFSI SSC credited in safety analyses? |
| ☐ | ☐ | Does the activity adversely affect the method of performing or controlling the *design function* of a plant, cask, or ISFSI SSC credited in the safety analyses? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary).

VII.2 Changes to a Method of Evaluation

(If the activity does not involve a method of evaluation, these questions are ☐ NOT APPLICABLE )

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in a cask SAR? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in a cask SAR? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary)

VII.3 Tests or Experiments

(If the activity is not a test or experiment, the questions in VII.3 a and VII.3.b are ☐ NOT APPLICABLE.)

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the cask ISFSI licensing basis? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the cask(s) or ISFSI facility? |

If the answer to both questions is NO, continue to VII 3.b. If the answer to EITHER question is YES, then briefly describe the basis.

b. Answer these additional questions ONLY for tests or experiments which do not meet the criteria given in VII.3.a above. If the answer to either question in VII.3 a is YES, then these three questions are ☐ NOT APPLICABLE:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity utilize or control a plant, cask or ISFSI facility SSC in a manner that is inconsistent with the analyses or descriptions in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity place the cask or ISFSI facility in a condition not previously evaluated or that could affect the capability of a plant, cask, or ISFSI SSC to perform its intended functions? |

If any answer in VII.3 b is YES, a 10 CFR 72.48 Evaluation is required. If the answers are all NO, describe the basis for the conclusion (attach additional discussion as necessary):

ART VIII - DOCUMENT THE CONCLUSION OF THE 10 CFR 72.48 SCREENING

Check all that apply:

A 10 CFR 72.48 Evaluation is ☐ required or ☒ NOT required. Obtain a screening number and provide the original to Records Management regardless of the conclusion of the 50.59 or 72.48 screening.

A VSC-24 cask Safety Analysis Report change is ☐ required or ☒ NOT required. If a VSC-24 cask SAR change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A change to the VSC-24 10 CFR 72.212 Site Evaluation Report is ☐ required or ☒ NOT required. If a VSC-24 10 CFR 72.212 Site Evaluation Report change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

A. PURPOSE

1. This procedure provides directions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, assess plant conditions, and direct the operator to the appropriate recovery procedure.

2. This procedure is applicable for all plant conditions where RCS hot leg temperature is greater than or equal to 350°F with accumulators in service, and assumes the RHR system is not in service for decay heat removal and all SI system components are available.

B. SYMPTOMS OR ENTRY CONDITIONS

1. The following are symptoms that require a reactor trip, if one has not occurred:

| REACTOR TRIP SIGNAL | SETPOINT |
|---|---|
| ΔT Overtemperature | Variable |
| ΔT Overpower | Variable |
| RCP Breaker Trip Low Voltage | STPT 21.1 |
| RCP Breaker Trip Low Frequency | STPT 21.1 |
| RCS Loop Low Flow | 93 % |
| S/G Low-Low Level | 25% |
| S/G Low Level with Flow Mismatch | 30% of span |
| PZR Pressure Low | 1925 psig |
| PZR Pressure High | 2365 psig |
| PZR Level High | 80% |
| NIS Power High Range, High Level | 107% |
| NIS Power Low Range, High Level | 20% |
| NIS Intermediate Range | Current equal to 25% |
| NIS source range | $5 \times 10^5$ counts/sec |
| Manual Reactor Trip | N/A |
| Turbine Trip | N/A |
| Safety Injection | N/A |

FOLDOUT PAGE FOR EOP-0 UNIT 2

1. RCP TRIP CRITERIA
   IF all conditions listed below occur, THEN trip both RCPs:
   - RCS subcooling - LESS THAN [60°F] 30°F
   - SI pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW
   - Operator controlled cooldown - NOT IN PROGRESS

2. FAULTED S/G ISOLATION CRITERIA
   IF any S/G pressure trending lower in an uncontrolled manner OR any S/G completely depressurized, THEN the following may be performed:
   a. Isolate feed flow to faulted S/G.
   b. Maintain total feed flow greater than or equal to 200 gpm until narrow range level in at least one S/G is greater than [51%] 29%.

3. RUPTURED S/G ISOLATION CRITERIA
   IF any S/G level rises in an uncontrolled manner OR any S/G has abnormal radiation, AND narrow range level in affected S/G(s) is greater than [51%] 29%, THEN feed flow may be isolated to affected S/G(s).

4. AFW SUPPLY SWITCHOVER CRITERIA
   IF CST level lowers to less than 8 feet, THEN switch to alternate AFW suction supply per AOP-23 UNIT 2, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY.

5. ADVERSE CONTAINMENT CONDITIONS
   IF any condition listed below occurs, THEN adverse containment setpoint values in brackets, [], shall be used:
   - o Containment pressure - GREATER THAN 10 PSIG
     OR
   - o Containment radiation level - GREATER THAN OR EQUAL TO $10^5$ R/HR
     OR
   - o Integrated dose to containment - GREATER THAN $10^6$ R

6. AFW MINIMUM FLOW REQUIREMENTS
   IF any AFW pump mini-recirc valve fails shut, THEN maintain minimum flow or stop the affected AFW pump as necessary to control S/G levels.
   - o P-38A minimum flow - GREATER THAN 50 GPM
   - o P-38B minimum flow - GREATER THAN 50 GPM
   - o P-29 minimum flow - GREATER THAN 75 GPM

Nuclear Power Business Unit

# TEMPORARY CHANGE REVIEW AND APPROVAL

*Note: Refer to NP 1.2.3, Temporary Procedure Changes, for requirements*



## I - INITIATION

Doc Number EOP-0 Current Rev 37 Unit PB2 Temp Change No TRC 2001-0914

Document Title REACTOR TRIP OR SAFETY INJECTION

Existing Effective Temporary Changes

Brief Description MODIFY FOP FOR MINIMUM AFW FLOW TO INCLUDE LOW IA HDR PRESSURE

(Identify specific changes on Form PBF-0026c, Document Review and Approval Continuation, and include with the package)

☒ Initiate PBF-0026h and include with the change.

Other documents required to be effective concurrently with the temporary change:

Changes pre-screened according to NP 10.3.1? ☒ NO ☐ YES (If Yes, list references and criteria on PBF-0026c, refer to NP 10.3.1)

Screening completed according to NP 10.3.1? ☐ NA ☒ YES

Safety Evaluation Required? ☒ NO ☐ YES (If Yes, a revision may be processed or final reviews and approvals shall be obtained before implementing)

Determine if the change constitutes a Change Of Intent to the procedure by evaluating the following questions
(If any answers are YES, a revision may be processed or final reviews and approvals shall be obtained before implementing)

| Will the proposed change: | YES | NO |
|---|---|---|
| 1. Require a change to, affect or invalidate a requirement, commitment, evaluation or description in the Current Licensing Basis (as defined in NP 10.3.1)? | ☐ | ☒ |
| 2. Cause an increase in magnitude, significance or impact such that it should be processed as a revision? | ☐ | ☒ |
| 3. Delete or modify a prerequisite, initial condition, precaution, limitation or other steps that could have safety significance or affect the procedure's margin of safety? | ☐ | ☒ |
| 4. Delete QC hold points, Independent Verification or Concurrent Check steps without the related step(s) that require the performance also being deleted? | ☐ | ☒ |
| 5. Change Tech Spec or other regulatory acceptance criteria other than for re-baselining purposes? | ☐ | ☒ |
| 6. Require a change to the procedure Purpose or change the procedure classification? | ☐ | ☒ |

Initiated By (print/sign) C. L. Grover / [signature] Date 12/20/01

## II - INITIAL APPROVAL

This change is correct and complete, can be performed as written, and does not adversely affect personnel or nuclear safety, or Plant operating conditions.

Group Supervisor (print/sign) TIEVANDORN-SCIT / [signature] Date 12/20/01
(Cannot be the Initiator)

This change does not adversely affect Plant operating conditions (Safety Related procedures only)

Senior Reactor Operator (print/sign) M. Schug / [signature] Date 12/20/01
(Cannot be the Initiator or Group Supervisor)

## III - PROCEDURE OWNER REVIEW

☒ Permanent ☐ One-time Use ☐ Expiration Date, Event or Condition:

☐ Hold change until procedure completed (final review and approval still required within 14 days of initial approval)

☒ QR/MSS Review NOT Required (Admin/NNSR only) ☒ QR Review Required ☐ MSS Review Required (Reference NP 1.6.5)

12/20/01 Procedure Owner (print/sign) BOB WHITTINGTON / [signature] Date 12/20/01

This Change and supporting requirements correctly completed and processed

## IV - FINAL REVIEW AND APPROVAL

(Must be completed within 14 days of Initial approval) (The Initiator, QR and Approval Authority shall be independent from each other)

QR/MSS (print/sign) B.B. WHITTINGTON / [signature] Date 12/20/01

Indicates 50.59/72.48 applicability assessed, any necessary screenings/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed

MSS Meeting No

Approval Authority (print/sign) D. SCHOON / [signature] Date 12-20-01

## V - REVISION INFORMATION FOR PERMANENT CHANGES

Post Typing Review (print/sign) [illegible] / [signature] Date 1/9/02

Indicates temporary change(s) incorporated exactly as approved and no other changes made to document

Incorporated into Revision Number 38 Effective Date JAN [illegible]

REC'D JAN 11 2002

SMB 1-7-01

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL CONTINUATION



Doc Number: EOP-0 Revision 37 Unit 2

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number TCN 2001-0914

Description of Changes:

| Step * | Change/Reason |
|---|---|
| FOLDOUT PAGE | CHANGE: ADDED REFERENCE TO ANNUNCIATOR C01 A 1-9, IA HEADER LOW PRESSURE FOR THE AFW MINIMUM FLOW REQUIREMENTS.<br>REASON: TO ENSURE MINIMUM FLOW IS MAINTAINED THROUGH THE AFW PUMPS DURING OPERATION. |

Other Comments

* Note: Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to reviewers



References NP 1.1.3, NP 1.2.3

Point Beach Nuclear Plant

# TEMPORARY CHANGE AFFECTED MANUAL LOCATION



Procedure Number EOP-0 Revision 37 Unit PB2

Title REACTOR TRIP OR SAFETY INJECTION

Temporary Change Number TCN 2001-0914

## I - IMMEDIATELY AFTER INITIAL APPROVAL ON PBF-0026e (Non-Intent changes)
(after Final Approval if change of intent involved)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☐ Copy included in work package for field implementation. (WO No. _____ ) | |
| ☒ Copy filed in Control Room temp change binder (Operations only). | 12-20-01 |
| ☒ Original change package provided to KJW to obtain Procedure Owner Review (e.g., Owner review may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 12-20-01 |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |

Performed By (print and sign) Carol Schroeder / Carol Schroeder Date 12-20-01

## II - PROCEDURE OWNER REVIEW ON PBF-0026e
(may be performed by OA II, Procedure Writer, etc.)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☒ Copy sent to Document Control Distribution Lead for Master File. (Not required for one-time use change) | 12-20-01 |
| ☐ Copy filed in Group satellite file. (Not required for one-time use changes) | |
| ☐ Copy filed in Group one-time use file. | |
| ☒ Original Temp Change provided to DJS to obtain Final Approvals (e.g., final approval may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 12-20-01 |
| ☒ U1/U2 | 12-20-01 |
| ☒ NAD | |
| ☒ Ops Shop | |
| ☒ Ops Office | |
| ☒ Simulator (by OA II) | |
| ☐ | |

Performed By (print and sign) Carol Schroeder / Carol Schroeder Date 12-20-01

A. PURPOSE

1. This procedure provides directions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, assess plant conditions, and direct the operator to the appropriate recovery procedure.

2. This procedure is applicable for all plant conditions where RCS hot leg temperature is greater than or equal to 350°F with accumulators in service, and assumes the RHR system is not in service for decay heat removal and all SI system components are available.

B. SYMPTOMS OR ENTRY CONDITIONS

1. The following are symptoms that require a reactor trip, if one has not occurred:

| REACTOR TRIP SIGNAL | SETPOINT |
|---|---|
| ΔT Overtemperature | Variable |
| ΔT Overpower | Variable |
| RCP Breaker Trip Low Voltage | STPT 21.1 |
| RCP Breaker Trip Low Frequency | STPT 21.1 |
| RCS Loop Low Flow | 93 % |
| S/G Low-Low Level | 25% |
| S/G Low Level with Flow Mismatch | 30% of span |
| PZR Pressure Low | 1925 psig |
| PZR Pressure High | 2365 psig |
| PZR Level High | 80% |
| NIS Power High Range, High Level | 107% |
| NIS Power Low Range, High Level | 20% |
| NIS Intermediate Range | Current equal to 25% |
| NIS source range | $5 \times 10^5$ counts/sec |
| Manual Reactor Trip | N/A |
| Turbine Trip | N/A |
| Safety Injection | N/A |

FOLDOUT PAGE FOR EOP-0 UNIT 2

1. RCP TRIP CRITERIA

IF all conditions listed below occur, THEN trip both RCPs:

- RCS subcooling - LESS THAN [60°F] 30°F
- SI pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW
- Operator controlled cooldown - NOT IN PROGRESS

2. FAULTED S/G ISOLATION CRITERIA

IF any S/G pressure trending lower in an uncontrolled manner OR any S/G completely depressurized, THEN the following may be performed:

a. Isolate feed flow to faulted S/G.

b. Maintain total feed flow greater than or equal to 200 gpm until narrow range level in at least one S/G is greater than [51%] 29%.

3. RUPTURED S/G ISOLATION CRITERIA

IF any S/G level rises in an uncontrolled manner OR any S/G has abnormal radiation, AND narrow range level in affected S/G(s) is greater than [51%] 29%, THEN feed flow may be isolated to affected S/G(s).

4. AFW SUPPLY SWITCHOVER CRITERIA

IF CST level lowers to less than 8 feet, THEN switch to alternate AFW suction supply per AOP-23 UNIT 2, ESTABLISHING ALTERNATE AFW SUCTION SUPPLY.

5. ADVERSE CONTAINMENT CONDITIONS

IF any condition listed below occurs, THEN adverse containment setpoint values in brackets, [], shall be used:

o Containment pressure - GREATER THAN 10 PSIG

OR

o Containment radiation level - GREATER THAN OR EQUAL TO $10^5$ R/HR

OR

o Integrated dose to containment - GREATER THAN $10^6$ R

6. AFW MINIMUM FLOW REQUIREMENTS

IF any AFW pump mini-recirc valve fails shut OR annunciator C01 A 1-9, INSTRUMENT AIR HEADER PRESSURE LOW in alarm, THEN monitor and maintain minimum AFW flow or stop the affected AFW pump as necessary to control S/G levels.

o P-38A minimum flow - GREATER THAN 50 GPM

o P-38B minimum flow - GREATER THAN 50 GPM

o P-29 minimum flow - GREATER THAN 75 GPM

Nuclear Power Business Unit

# TEMPORARY CHANGE REVIEW AND APPROVAL

*Note: Refer to NP 1.2.3, Temporary Procedure Changes, for requirements.*



## I - INITIATION

Doc Number EOP-0.1 Current Rev 24 Unit PB1 Temp Change No. 2001-0873

Document Title REACTOR TRIP RESPONSE

Existing Effective Temporary Changes

Brief Description ADDED FOP ITEM TO ADDRESS AFW MINIMUM FLOW

(Identify specific changes on Form PBF-0026c, Document Review and Approval Continuation, and include with the package)

☒ Initiate PBF-0026h and include with the change.

Other documents required to be effective concurrently with the temporary change: NONE

Changes pre-screened according to NP 10.3.1? ☒ NO ☐ YES (If Yes, list references and criteria on PBF-0026c)(refer to NP 10.3.1)

Screening completed according to NP 10.3.1? ☐ NA ☒ YES

Safety Evaluation Required? ☒ NO ☐ YES (If Yes, a revision may be processed or final reviews and approvals shall be obtained before implementing)

Determine if the change constitutes a Change Of Intent to the procedure by evaluating the following questions.
(If any answers are YES, a revision may be processed or final reviews and approvals shall be obtained before implementing)

| Will the proposed change: | YES | NO |
|---|---|---|
| 1. Require a change to, affect or invalidate a requirement, commitment, evaluation or description in the Current Licensing Basis (as defined in NP 10.3.1)? | ☐ | ☒ |
| 2. Cause an increase in magnitude, significance or impact such that it should be processed as a revision? | ☐ | ☒ |
| 3. Delete or modify a prerequisite, initial condition, precaution, limitation or other steps that could have safety significance or affect the procedure's margin of safety? | ☐ | ☒ |
| 4. Delete QC hold points, Independent Verification or Concurrent Check steps without the related step(s) that require the performance also being deleted? | ☐ | ☒ |
| 5. Change Tech Spec or other regulatory acceptance criteria other than for re-baselining purposes? | ☐ | ☒ |
| 6. Require a change to the procedure Purpose or change the procedure classification? | ☐ | ☒ |

Initiated By (print/sign) Bob Westerberg / [signature] Date 11/30/01

## II - INITIAL APPROVAL

This change is correct and complete, can be performed as written, and does not adversely affect personnel or nuclear safety, or Plant operating conditions.

Group Supervisor (print/sign) T. Vandenbosch / [signature] Date 11/30/01
(Cannot be the Initiator)

This change does not adversely affect Plant operating conditions. (Safety Related procedures only)

Senior Reactor Operator (print/sign) Rick Merkes / [signature] Date 11/30/01
(Cannot be the Initiator or Group Supervisor)

## III - PROCEDURE OWNER REVIEW

☒ Permanent ☐ One-time Use ☐ Expiration Date, Event or Condition:

☐ Hold change until procedure completed (final review and approval still required within 14 days of initial approval)

☐ QR/MSS Review NOT Required (Admin/NNSR only) ☒ QR Review Required ☐ MSS Review Required (Reference NP 1.6.5)

Procedure Owner (print/sign) C. R. Gadow / [signature] Date 11/30/01
This Change and supporting requirements correctly completed and processed

## IV - FINAL REVIEW AND APPROVAL

(Must be completed within 14 days of initial approval) (The Initiator, QR and Approval Authority shall be independent from each other)

QR/MSS (print/sign) Terry N. Mount / Terry N. Mount Date 11/30/01
Indicates 50.59/72.48 applicability assessed, any necessary screenings/evaluations performed, determination made as to whether additional cross-disciplinary review required, and if required, performed.

MSS Meeting No.

Approval Authority (print/sign) D. Schoon / [signature] Date 11-30-01

## V - REVISION INFORMATION FOR PERMANENT CHANGES

Post Typing Review (print/sign) T. Vandenbosch / [signature] Date 12/5/01
Indicates temporary change(s) incorporated exactly as approved and no other changes made to document

Incorporated into Revision Number 25 Effective Date DEC 14 2001

REC'D DEC 17 2001

MICROFILMED FEB 01 2002

SMB 12-7-01

DEC 1 4 2001

Point Beach Nuclear Plant
DOCUMENT REVIEW AND APPROVAL CONTINUATION



Doc Number EOP-0.1 Revision 24 Unit 1

Title REACTOR TRIP RESPONSE

Temporary Change Number 2001-0873

Description of Changes:

| Step * | Change/Reason |
|---|---|
| FOP | CHANGE: Added AFW minimum flow requirements for the AFW pumps.<br>REASON: To prevent damage to the AFW pumps on a loss of instrument air due to the AFW pump mini-recirc valve failing shut with minimum flow through the pump is less than required to cool the pump |

Other Comments

* Note Recording of Step Number(s) is not required for multiple occurrences of identical information or when not beneficial to reviewers

Point Beach Nuclear Plant

# TEMPORARY CHANGE AFFECTED MANUAL LOCATION



Procedure Number EOP-0.1 Revision 24 Unit PB1

Title REACTOR TRIP RESPONSE

Temporary Change Number 2001-0873

## I - IMMEDIATELY AFTER INITIAL APPROVAL ON PBF-0026e (Non-Intent changes)
(after Final Approval if change of intent involved)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☐ Copy included in work package for field implementation. (WO No. _____ ) | |
| ☒ Copy filed in Control Room temp change binder (Operations only). | 11/30/01 |
| ☒ Original change package provided to GRC to obtain Procedure Owner Review (e.g., Owner review may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 11/30/01 |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |
| ☐ | |

Performed By (print and sign) Sharon Schweigl / Sharon Schweigl Date 11/30/01

## II - PROCEDURE OWNER REVIEW ON PBF-0026e
(may be performed by OA II, Procedure Writer, etc.)

| This procedure change has been processed as follows: (Manual/Location) | Date Performed |
|---|---|
| ☒ Copy sent to Document Control Distribution Lead for Master File. (Not required for one-time use change) | 12-3-01 |
| ☐ Copy filed in Group satellite file. (Not required for one-time use changes) | |
| ☐ Copy filed in Group one-time use file. | |
| ☒ Original Temp Change provided to DCS to obtain Final Approvals (e.g. final approval may be coordinated by In-Group OA II, Procedure Writer, Procedure Supervisor, etc.) | 11/30/01 |
| ☒ U1/U2 CR (3) | 11/30/01 |
| ☒ PAB | 12-3-01 |
| ☒ OPS Shop | ↓ |
| ☒ OPS Office | ↓ |
| ☒ Simulator (Trng OA II) | ↓ |
| ☐ | |

Performed By (print and sign) Sharon Schweigl / Sharon Schweigl Date 11/30/01

of Proposed Activity: Unit 1 EOP-0 - Rev. 35, Unit 2 EOP-0 - Rev. 36, Unit 1 EOP-0.1, Rev 24, Unit 2 EOP-0.1 - Rev 23

Associated Reference(s) #: CR 01-2278 Action 2

| | | | |
|---|---|---|---|
| Prepared by: | Bob Wartenberg (Name (Print)) | (Signature) | Date: 11/30/01 |
| Reviewed by | Clayton Graves (Name (Print)) | (Signature) | Date 11/30/01 |

PART I (50.59/72.48) - DESCRIBE THE PROPOSED ACTIVITY AND SEARCH THE PLANT AND ISFSI LICENSING BASIS (Resource Manual 5.3.1)

NOTE: The "NMC 10 CFR 50.59 Resource Manual" (Resource Manual) and NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 10 CFR 50.59 and 10 CFR 72.48 screenings.

I.1 Describe the proposed activity and the scope of the activity being covered by this screening. (The 10 CFR 50.59 / 72.48 review of other portions of the proposed activity may be documented via the applicability and pre-screening process requirements in NP 5.1.8.) Appropriate descriptive material may be attached.

A foldout-page item is being added to Units 1 & 2 procedures EOP-0 and EOP-0.1. The foldout page item, "AFW Minimum Flow Requirements", shall address minimum flow required by the AFW pumps in the case of a failed closed mini-recirc valve on any running AFW pumps.

I.2 Search the PBNP Current Licensing Basis (CLB) as follows: Final Safety Analysis Report (FSAR), FSAR Change Requests (FCRs) with assigned numbers, the Fire Protection Evaluation Report (FPER), the CLB (Regulatory) Commitment Database, the Technical Specifications (both Custom and Improved), the Technical Specifications Bases, and the Technical Requirements Manual. Search the ISFSI licensing basis as follows: VSC-24 Safety Analysis Report, the VSC-24 Certificate of Compliance, the CLB (Regulatory) Commitment Database, and the VSC-24 10 CFR 72.212 Site Evaluation Report. Describe the pertinent design function(s), performance requirements, and methods of evaluation for both the plant and for the cask/ISFSI as appropriate. Identify where the pertinent information is described in the above documents (by document section number and title). (Resource Manual 5.3.1 and NEI 96-07, App. B, B.2)

FSAR 10.2, Auxiliary Feedwater System

I.3 Does the proposed activity involve a change to any Custom or Improved Technical Specification (ITS)? Changes to Technical Specifications require a License Amendment Request (Resource Manual Section 5.3.1.2)

Technical Specification Change: ☐ Yes ☒ No

If a Technical Specification change is required, explain what the change should be and why it is required.

I.4 Does the proposed activity involve a change to the terms, conditions or specifications incorporated in any VSC-24 cask Certificate of Compliance (CoC)? Changes to a VSC-24 cask Certificate of Compliance require a CoC amendment request

☐ Yes ☒ No

If a storage cask Certificate of Compliance change is required, explain what the change should be and why it is required

---------- 10 CFR 50.59 SCREENING ----------

PART II (50.59) - DETERMINE IF THE CHANGE INVOLVES A *DESIGN FUNCTION* (Resource Manual 5.3.2)

Compare the proposed activity to the relevant CLB descriptions, and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☒ | ☐ | Does the proposed activity involve Safety Analyses or structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve SSCs whose failure could initiate a transient (e.g., reactor trip, loss of feedwater, etc.) or accident, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☒ | Does the proposed activity involve CLB-described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications? |
| ☐ | ☒ | Does the activity involve a *method of evaluation* described in the FSAR? |
| ☐ | ☒ | Is the activity a *test or experiment*? (i.e., a non-passive activity which gathers data) |
| ☐ | ☒ | Does the activity exceed or potentially affect a *design basis limit for a fission product barrier (DBLFPB)*? (NOTE: If THIS questions is answered YES, a 10 CFR 50.59 Evaluation is required.) |

If the answers to ALL of these questions are NO, mark Part III as not applicable, document the 10 CFR 50.59 screening in the Conclusion section (Part IV), then proceed directly to Part V - 10 CFR 72.48 Pre-screening Questions

If any of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) involved.

FSAR 10.2 states each AFW pump has an AOV controlled recirc line back to the CST to ensure minimum flow to dissipate heat. This change ensures the minimum AFW flow requirements will be maintained on any running AFW pump in the case of a failed shut AFW mini-recirc flow control valve.

PART III (50.59) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (Resource Manual 5.3.3)

If ALL the questions in Part II are answered NO, then Part III is ☐ NOT APPLICABLE.

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 50.59 Evaluation is required; EXCEPT where noted in Part III.3.

III.1 CHANGES TO THE FACILITY OR PROCEDURES

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☒ | Does the activity adversely affect the *design function* of an SSC credited in safety analyses? |
| ☐ | ☒ | Does the activity adversely affect the method of performing or controlling the *design function* of an SSC credited in the safety analyses? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion as necessary)

This change ensures that minimum recirc flow requirements as stated in FSAR 10.2 are not violated

CHANGES TO A METHOD OF EVALUATION

(If the activity does not involve a method of evaluation, these questions are ☒ NOT APPLICABLE.)

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in the CLB? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in the CLB? |

If any answer is YES, a 10 CFR 50.59 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary).

III.3 TESTS OR EXPERIMENTS

If the activity is not a test or experiment, the questions in III.3.a and III.3.b are ☒ NOT APPLICABLE.

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the CLB? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the facility? |

If the answer to BOTH questions in V.3.a is NO, continue to III.3.b. If the answer to EITHER question is YES, then describe the basis.

b. Answer these additional questions ONLY for tests or experiments which do NOT meet the criteria given in III.3.a above. If the answer to either question in III.3.a is YES, then these three questions are ☐ NOT APPLICABLE.

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the CLB? |
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is inconsistent with the analyses or descriptions in the CLB? |
| ☐ | ☐ | Does the activity place the facility in a condition not previously evaluated or that could affect the capability of an SSC to perform its intended functions? |

If any answer in III.3.b is YES, a 10 CFR 50.59 Evaluation is required. If the answers in III.3.b are ALL NO, describe the basis for the conclusion (attach additional discussion, as necessary):

rt IV - 10 CFR 50.59 SCREENING CONCLUSION (Resource Manual 5.3.4)

Check all that apply:

A 10 CFR 50.59 Evaluation is ☐ required or ☒ NOT required.

A Point Beach FSAR change is ☐ required or ☒ NOT required. If an FSAR change is required, then initiate an FSAR Change Request (FCR) per NP 5.2.6.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A Technical Specification Bases change is ☐ required or ☒ NOT required. If a change to the Technical Specification Bases is required, then initiate a Technical Specification Bases change per NP 5.2.15.

A Technical Requirements Manual change is ☐ required or ☒ NOT required. If a change to the Technical Requirements Manual is required, then initiate a Technical Requirements Manual change per NP 5.2.15.

---------- 10 CFR 72.48 SCREENING ----------

NOTE: NEI 96-07, Appendix B, Guidelines for 10 CFR 72.48 Implementation should be used for guidance to determine the proper responses for 72.48 screenings.

PART V (72.48) - 10 CFR 72.48 INITIAL SCREENING QUESTIONS

Part V determines if a full 10 CFR 72.48 screening is required to be completed (Parts VI and VII) for the proposed activity.

| S | NO | QUESTION |
|---|---|---|
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER the dry fuel storage cask(s), the cask transfer/transport equipment, any ISFSI facility SSC(s), or any ISFSI facility monitoring as follows: Multi-Assembly Sealed Basket (MSB), MSB Transfer Cask (MTC), MTC Lifting Yoke, Ventilated Concrete Cask (VCC), Ventilated Storage Cask (VSC), VSC Transporter (VCST), ISFSI Storage Pad Facility, ISFSI Storage Pad Data/Communication Links or PPCS/ISFSI Continuous Temperature Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) installed in the plant specifically added to support cask loading/unloading activities, as follows: Cask Dewatering System (CDW), Cask Reflood System (CRF), or Hydrogen Monitoring System? |
| ☐ | ☒ | Does the proposed activity involve IN ANY MANNER SSC(s) needed for plant operation which are also used to support cask loading/unloading activities, as follows: Spent Fuel Pool (SFP), SFP Cooling and Filtration (SF), Primary Auxiliary Building Ventilation System (VNPAB), Drumming Area Ventilation System (VNDRM), RE-105 (SFP Low Range Monitor), RE-135 (SFP High Range Monitor), RE-221 (Drumming Area Vent Gas Monitor), RE-325 (Drumming Area Exhaust Low-Range Gas Monitor), PAB Crane, SFP Platform Bridge, Truck Access Area, or Decon Area? |
| ☐ | ☒ | Does the proposed activity involve a change to Point Beach CLB design criteria for external events such as earthquakes, tornadoes, high winds, flooding, etc.? |
| ☐ | ☒ | Does the activity involve plant heavy load requirements or procedures for areas of the plant used to support cask loading/unloading activities? |
| ☐ | ☒ | Does the activity involve any potential for fire or explosion where casks are loaded, unloaded, transported or stored |

If ANY of the Part V questions are answered YES, then a full 10 CFR 72.48 screening is required and answers to the questions in rt VI and Part VII are to be provided. If ALL the questions in Part V are answered NO, then check Parts VI and VII as not pplicable. Complete Part VIII to document the conclusion that no 10 CFR 72.48 evaluation is required.

RT VI (72.48) - DETERMINE IF THE CHANGE INVOLVES A ISFSI LICENSING BASIS *DESIGN FUNCTION*

(If ALL the questions in Part V are NO, then Part VI is ☒ NOT APPLICABLE.)

Compare the proposed activity to the relevant portions of the ISFSI licensing basis and answer the following questions:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI Safety Analyses or plant/cask/ISFSI structures, systems and components (SSCs) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs that support SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve plant, cask or ISFSI SSCs whose function is relied upon for prevention of a radioactive release, OR whose failure could impact SSC(s) credited in the Safety Analyses? |
| ☐ | ☐ | Does the proposed activity involve cask/ISFSI described SSCs or procedural controls that perform functions that are required by, or otherwise necessary to comply with, regulations, license conditions, CoC conditions, or orders? |
| ☐ | ☐ | Does the activity involve a *method of evaluation* described in the ISFSI licensing basis? |
| ☐ | ☐ | Is the activity a *test or experiment?* (i e , a non-passive activity which gathers data) |
| ☐ | ☐ | Does the activity exceed or potentially affect a cask *design basis limit for a fission product barrier (DBLFPB)?* (NOTE: If THIS questions is answered YES, a 10 CFR 72.48 Evaluation is required.) |

If the answers to ALL of these questions are NO, mark Parts VII as not applicable, and document the 10 CFR 72.48 screening in the conclusion section (Part VIII).

-ny of the above questions are marked YES, identify below the specific design function(s), method of evaluation(s) or DBLFPB(s) ved.

PART VII (72.48) - DETERMINE WHETHER THE ACTIVITY INVOLVES ADVERSE EFFECTS (NEI 96-07, Appendix B, Section B.4.2.1)

(If ALL the questions in Part V or Part VI are answered NO, then Part VII is ☒ NOT APPLICABLE.)

Answer the following questions to determine if the activity has an *adverse effect* on a design function. Any YES answer means that a 10 CFR 72.48 Evaluation is required; EXCEPT where noted in Part VII.3.

VII.1 Changes to the Facility or Procedures

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity adversely affect the *design function* of a plant, cask, or ISFSI SSC credited in safety analyses? |
| ☐ | ☐ | Does the activity adversely affect the method of performing or controlling the *design function* of a plant, cask, or ISFSI SSC credited in the safety analyses? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary)

VII.2 Changes to a Method of Evaluation

(If the activity does not involve a method of evaluation, these questions are ☐ NOT APPLICABLE.)

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for performing safety analyses than that described in a cask SAR? |
| ☐ | ☐ | Does the activity use a revised or different method of evaluation for evaluating SSCs credited in safety analyses than that described in a cask SAR? |

If any answer is YES, a 10 CFR 72.48 Evaluation is required. If both answers are NO, describe the basis for the conclusion (attach additional discussion, as necessary)

VII.3 Tests or Experiments

(If the activity is not a test or experiment, the questions in VII.3.a and VII.3.b are ☐ NOT APPLICABLE.)

a. Answer these two questions first:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Is the proposed test or experiment bounded by other tests or experiments that are described in the cask ISFSI licensing basis? |
| ☐ | ☐ | Are the SSCs affected by the proposed test or experiment isolated from the cask(s) or ISFSI facility? |

If the answer to both questions is NO, continue to VII.3.b. If the answer to EITHER question is YES, then briefly describe the basis

b. Answer these additional questions ONLY for tests or experiments which do not meet the criteria given in VII.3 a above. If the answer to either question in VII.3 a is YES, then these three questions are ☐ NOT APPLICABLE:

| YES | NO | QUESTION |
|---|---|---|
| ☐ | ☐ | Does the activity utilize or control an SSC in a manner that is outside the reference bounds of the design bases as described in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity utilize or control a plant, cask or ISFSI facility SSC in a manner that is inconsistent with the analyses or descriptions in the ISFSI licensing basis? |
| ☐ | ☐ | Does the activity place the cask or ISFSI facility in a condition not previously evaluated or that could affect the capability of a plant, cask, or ISFSI SSC to perform its intended functions? |

If any answer in VII.3 b is YES, a 10 CFR 72.48 Evaluation is required. If the answers are all NO, describe the basis for the conclusion (attach additional discussion as necessary):

PT VIII - DOCUMENT THE CONCLUSION OF THE 10 CFR 72.48 SCREENING

Check all that apply:

A 10 CFR 72.48 Evaluation is ☐ required or ☒ NOT required. Obtain a screening number and provide the original to Records Management regardless of the conclusion of the 50.59 or 72.48 screening

A VSC-24 cask Safety Analysis Report change is ☐ required or ☒ NOT required. If a VSC-24 cask SAR change is required, then contact the Point Beach Dry Fuel Storage group supervisor.

A Regulatory Commitment (CLB Commitment Database) change is ☐ required or ☒ NOT required. If a Regulatory Commitment Change is required, initiate a commitment change per NP 5.1.7.

A change to the VSC-24 10 CFR 72.212 Site Evaluation Report is ☐ required or ☒ NOT required. If a VSC-24 10 CFR 72.212 Site Evaluation Report change is required, then contact the Point Beach Dry Fuel Storage group supervisor.