



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 10 1990

MEMORANDUM FOR: James L. Blaha, Assistant
for Operations
Office of the Executive Director for Operations

FROM: Edward L. Jordan, Director
Office for Analysis and Evaluation of
Operational Data

SUBJECT: TRACKING AND CLOSE OUT OF ISSUES RAISED AS A RESULT OF
THE SAN ONOFRE, RANCHO SECO, AND DAVIS-BESSE INCIDENT
INVESTIGATION TEAM (IIT) INVESTIGATIONS

References: A. OIG 88A-11, "OIG Review of NRC's Incident Investigation
Program," dated August 1989

B. Memorandum from Edward L. Jordan to Patricia G. Norry,
"NRC Incident Investigation Program Manual Chapter
0513," dated November 30, 1989

This is in response to your request for information and documentation in response to congressional staff inquiries on the above subject.

As you know, in accordance with NRC Manual Chapter 0513, NRC Incident Investigation Program, dated August 5, 1987, upon receipt of the IIT report, the Executive Director for Operations (EDO) shall identify and assign NRC office responsibility for generic and plant-specific actions resulting from the investigation that are safety significant and warrant additional attention or action. Office Directors designated by the EDO as having responsibility for the resolution of issues or concerns are responsible for providing written status reports on the disposition of assigned actions. In addition, followup actions associated with the IIT report do not necessarily include all licensee actions, nor do they cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or possible enforcement actions. These items are expected to be defined and implemented through the normal organizational structure and procedures.

Enclosure 1 provides a written disposition and/or status, along with appropriate references, for each of the NRC staff action items that were assigned by the EDO (Enclosure 2) to the various NRC offices associated with the San Onofre, Rancho Seco, and Davis Besse IIT reports, respectively. Per our discussion, details of resolution of plant specific items were not included for Rancho Seco because of the shutdown status and the utility decision to decommission. AEOD plans to document the status of the action items from Enclosure 1 in the AEOD Annual Report for CY 1989. Based on this review, it is concluded that all actions assigned by the EDO in response to the Davis Besse, Rancho Seco, and San Onofre IITs have been adequately addressed.

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211

James L. Blaha

- 2 -

Gathering this information was complicated by the extensive NRC reorganization involving personnel reassignments and elimination of many files and records, which became evident during the Office of the Inspector General (OIG) audit of the Incident Investigation Program conducted during January 1988 through August 1988 (reference A). The EDO decided to make specific changes to NRC Manual Chapter 0513 in response to the audit to improve the documentation, tracking, and closeout requirements of followup actions resulting from IITs (reference B). These changes are effective with the next IIT report, and the revised NRC Manual Chapter 0513 is currently being reviewed by the appropriate NRC offices for final concurrence.

If you have any questions, or if I can be of further assistance, please feel free to contact me at 492-4848, or Robert Freeman of my staff at 492-7613. In addition, for your convenience, a listing of NRC staff personnel who were contacted to provide information and for reviewing Enclosure 1 for accuracy and completeness is provided in Enclosure 3. Their input and support in this effort were greatly appreciated.

Original Signed by:
E. L. Jordan

Edward L. Jordan, Director
Office for Analysis and Evaluation of
Operational Data

Enclosures:

1. Disposition of NRC Staff Followup Actions
2. EDO Staff Action Memoranda
3. NRC Staff Listing

cc w/encls:

W. Kennedy, OEDO
M. Callahan, OCA

Distribution: See page 3

*See previous concurrence

:DEIIB	:DEIIB	:DEIIB	:D:DOA:AEOD	:DD:AEOD	:D:AEOD	:
:RFreeman:vb	:RLloyd	:DSRubin	:GZech	:DFRoss	:ELJordan	:
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ENCLOSURE 1

*STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE
NOVEMBER 21, 1985 SAN ONOFRE NUCLEAR GENERATING STATION
(SONGS) UNIT 1 EVENT

(Reference: NUREG-1190)

Adequacy of feedwater check valves to perform safety function.
References: Commission Briefing, Sections 6.2.4, 6.4, 6.7, and Principal
g)

ent and coordinate the staff and industry actions necessary to
the reliability of safety-related check valves. (Responsible
Office of Inspection and Enforcement (IE), other offices to
as requested).

to be evaluated include:

licensee's engineering report on root cause analysis and proposed
corrective actions

adequacy of check valve design for this application

adequacy of Inservice Testing (IST) Program and procedures to detect
degraded and failed valves

adequacy of check valves (and related testing programs) in other
systems such as residual heat removal (RHR) system

Conclusion

Enclosure 1 documents the IE review of plant-specific aspects of the event
response to item 1 and concludes that plant-specific check valve
problems have been adequately resolved. Generic adequacy of check valve

Following the loss of power and water hammer event which occurred
on November 21, 1985, the NRC developed an action plan for the review
of licensee actions required as a result of the event and the
subsequent IIT report. This plan involved actions by Region V (RV)
Office of Nuclear Reactor Regulation (NRR) and IE. The NRC concerns
and licensee commitments were predominately addressed prior to the
startup of Unit 1 (July 15, 1986). This effort was documented in
Inspection Report No. 50-206/86-34 (reference 4).

designs and IST programs was coordinated with utility owners' groups and the Institute of Nuclear Power Operations (INPO), and the results were fed back to the industry; this is documented in references 2 and 3. Concurrent with this industry activity, the NRC staff conducted in-depth plant inspections to assess check valves in accordance with NRC Inspection Manual Inspection Procedure 73756, "Inservice Testing of Pumps and Valves," dated March 16, 1987. These inspections found that licensee programs to address check valves ranged from little attention to reasonably good. In April 1989, the Nuclear Industry Check Valve Group was formed. This group plans to develop specific guidelines for plant actions pertaining to database information on check valve performance, non-intrusive examination procedures to assess check valve performance, and maintenance procedures. In addition, the NRC recently issued a generic letter (reference 5), which covers certain aspects of IST requirements for check valves.

2. Item: Completeness of resolved Unresolved Safety Issue (USI) A-1, "Water Hammer." (References: Finding Numbers 1, 2, 3, 8, and 9)

Action

Assess the need to reevaluate USI A-1 to specifically address the potential for and prevention of condensation-induced water hammers in feedwater piping (assume the issue concerning check valve integrity will be resolved in item 1). (Responsible Office: NRR)

Disposition

Reference 6 documents the results of NRR reassessment of USI A-1 in response to item 2 and concludes that the event at San Onofre was due to grossly failed check valves in the feedwater system and a basis did not exist for reopening the water hammer safety issue. Additionally, resolution of USI A-1 recognized that water hammer events would continue to occur and that such events would not produce unacceptably high contributions to core-melt frequency or public risk. The imposition of new requirements to reduce water hammer events is not supportable by current cost-benefit guidelines and, thus, no further action is warranted at this time.

3. Item: Adequacy of San Onofre Unit 1 design.
(Commission Briefing, Finding Numbers 11 and 13)

Action

Implement and coordinate the staff's actions to reevaluate the following San Onofre design features: (Responsible Office: NRR)

- . Manual loading of the diesel generators following a loss of power event
- . Manual actuation of steam-line isolation valves and assurance of steam generator availability to remove decay heat

- . Lack of steam generator blowdown status in control room
- . Adequacy of the licensee's design change to eliminate spurious safety injection (SI) indication on loss of power

Disposition

Reference 7 documents the resolution of design issues performed by NRR in response to item 3. Reference 8 documents the resolution of the last outstanding item identified in reference 7 and, thus, all plant-specific design issues have been resolved.

4. Item: Adequacy of post-trip review.
(References: Sections 6.6 and 7.2.2.4 and Finding Number 17)

Action

- a. Evaluate NRC requirements for ensuring that sufficient event data are retrievable to accurately reconstruct the event following a loss of offsite power. (Responsible Office: NRR)
- b. Evaluate the licensee's process for post-trip review and evaluation, including the thoroughness of review and oversight provided by the onsite and offsite nuclear safety review groups. (Responsible Office: RV)

Disposition

Reference 9 documents the NRR evaluation in response to item 4a and concludes that this issue is adequately addressed as part of item 1.2 of NRC Generic Letter 83-28.

Reference 10 documents the RV assessment of the adequacy of the licensee's safety review process in response to item 4b and concludes that the licensee's process for post-trip review and evaluation meets regulatory requirements.

5. Item: Adequacy of licensee's recordkeeping practices. (References: Section 6.5 and Finding Number 20)

Action

Evaluate the adequacy of the licensee's maintenance records.
(Responsible Office: RV)

Disposition

Reference 11 documents the RV assessment of the adequacy of the licensee's maintenance records in response to item 5 and concludes that the licensee's recordkeeping practices were satisfactory.

Item: Adequacy of operator training and/or procedures.
(References: Section 7 and Finding Numbers 14, 15, and 16)

Action

Review the implementation of the training program regarding operator understanding and actions in the area of electrical systems, and invoking technical specification action statements. (Responsible Office: RV)

Disposition

Reference 10 documents the RV assessment of the licensee's training program in response to item 6 and concludes that operator training and/or procedures were adequate.

Item: Adequacy of Emergency Notifications and NRC response.
(References: Section 7.3 and Finding Number 22)

Action

- a. Verify the adequacy of the licensee's procedures and training for reporting of events to NRC Operations Center. (Responsible Office: RV)
- b. Evaluate the need for changes in NRC policy or guidance regarding: the use of the emergency notification system (ENS) line; the use of NRC personnel as ENS communicators; and possible approaches to improve the ability to determine the overall plant status. (Responsible Office: IE)

Disposition

Reference 12 documents the RV assessment of the adequacy of the licensee's reporting to the NRC in response to item 7a and concludes that it was adequate.

References 13 and 14 documents the IE evaluation in response to item 7b, and concludes that the NRC policy on the use of the ENS was clear. To ensure that the NRC policy regarding the use of NRC personnel as ENS communicators would be followed, action was taken, as documented in reference 14, to communicate NRC's policy and guidance regarding the resident inspector's role during licensee events to all regions. Actions that were taken to improve the ability to determine the overall plant status included: development of the Emergency Response Data System (ERDS), increased emphasis on site specific training for NRC Headquarters Operations Officers (HOOs), and development of various site specific information systems for use by the HOOs. These actions, as documented in references 14 and 15, are in various stages of implementation and/or are being upgraded, and with the exception of ERDS, are expected to be fully implemented within a year. The ERDS is expected to be fully operational in about 3 years.

Item: Significance of backlog of license amendments.
(Reference: Commission Briefing)

Action

Evaluate whether a backlog of license amendments and technical specification changes contributed to delays in approving the licensee's IST program. (Responsible Office: NRR)

Disposition

Reference 16 documents the NRR evaluation of the backlog of licensee amendments in response to item 8 and concludes that the backlog of license amendments did not delay approval of the licensee's IST program.

STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE
DECEMBER 26, 1985 RANCHO SECO INCIDENT

(Reference: NUREG-1195)

Issue: Adequacy of the Auxiliary Feedwater (AFW) System.

Action

- a. Verify the acceptability of the existing initiation and control of the AFW system. (Section 7.2) (Responsible Office: NRR)
- b. Determine the status of any licensee commitment to install the emergency feedwater initiation and control (EFIC) system, and determine the acceptability of the current schedule for installation. (Principal Finding No. 11) (Responsible Office: NRR)

Disposition

For disposition of plant-specific issue 1a, see footnote below.

Reference 17 documents the status of licensee commitments deemed acceptable by the staff in response to issue 1b, which stated that all other Babcock & Wilcox (B&W) designed plants have completed the safety-grade upgrade of the AFW system, except Three Mile Island (TMI) Unit 1, which was completed during the November 1986 refueling outage.

Issue: Completeness of various staff and licensee actions associated with control systems.

Action

- a. In light of the ongoing B&W generic review, assess the need to reevaluate the actions taken by the staff and by the licensees in response to the findings, conclusions, and recommendations associated with BAW-1564; Bulletin 79-27; NUREG-0667; the February 1980 loss of non-nuclear instrumentation (NNI) power at Crystal River; the March 19, 1984 partial loss of NNI at Rancho Seco; and BAW-1791. (Principal Finding No. 15 and other Finding No. 11) (Responsible Office: NRR)

ote: Only the disposition, and/or status of generic actions for Rancho Seco are addressed due to the change in operational status of the facility. However, it should be noted that all plant-specific issues that require resolution for plant restart were resolved, as documented in references 18 and 19.

- b. Assess the need to expand the scope of USI A-47 to include additional consideration of frequent events with undesirable consequences even if the consequences of a particular event are bounded by the Final Safety Analysis Report (FSAR) analysis, and the degree to which events that are not significant at the referenced plants might be significant at other plants. (Principal Finding No. 15h)
(Responsible Office: NRR)
- c. Consider the need for issuing further generic communications.
(Responsible Office: IE)

Disposition

Reference 20 documents NRR assessment of prior actions taken by the licensees and NRC staff and details the NRC staff's plans and ongoing activities by the B&W Owner's Group (B&WOG) in response to issue 2a. This issue, along with others, was examined by the B&WOG Plant Reassessment Program, which was initiated in response to a number of significant events at B&W-designed reactors that led the NRC staff to conclude that the basic design requirements for B&W reactors needed to be reexamined. The B&WOG assumed leadership role in this effort. The 208 specific tasks that resulted from this effort were evaluated by the NRC staff. References 21 and 22 document the results of the NRC staff's evaluation. In addition, the staff is currently auditing the implementation of these tasks at each B&W designed facility.

Reference 17 documents the NRR assessment of the need to expand the scope of USI A-47 in response to issue 2b and concludes that further expansion does not appear appropriate.

Reference 23 documents the IE assessment of the need for further generic communications in response to issue 2c and concludes that further generic communications do not appear warranted since the issues are specific to B&W designed plants and are already being addressed by the B&WOG and the NRC staff.

- 3. Issue: Adequacy of the design of the Integrated Control System (ICS).

Action

Assess the adequacy of the design of the ICS. (Responsible Office: NRR)

Particular features to be included are:

- a. Whether the ICS is sufficiently reliable to assure that the frequency of unnecessary safety challenges is acceptably low (i.e., is the ICS properly classified as a nonsafety-related system).
- b. Loss of remote (i.e., hand) power coincident with loss of automatic control. (Principal Finding No. 2)

Results of Sacramento Municipal Utility District (SMUD's) analysis to date of the power supply monitor. (Principal Finding No. 1)

Results of SMUD's contractor analysis of the power supply monitor. (Principal Finding No. 1)

Role of the power supply monitor as a potential single failure in the ICS and/or NNI. (Principal Finding No. 1)

Results of a study of the response of the ICS upon restoration of power. (Other Finding No. 3)

Acceptability of the failure mode of meters and recorders that are affected by a loss of ICS power. (Other Finding No. 4)

Conclusion

Evaluation of the adequacy of the ICS in response to issues 3a through 3d addressed by the B&WOG Plant Reassessment, and the improvements implemented by the B&WOG were evaluated by the staff in section 6.1 of Appendix 21. The staff concluded in its safety analysis report that with implementation of the B&WOG recommendations, the performance of the ICS will be significantly improved.

Adequacy of the maintenance program for manual isolation valves.

Evaluate the need for a maintenance program for manual isolation valves in safety-related systems. (Principal Finding No.3)
Responsible Office: RV)

Evaluate the adequacy of industry standards and NRC requirements regarding periodic testing and maintenance of manual valves in safety-related systems. (Principal Finding No. 3) (Responsible Office: NRR)

Conclusion

Disposition of plant-specific issue 4a, see footnote on page 6.

Appendix 17 documents the NRR evaluation of the adequacy of industry standards and NRC requirements in response to issue 4b and concludes that the industry does not have a formal maintenance program that applies to manual valves, nor are there NRC requirements for maintenance and testing of "convenience" valves. Actions taken to address this issue included issuance of an NRC information notice (reference 24) and implementation of a generic issue (Generic Issue 127, "Maintenance and Testing of Manual Valves in Safety-Related Systems"). Evaluation of this issue indicates that there is little or no prospect of safety

improvements that are both substantial and worthwhile and that it is not clear from currently available information whether the generic issue merits further pursuit.

5. Issue: Adequacy of procedures and training.

Action

Evaluate the adequacy of the procedures and operator (licensed and nonlicensed) training, particularly with regard to:

- a. The degree to which event specific procedures (e.g., loss of ICS, station blackout) are needed to quickly recover from events that have been diagnosed by the operators, and to mitigate such events if the initiating condition cannot be immediately corrected. (Principal Findings Nos. 4 and 7) (Responsible Office: NRR)
- b. The consistency between EOPs and NRC approved procedure generation packages. (Responsible Office: RV)
- c. The consistency between emergency operating procedures (EOPs) and NRC approved procedure generation packages. (Responsible Office: IE)
- d. The adequacy of procedural guidance concerning: (1) when to trip auxiliary feedwater pumps, and (2) the relative priorities of avoiding the pressurized thermal shock (PTS) region and maintaining pressurizer level. (Principal Finding Nos. 5 and 6) (Responsible Office: NRR & RV)
- e. The adequacy of the training of nonlicensed operators in the use of valve wrenches, the methods for manually overriding and operating valves, and the use of various indications of valve position. (Principal Finding #9) (Responsible Office: RV)
- f. The adequacy of the annunciator response procedures (e.g., the Annunciator Procedures Manual). (Other Finding #2) (Responsible Office: RV)
- g. The adequacy of the program to insure that all applicable procedures and operator training are reviewed and updated when plant modifications are made. (Other Finding #6) (Responsible Office: RV)

Disposition

For disposition of plant-specific issues 5b, 5d, and 5e through 5g, see footnote on page 6.

Reference 17 documents the NRR evaluation of event specific procedures and concludes that there appears to be no basis for the generic concern regarding the need for event-specific procedures expressed under issue 5a. This issue, however, was included in the B&WOG Plant Reassessment Program effort.

Reference 17 documents the actions taken by IE and NRR in response to issue 5c, which included: the issuance of an NRC information notice (reference 25); implementation of an EOP inspection program by NRR; and the issuance of a temporary instruction to the regions to inspect EOPs at operating plants (issued in June 1986).

Reference 17 documents that the generic aspects of issue 5d was addressed by the B&WOG Plant Reassessment Program as part of their procedural review effort.

6. Issue: Adequacy of the radiological control and emergency preparedness program.

Action

Evaluate the adequacy of the radiological control and emergency preparedness program, including procedures, operator training, equipment availability, and coordination between operating and health physics personnel. (Principal Finding #10 and Other Finding #5) (Responsible Office: RV)

Disposition

For disposition of plant-specific issue 6, see footnote on page 6.

7. Issue: Adequacy of the FSAR accident analysis.

Action

Verify the adequacy of the Rancho Seco FSAR accident analysis, particularly the degree to which credit is given for the nonsafety-related ICS and the nonsafety-related Main Steamline Failure Logic. (Section 9) and (Principal Finding No. 14) (Responsible Office: NRR)

Disposition

Reference 17 documents that issue 7 was handled as a plant-specific issue and was subsequently addressed as part of licensee restart evaluation documented in reference 18. The scope of the review performed focused primarily on the effects of the ICS on the FSAR accident analysis. In addition, reference 19 documents the resolution of all of the remaining plant-specific issues that had not been resolved in reference 18. Thus, all plant-specific issues that required resolution for restart of Rancho Seco were resolved.

8. Issue: Adequacy of required staffing.

Action

Evaluate the adequacy of plant staffing to deal with expected operational transients. (Other Finding No. 7) (Responsible Office: NRR)

Disposition

Reference 17 documents that issue 8 was addressed by the B&WOG Plant Reassessment Program as part of the procedural review effort.

Issue: Adequacy of the troubleshooting program.

Action

Evaluate the adequacy of the licensee's program to troubleshoot damaged equipment in a controlled and systematic manner to determine the root cause and appropriate corrective actions. (Other Finding #9)
(Responsible Office: RV)

Disposition

For disposition of plant-specific issue 9, see footnote on page 6.

STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE
JUNE 9, 1985 DAVIS-BESSE EVENT

(Reference: NUREG-1154)

1. Item: Adequacy of the licensee's management and maintenance practices.
(Reference: Conclusion Section 8)

Action

- (a) Evaluate and take action on the licensee's response to findings relating to corrective actions and preventive maintenance problems (including testing, root cause determination of equipment misoperation and operating experience). (Responsible Office: NRR)
- (b) Evaluate and take action on the licensee's response to findings concerning management practices (e.g., control of maintenance programs and post-trip reviews). (Responsible Office: Region III (RIII))

Disposition

Reference 26 documents the NRC staff's evaluation of the licensee's response to all plant-specific findings regarding the event. In a number of cases, the staff relied upon commitments made by the licensee to complete certain actions to adequately address certain issues (reference 27). All licensee actions required prior to restart were completed prior to restart. All other plant-specific actions required to resolve the IIT findings have been completed except for the human engineering enhancements in the control room that are being completed during the January 1990 through May 1990 refueling outage.

2. Item: Completion of analyses for loss of feedwater events. (Reference: Section 7)

Action

Evaluate the time margins and consequences of alternative sequences for a loss of feedwater event at Davis-Besse. (Responsible Office: NRR)

Disposition

For plant-specific actions in response to item 2, see the disposition of item 1.

3. Item: Adequacy of the Steam Feedwater Rupture Control System (SFRCS).
(References: Section 5.2.2 and Finding 6)

Action

Review the design basis for SFRCS and the susceptibility of the SFRCS to:
(a) spurious actuations involving such items as main steam isolation valve (MSIV) closure; and (b) single failures. (Responsible Office: NRR)

Disposition

For plant-specific actions in response to item 3, see the disposition of item 1.

Item: Interaction of plant security features and operator actions.
(References: Section 3.6 and Finding 9)

Action

Evaluate the effect of security features (locked doors, locked equipment, etc.) on the operator's ability to gain prompt access to equipment required to perform safety actions outside the control room in accordance with emergency procedures. (Responsible Office: NRR)

Disposition

For plant-specific actions in response to item 4, see the disposition of item 1.

Reference 28 documents the staff action taken regarding item 4 and a related operational experience evaluation performed by the Office for Analysis and Evaluation of Operational Data (AEOD) (reference 29), which subsequently resulted in the issuance of an NRC information notice (reference 30). In addition, reference 31 documents that item 4 was also evaluated under Generic Issue 122.3, "Physical Security System Constraints," which resulted in a low priority ranking.

Item: Availability of the Shift Technical Advisor (STA). (References: Section 6.1.3 and Finding 14)

Action

Evaluate the time available and role for STA assistance during a complex operating event. (Responsible Office: NRR)

Disposition

For plant-specific actions in response to item 5, see the disposition of item 1.

References 32 and 33 documents the NRR review of the IIT report, which resulted in the identification of a number of short-term and long-term generic issues for staff action. The status of these generic actions is documented in reference 31. The generic issue of the availability of the shift technical advisor (Generic Issue 125.I.1, "Availability of the

Shift Technical Advisor") was dropped based on the results of a survey of licensees, which indicated that STAs are in the control room or immediately available at the majority of operating plants. For the three plants identified with a deficiency, licensee corrective action was reviewed by the staff on a plant-specific basis.

Item: Reliability of the AFW containment isolation valves and other safety-related valves. (References: Section 5.2.5 and Finding 4, 5, 6, and 15)

Action

- (a) Monitor the licensee's troubleshooting activities. (Responsible Office: RIII)
- (b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions. (Responsible Office: NRR)
- (c) Determine if the safety function of the AFW containment isolation valves has been properly specified, i.e., are the valves required to open as well as close for design basis events. (Responsible Office: NRR)
- (d) Verify that these valves constitute a single failure point of the AFW system for certain design basis events. (Responsible Office: NRR)
- (e) Determine that the procedures for adjustments of the AFW isolation valves such as torque switch bypass switches are clear and proper, and that the associated training programs are adequate. Confirm that adjustment settings are consistent with plant procedures. (Responsible Office: RIII)
- (f) Determine if the engineering basis for the specification of the adjustments for safety-related valves such as the torque switch and torque switch bypass switch setting are adequate for all design basis events. (Responsible Office: NRR)
- (g) Evaluate the test program for the AFW containment isolation valves to confirm operability for all design basis events. (Responsible Office: NRR)
- (h) Evaluate whether other safety-related valves in Davis-Besse may be subject to the same type/cause of failure. (Responsible Office: NRR)
- (i) Conduct a review of failures of safety-related motor-operated valves and provide an assessment of pertinent failure modes affecting valve performance under design basis conditions. (Responsible Office: AEOD)
- (j) Determine if further generic correspondence, such as an NRC Bulletin, is warranted on this type/cause of failure of safety-related valves. (Responsible Office: IE)

Disposition

For plant-specific actions in response to items 6a through 6h, see the disposition of item 1.

Reference 34 documents the results of AEOD's review of safety-related motor-operated valve experience in response to item 6i.

Reference 27 documents the IE evaluation of item 6j. This evaluation determined that improper setting of torque and limit switches on motor-operated valves was a significant contributor to the event. As a result, the NRC issued a bulletin regarding this subject (reference 35.) The NRC subsequently issued a generic letter (reference 36) that expanded the scope of the bulletin based on a NRC staff assessment which indicated that the program to verify switch settings should be extended to ensure the operability of all safety-related fluid systems.

Item: Adequacy of emergency notifications. (References: Section 6.1.4 and Finding 12)

Action

- (a) Verify the adequacy of the licensee's procedures and training for reporting of events to the NRC Operations Center. (Responsible Office: RIII)
- (b) Review the adequacy of NRC guidance for determination of severity levels when plant conditions vary and may be stable when the licensee has an opportunity to report. (Responsible Office: IE)
- (c) Review the adequacy of shift staffing for assuring that knowledgeable individuals will be available for properly implementing the emergency plan during complex and long operational events. (Responsible Office: IE)

Disposition

For plant-specific actions in response to item 7a, see the disposition of item 1.

Reference 37 documents the IE review of item 7b. As a result of this review, an NRC information notice (reference 38) was issued to clarify the requirement for licensees to notify the NRC Operations Officer of emergencies.

Reference 37 documents the IE review of item 7c and concludes that it is not necessary or desirable to require additional personnel on shift to implement the emergency plan. NRC Information Notice 85-80 addresses the specific corrective actions undertaken at Davis-Besse and provides generic recommendations to licensees to minimize the potential emergency planning problem raised by the Davis-Besse event.

Item: Reliability of the AFW pump turbines. References: Section 5.2.4 and 6.2.4 and Findings 4, 8, and 15)

Action

- (a) Monitor the licensee's troubleshooting activities including possible hot plant operation to confirm failure mode. (Responsible Office: RIII)
- (b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions. (Responsible Office: NRR)
- (c) Evaluate the licensee's response and corrective actions relating to the unreliability of the auxiliary feedwater system (including the need for a third pump and turbine trip reset capability). (Responsible Office: NRR)
- (d) Verify that the AFW system has been adequately tested to confirm system configuration involved with design basis events. (Responsible Office: RIII)
- (e) Review the implementation of the operator training program to assure proper operator actions, such as resetting of trip throttle valve. (Responsible Office: RIII)
- (f) Conduct a review of past operating experience and determine the causes for overspeed turbine trips. (Responsible Office: AEOD)
- (g) Determine the need for further generic correspondence on this mode/cause. (Responsible Office: IE)

Disposition

For plant-specific actions in response to items 8a through 8e, see the disposition of item 1.

Reference 39 documents AEOD's review of operating experience in response to item 8f.

Reference 37 documents the IE assessment of the need for generic correspondence in response to item 8g. It was determined that an NRC information notice was warranted (reference 40). Following AEOD review of operating experience, a supplemental NRC information notice was issued (reference 41) to address and communicate the results of the AEOD evaluation.

Item: Reliability of the pilot-operated relief valve (PORV).
(References: Sections 5.2.8 and 6.2.1 and Findings 10 and 13)

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- a) Monitor the licensee's troubleshooting activities. (Responsible Office: RIII)
- b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions. (Responsible Office: NRR)
- c) Determine the need for a test program to establish reliability. (Responsible Office: NRR)
- d) Determine if surveillance tests are necessary to confirm operational readiness. (Responsible Office: NRR)
- e) Determine if additional protection against PORV failure is necessary, i.e., automatic block valve closure. (Responsible Office: NRR)

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or plant-specific actions in response to items 9a and 9b, see the disposition of item 1.

Reference 31 documents the status of two generic issues (Generic Issue 125.I.2.A, "Need for a Test Program to Establish Reliability of the PORV in response to item 9c, and Generic Issue 125.I.2.B, "Need for PORV Surveillance Tests to Confirm Operational Readiness in response to item 9d). Both issues are being addressed in the resolution of Generic Issue 70, "PORV and Block Valve Reliability." Generic Issue 70 is currently considered resolved based on the development of an NRC generic letter, which was presented and subsequently approved during Committee for the Review of Generic Requirements (CRGR) Meetings 167 and 168. Issuance of the NRC generic letter is expected in early 1990.

Reference 31 documents another generic issue (Generic Issue 125.I.2.C, "Need for Additional Protection Against PORV Failure" developed in response to item 9e) was dropped based on the results of safety analysis reports performed on the three vendor group responses (Combustion Engineering (CE), B&W, Westinghouse (W)) to TMI Action Plan Item II.K.3(2), "Report in Overall Safety Effect of PORV Isolation System." The safety analysis reports concludes that an automatic PORV isolation system was not necessary and in light of the control room indications available to the operators, the safety concerns of issue 9e have been adequately resolved.

Item: Adequacy of control room instrumentation and controls. (References: Sections 6.1.1, 6.1.2, and 6.2.2 and Findings 10, 11, 17, and 18)

Action

- (a) Evaluate the adequacy of the SFRCS actuation controls and associated training program. (Responsible Office: NRR)
- (b) Evaluate the adequacy of the installed control room instrumentation to allow operators to make the necessary and prompt determination for procedure conformance and PORV position. (Responsible Office: NRR)
- (c) Determine if NRC requirements should be revised regarding:
(1) safety parameter display system (SPDS) availability; and (2) the need for plant-specific simulator. (Responsible Office: NRR)

Disposition

For plant-specific actions in response to items 10a and 10b, see the disposition of item 1.

Reference 31 documents the status of the generic issue developed in response to item 10c (Generic Issue 125.I.3, "SPDS Availability"). The staff evaluated licensee/applicant implementation of the SPDS requirements at 57 units and found that a large percentage of designs did not satisfy requirements identified in Supplement 1 to NUREG-0737. As a result, NRC Generic Letter 89-06 was issued to inform licensees of the staff's findings to aid in implementing SPDS requirements. With the issuance of NRC Generic Letter 89-06, the generic issue of SPDS availability was resolved and requirements were established. Reference 31 documents another generic issue, (Generic Issue 125.I.4, "Plant-Specific Simulator") developed in response to item 10c, which was addressed in the rulemaking amendments to 10 CFR 55 under TMI Action Plan Item I.A.4.2(4).

Item: Need for isolation of the startup feedwater pump. (References: Section 5.1.3 and Finding 7)

Action

Reassess acceptability of the provisions which resulted in the inability to place the startup feedwater pump in service from the control room. (Responsible Office: NRR)

Disposition

For plant-specific actions in response to item 11, see the disposition of item 1.

Item: Resolution of equipment deficiencies. (References: Section 5 and Table 5.1)

ion

Monitor the licensee's troubleshooting activities. (Responsible Office: RIII)

Evaluate the licensee's report on the root cause analysis and corrective action for the equipment listed on Table 5.1 and not addressed by other items in this action plan. (Responsible Office: NRR)

Determine the need for generic correspondence on equipment problems. (Responsible Office: IE)

position

plant-specific actions in response to items 12a and 12b, see the position of item 1.

erence 37 documents the IE evaluation of the need for generic correspondence on equipment problems in response to item 12c, which determined that no further generic correspondence was warranted.

m: Adequacy of plant procedures. (References: Sections 6.1.1 and 6.2 and Findings 10 and 17)

ion

ify that plant procedures involving "drastic" actions are required be sufficiently precise and clear to ensure prompt implementation. (Responsible Office: NRR)

position

erence 31 documents the status of the staff's evaluation of this generic issue (Generic Issue 122.2, "Initiating Feed-and-Bleed") in response to item 13 and concludes that there was no need for new regulatory requirements/guidance. This conclusion was based on the determination that there is adequate reactor safety and ongoing industry initiatives to continue enhancing safety involving feed-and-bleed. More specifically, the staff's conclusion was based on the following: (1) as a result of the accident, NRC required licensees to have new EOPs to prevent/mitigate accidents; (2) licensees currently have EOPs in place that incorporate clear steam supply system (NSSS) vendor guidance for feed-and-bleed; (3) licensees are continuing to enhance feed-and-bleed procedures taking into account current NSSS vendor recommendations; and (4) NRC has ongoing ongoing review/inspection activities concerning NSSS vendor/licensee enhancement of EOPs including feed-and-bleed. Thus, this issue was considered resolved and no new requirements were established.

14. Item: Adequacy of safety system testing. (Reference: Finding 15)

Action

Evaluate the NRC requirements and guidance to assure that safety systems are tested in all configurations required by the design basis analysis. (Responsible Office: NRR)

Disposition

Reference 31 documents the status of the staff's evaluation of a generic issue (Generic Issue 125.I.5, "Safety Systems Tested in All Conditions Required by Design Basis Accident (DBA)") in response to item 14, and concludes that using integral plant/system testing, as near as practical to design basis accident conditions to detect common mode design deficiencies seems too limited in its goal, considering the potentially large expenditure of time and resources that may be needed to develop the program. The estimated success probability of the tests to detect all unforeseen common mode design deficiencies results in a potential reduction in core-melt frequency of 3.4×10^{-7} /reactor year, which borders between the generic issue ratings of drop and low priority. Based on the results of the staff's evaluation and other considerations, this generic issue was dropped.

15. Item: Acceptability of current safety assessment methods. (References: Findings 1 and 2)

Action

Assess the implications of multiple independent and common mode failures as they relate to departures from design assumptions and specifications used in probabilistic safety analyses. (Responsible Office: Office of Nuclear Regulatory Research (RES))

Disposition

Reference 42 documents the RES actions to be taken in response to item 15. References 43 through 48 describe the research results. This research developed reliability technology applicable to operational safety of nuclear power plants, and the results are being applied to improve the regulatory program.

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2. Institute of Nuclear Power Operations (INPO) Significant Operations Experience Report (SOER) 86-03, "Check Valve Failures or Degradations" dated Oct 1986.
3. Electric Power Research Institute (EPRI) Report NP-5479 "Check Valve Application Guidelines," dated Jan 1988.
4. NRC Inspection Report No. 50-206/86-34, dated Jul 28, 1986.
5. NRC Generic Letter No. 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," dated Apr 3, 1989.
6. Memorandum from H. Denton for V. Stello, Jr., dated Jul 7, 1986, "Review and Assessment of Waterhammer Occurrences Since CY 1981."
7. Letter from T. M. Novak to K. P. Baskin (SCE) dated Nov 12, 1986, "Technical Evaluation of Issues from the NRC Staff Action Plan in Response to the November 21, 1985 Event at San Onofre Unit 1."
8. Letter from C. M. Trammell to H. B. Ray (SCE) dated Dec 8, 1989, "Long-Term Electrical Cable Monitoring Program San Onofre Nuclear Generating Station, Unit 1 (TAC 73782)."
9. Memorandum from H. Denton for V. Stello, Jr., dated Aug 4, 1986, "Post-Trip Data Following Loss of Offsite Power."
10. NRC Inspection Report No. 50-206/86-20, dated Jun 9, 1986.
11. NRC Inspection Report No. 50-206/86-22, dated Jun 9, 1986.
12. NRC Inspection Report No. 50-206/86-16, dated Jun 2, 1986.
13. Memorandum from K. Perkins for G. Zech dated Feb 10, 1986, "Response to 01/04/86 Memo from Stello to Denton on the 11/21/86 San Onofre Investigation."
14. Memorandum from G. Grimes for D. Kirsch dated Jun 11, 1986, "IRB Response to San Onofre 1 IIT Action Item List."
15. Memorandum from K. Perkins for G. Zech, dated Feb 13, 1986, "IRB Response to San Onofre 1 IIT ACTION ITEM LIST."

Memorandum from H. Denton for V. Stello, Jr., dated Apr 28, 1986, "San Onofre Nuclear Generating Station, Unit 1 (SONGS-1) Inservice Testing (IST) Review."

Memorandum from H. Denton to V. Stello, Jr., dated Oct 7, 1987, "Staff Actions Resulting from the Investigation of the December 26, 1985 Incident at Rancho Seco (NUREG-1195)."

NUREG-1286, "Safety Evaluation Report Related to the Restart of Rancho Seco Nuclear Generating Station Unit 1, Following the Event of December 26, 1985," dated Oct 1987.

NUREG-1286, Supplement 1, "Safety Evaluation Report Related to the Restart of Rancho Seco Nuclear Generating Station, Unit 1, Following the Event of December 26, 1985," dated Mar 1988.

Memorandum from H. Denton to V. Stello, Jr., dated Jul 23, 1986, "Completeness of Various Staff And Licensee Actions Associated With Control Systems."

NUREG-1231, "Safety Evaluation Report Related to Babcock & Wilcox Owner's Group Plant Reassessment Program," dated Nov 1987.

NUREG-1231, Supplement 1, "Safety Evaluation Report Related to Babcock & Wilcox Owner's Group Plant Reassessment Program," dated Mar 1988.

Memorandum from J. Taylor for V. Stello, Jr., dated Sep 24, 1986, "Staff Actions Resulting from the Investigation of the December 26, 1985 Incident at Rancho Seco (NUREG-1195)."

IE Information Notice 86-61 dated Jul 28, 1986, "Failure of Auxiliary Feedwater Manual Isolation Valve."

IE Information Notice 86-64 dated Aug 14, 1986, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures."

NUREG-1177, "Safety Evaluation Report related to the restart of Davis-Besse Nuclear Power Station, Unit 1, following the event of Jun 9, 1985," dated Jun 1986.

Letter from J. Stolz (NRC) to J. Williams, Jr. (TE), "Safety Evaluation Report Supporting Davis-Besse Restart NUREG-1177; License Action Commitments," dated Jun 10, 1986.

Memorandum from R. Vollmer to F. Hebdon, "Delayed Access to Safety-Related Areas During Plant Operation," dated Mar 19, 1986.

AEOD Engineering Evaluation E403, "Delayed Access to Safety-Related Areas During Plant Operation," dated Feb 19, 1986.

IE Information Notice 86-55, "Delayed Access to Safety-Related Areas and Equipment During Plant Emergencies," dated Jul 10, 1986.

EG-0933, Supplements 1 to 20, "A Prioritization of Generic Safety Issues," Nov 1983 to Jun 30, 1989.

Memorandum from D. Crutchfield to H. Thompson, "Potential Immediate Generic Actions as a Result of the Davis Besse Event of June 9, 1985," dated Aug 5, 1985.

Memorandum from H. Thompson to T. Speis, "Longer-Term Generic Actions as a Result of the Davis Besse Event of June 9, 1985," dated Nov 6, 1985.

DO Case Study C603, "A Review of Motor-Operated Valve Performance," dated Dec 1986.

Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," dated Nov 15, 1985.

Generic Letter No. 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated Jun 28, 1989.

Memorandum from J. Taylor to V. Stello, Jr., "Staff Actions Resulting from the Investigation of the June 9, 1985 Davis-Besse Event (REG-1154)," dated Mar 7, 1986.

Information Notice 85-80, "Timely Declaration of an Emergency Class, Implementation of an Emergency Plan, and Emergency Notification," dated May 15, 1985.

DO Case Study C602, "Operational Experience Involving Turbine Overspeed Events," dated Aug 1986.

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Memorandum from R. Minogue to W. Dircks, "RES Actions Resulting from Investigation of Davis Besse Event," dated Oct 4, 1985.

EG/CR-4618, "Evaluation of Reliability Technology Applicable to LWR Operational Safety," dated Aug 1988.

EG/CR-5425, "Evaluation of Allowed Outage Times (AOTs) From a Risk and Reliability Standpoint," dated Jul 1989.

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6. NUREG/CR-5021, "Users' Guide for PRISIM Arkansas Nuclear One Unit 1," Volumes 1 & 2. dated Mar 1988.
7. NUREG-1377, NRC Research Program on Plant Aging: Listing and Abstracts of Reports issued through February 1, 1989," dated Aug 1989.
8. "Engineering Simulator Applications to Emergency Response Training," R.J. Beelman et al., Proceedings of the SCS Multi Conference on Simulators VI, 28-31 Mar 1989, Tampa Florida (Available from SCS, P.O. Box 17900, San Diego, California.)

ACRONYMS

AEOD	OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA
AFW	AUXILIARY FEEDWATER
OT	ALLOWED OUTAGE TIME
B&W	BABCOCK AND WILCOX
B&WOG	BABCOCK AND WILCOX OWNERS GROUP
CE	COMBUSTION ENGINEERING
CFR	CODE OF FEDERAL REGULATIONS
CRGR	COMMITTEE FOR THE REVIEW OF GENERIC REQUIREMENTS
DBA	DESIGN BASIS ACCIDENT
EDO	EXECUTIVE DIRECTOR FOR OPERATIONS
EFIC	EMERGENCY FEEDWATER INITIATION AND CONTROL
ENS	EMERGENCY NOTIFICATION SYSTEM
EOP	EMERGENCY OPERATING PROCEDURES
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
ERDS	EMERGENCY RESPONSE DATA SYSTEM
FSAR	FINAL SAFETY ANALYSIS REPORT
HOO	HEADQUARTERS OPERATIONS OFFICE
HPCI	HIGH PRESSURE COOLANT INJECTION
ICS	INTEGRATED CONTROL SYSTEM
E	OFFICE OF INSPECTION AND ENFORCEMENT
IT	INCIDENT INVESTIGATION TEAM
NPO	INSTITUTE OF NUCLEAR POWER OPERATIONS
RB	INCIDENT RESPONSE BRANCH
ST	INSERVICE TESTING
LWR	LIGHT WATER REACTOR
MSIV	MAIN STEAM ISOLATION VALVE
MNI	NON-NUCLEAR INSTRUMENTATION
NRR	OFFICE OF NUCLEAR REACTOR REGULATION
NSSS	NUCLEAR STEAM SUPPLY SYSTEM
DIG	OFFICE OF INSPECTOR GENERAL
PORV	PILOT-OPERATED RELIEF VALVE
PTS	PRESSURIZED THERMAL SHOCK
PWR	PRESSURIZED WATER REACTOR
RCIC	REACTOR CORE ISOLATION COOLING
RES	OFFICE OF NUCLEAR REGULATORY RESEARCH
RHR	RESIDUAL HEAT REMOVAL
RIII	REGION 3
RV	REGION 5
SCE	SOUTHERN CALIFORNIA EDISON
SFRCS	STEAM FEEDWATER RUPTURE CONTROL SYSTEM
SI	SAFETY INJECTION
SMUD	SACRAMENTO MUNICIPAL UTILITY DISTRICT
SOER	SIGNIFICANT OPERATIONS EXPERIENCE REPORT
SONGS	SAN ONOFRE NUCLEAR GENERATING STATION
SPDS	SAFETY PARAMETER DISPLAY SYSTEM
STA	SHIFT TECHNICAL ADVISOR
TAC	TASK ACTION COMMITMENT
TE	TOLEDO EDISON
MI	THREE MILE ISLAND
USI	UNRESOLVED SAFETY ISSUE
W	WESTINGHOUSE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20540

FEB 04 1986

MEMORANDUM FOR:

Harold R. Denton, Director, NRR
James M. Taylor, Director, IE
John B. Martin, Regional Administrator, Region V

FROM:

Victor Stello, Jr.
Acting Executive Director
of Operations

SUBJECT:

STAFF ACTIONS RESULTING FROM THE INVESTIGATION
OF THE NOVEMBER 21 SAN ONOFRE NUCLEAR GENERATING
STATION, UNIT 1 EVENT (NUREG-1190)

REC'D

FEB 04 1986

RECEIVED
NRC

An advance copy of the subject report was transmitted to you by memorandum dated January 20, 1986 from the San Onofre Team Leader, Thomas T. Martin. The report documents the Team's efforts in identifying the circumstances and causes of the November 21, 1985 event, together with findings and conclusions which form the basis for identifying follow-on actions.

You will note from the report that the licensee has not completed troubleshooting and the determination of root causes for all equipment failures or malfunctions. Consequently, the results of future troubleshooting or analysis activities may form the basis for additional follow-on actions. The identification of these additional actions is a responsibility of the normal program office. The responsibility for the followup and reporting on the licensee's continued troubleshooting and determination of root cause for equipment failures is Region V.


The purpose of this memorandum is to identify and assign responsibility for generic and plant-specific actions resulting from the investigation of the San Onofre event (documented in NUREG-1190). In this regard, you are requested to review the enclosure which specifies staff actions resulting from the investigation of the San Onofre event. You are requested to determine the actions necessary to resolve each of the items in your area of responsibility and, where appropriate, identify additional staff actions or revisions as our review and understanding of this event are refined. Plant-specific actions required for plant restart should receive priority attention.

In view of the importance of this subject, I intend to closely monitor the resolution of these items. By March 1, 1986, please provide a written summary of the schedule and status of each item within your responsibility listed in the enclosure or that you have identified. Further, I request that you prepare written status report on the disposition of your items (and anticipated actions for uncompleted items) within three to six months. Every effort should be made to dispose of these items promptly.

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The enclosure is based on the Team's report and its presentation to the Commission on January 22, 1986. Accordingly, it does not include all licensee actions, nor does it cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or possible enforcement items. These items are expected to be defined and implemented in a routine manner. Overall lead responsibility for staff actions relating to facility restart is separate from this effort and rests with Region V. Additionally, RV is responsible for coordinating and promptly communicating the staff's requirements which must be resolved before operations at San Onofre may be resumed.


Victor Stello, Jr.
Acting Executive Director
for Operations

Enclosure:
As stated

cc w/enclosure:
J. Davis, WMSS
T. Murley, RI
J. N. Grace, RII
J. Yeopler, RIII
R. I. tin, RIV

ACTIONS RESULTING FROM THE INVESTIGATION

OF THE NOVEMBER 21 SONGS-1 EVENT

(Reference: NUREG-1190)

of feedwater check valves to perform safety function.
omission briefing, Sections 6.2.4, 6.4, 6.7, and Principal

	<u>Responsible Office</u>	<u>Category</u>
1. Coordinate industry actions to ensure the reliability of feedwater check valves. 2. To assist as needed to be done: a. Engineering root cause and proposed actions b. Check valve testing where this applica-	IE	Plant-specific Generic
3. Inservice (ISI) Program and to detect and failed valves 4. Check valves and testing in other systems R system 5. Success of resolved USI A-1, "Water Hammer". (Finding numbers 1, 2, 3, 8 and 9)		

	<u>Responsible Office</u>	<u>Category</u>
1. To re- 2. To address the and prevention -induced water water piping use concerning integrity will be in 1).	NRR	Generic

3. Item: Adequacy of San Onofre Unit 1 design.
(Commission briefing, Finding numbers 11 and 13)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Implement and coordinate the staff's actions to re-evaluate the following San Onofre design features:	NRR	Plant-specific
- manual loading of the diesel generators following a loss of power event		
- manual actuation of steam line isolation valves and assurance of steam generator availability to remove decay heat		
- lack of steam generator blowdown status in control room		
- adequacy of the licensee's design change to eliminate spurious SI indication on loss of power		

4. Item: Adequacy of post-trip review.
(References: Sections 6.6 and 7.2.2.4 and Finding number 17)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
a. Evaluate NRC requirements for ensuring that sufficient event data are retrievable to accurately reconstruct the event following a loss of offsite power.	NRR	Generic
b. Evaluate the licensee's process for post-trip review and evaluation, including the thoroughness of review and oversight provided by the onsite and offsite nuclear safety review groups.	Region V	Plant-specific

Item: Adequacy of licensee's recordkeeping practices.
References: Section 6.5 and Finding number 20)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the adequacy of the licensee's maintenance records.	Region V	Plant-specific

Item: Adequacy of operator training and/or procedures.
References: Section 7 and Finding numbers 14, 15 and 16)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Review the implementation of the training program regarding operator understanding and actions in the area of electrical systems, and reviewing technical specification revision statements.	Region V	Plant-specific

Item: Adequacy of emergency notifications and NRC response.
References: Section 7.3 and Finding number 22)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Verify the adequacy of the licensee's procedures and training for reporting of events to NRC Operations Center.	Region V	Plant-specific
Evaluate the need for changes in NRC policy or guidance regarding: the use of the EMS line; the use of NRC personnel as EMS communicators; and possible approaches to improve the ability to determine the overall plant status.	IE	Generic

as: Significance of backlog of license amendments.
(reference: Commission briefing)

<u>Item</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate whether a backlog of license amendments and technical specification changes contributed to delays in approving the licensee's program.	NRR	Plant-specific



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 13 1986

MEMORANDUM FOR: / Harold R. Denton, Director, NRR
James M. Taylor, Director, IE
John B. Martin, Regional Administrator, Region V

FROM: Victor Stallo, Jr.
Acting Executive Director
for Operations

SUBJECT: STAFF ACTIONS RESULTING FROM THE INVESTIGATION
OF THE DECEMBER 26, 1985 INCIDENT AT RANCHO
SECO (NUREG-1195)

An advance copy of the subject report was transmitted to you by memorandum dated February 15, 1986 from the Rancho Seco Team Leader, Frederick J. Hebdon. The report documents the Team's efforts in identifying the circumstances and causes of the December 26, 1985 incident, together with findings and conclusions which form the bases for follow-on actions.

You will note from the report that the licensee has completed troubleshooting the quarantined equipment; however, some additional analysis and testing of the Integrated Control System (ICS), and particularly the ICS Power Supply Monitor, is still planned by the licensee. Consequently, the results of future analysis and testing may form the basis for additional follow-on actions. The identification of these additional actions is a responsibility of the normal program offices. The responsibility for the follow-up and reporting on the licensee's continued analysis and testing is Region V.

The purpose of this memorandum is to identify and assign responsibility for generic and plant-specific actions resulting from the investigation of the Rancho Seco incident as documented in NUREG-1195. In this regard, you are requested to review the enclosure which specifies staff actions resulting from the investigation of the Rancho Seco incident. You are requested to determine the actions necessary to resolve each of the issues in your area of responsibility and, where appropriate, identify additional staff actions or revisions as our review and understanding of this event are refined. Plant-specific actions required for plant restart should receive priority attention.

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view of the importance of this subject, I intend to closely monitor the resolution of these items. By April 14, 1986, please provide a written summary of the plans, schedule, and status for each item within your responsibility listed in the enclosure or that you have identified. Further, I request that you prepare a written status report on the disposition of your items (and anticipated actions for uncompleted items) within 6 months. Every effort should be made to resolve these items promptly.

This enclosure is based on the Team's report and its presentation to the Commission on February 25, 1986. Accordingly, it does not include all licensee actions, nor does it cover NRC staff activities associated with normal event follow-up such as authorization for restart, plant inspections, possible enforcement items. These items are expected to be defined and implemented in a routine manner. Overall lead responsibility for staff actions relating to facility restart is separate from this effort and rests with NRR. Thus, NRR is responsible for coordinating and promptly communicating staff's requirements which must be resolved before operations at Rancho Seco may be resumed. Other offices involved in plant-specific actions are to coordinate their efforts with NRR.

Original signed by
Victor Stello

Victor Stello, Jr.
Acting Executive Director
for Operations

Enclosure:
as stated

With enclosure:
1. Davis, NMSS
1. Murley, RI
1. N. Grace, RII
1. Keppler, RIII
1. Martin, RIV

Distribution
DO S/F
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Reading File
See previous concurrence - Revised per Stello 3/13/86

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**STAFF ACTIONS RESULTING FROM THE INVESTIGATION
OF THE DECEMBER 26, 1985 RANCHO SECO INCIDENT
(Reference: NUREG-1195)**

Issue: Adequacy of the Auxiliary Feedwater System.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
a. Verify the acceptability of the existing initiation and control of the AFW system. (Section 7.2)	NRR	Plant Specific
b. Determine the status of any licensee commitment to install the EFIC system, and determine the acceptability of the current schedule for installation. (Principal Finding #11)	NRR	B&W Generic Review

Issue: Completeness of various staff and licensee actions associated with control systems.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
a. In light of the ongoing B&W generic review, assess the need to reevaluate the actions taken by the staff and by the licensees in response to the findings, conclusions, and recommendations associated with BAW-1564; Bulletin 79-27; NUREG-0667; the February 1980 loss of NNI power at Crystal River; the March 19, 1984 partial loss of NNI at Rancho Seco; and BAW-1791. (Principal Finding #15 and Other Finding #11).	NRR	Generic
b. Assess the need to expand the scope of USI A-47 to include additional consideration of frequent events with undesirable consequences even if the consequences of a particular event are bounded by the FSAR analysis, and the degree to which events that are not significant at the referenced plants might be significant at other plants. (Principal Finding #15h)	NRR	Generic
c. Consider the need for issuing further generic communications.	IE	Generic

ue: Adequacy of the design of the integrated control system (ICS).

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
ess the adequacy of the design the ICS. Particular features be included are:	NRR	B&W Generic Review

Whether the ICS is sufficiently reliable to assure that the frequency of unnecessary safety challenges is acceptably low (i.e., is the ICS properly classified as a nonsafety-related system).

Loss of remote (i.e., hand) power coincident with loss of automatic control. (Principal Finding #2)

Results of SMUD's analysis to date of the power supply monitor. (Principal Finding #1)

Results of SMUD's contractor analysis of the power supply monitor. (Principal Finding #1)

Role of the power supply monitor as a potential single failure in the ICS and/or NNI. (Principal Finding #1)

Results of a study of the response of the ICS upon restoration of power. (Other Finding #3)

Acceptability of the failure mode of meters and recorders that are affected by a loss of ICS power. (Other Finding #4)

ssue: Adequacy of the maintenance program for manual isolation valves.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the need for a maintenance program for manual isolation valves in safety-related systems. (Principal Finding #3)	Region V	Plant Specific

Evaluate the adequacy of industry standards and NRC requirements regarding periodic testing and maintenance of manual valves in safety-related systems. (Principal Finding #3)

NRR

Generic

ie: Adequacy of procedures and training.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the adequacy of the procedures and operator (licensed nonlicensed) training, particularly with regard to:		
The degree to which event specific procedures (e.g., loss of ICS, station black-out) are needed to quickly recover from events that have been diagnosed by the operators, and to mitigate such events if the initiating condition cannot be immediately corrected. (Principal Findings #4 and #7)	NRR	Generic
The consistency between the EOPs and the ATOGs, and the adequacy of operator training on EOPs.	Region V	Plant Specific
The consistency between EOPs and NRC approved procedure generation packages.	IE	Generic
The adequacy of procedural guidance concerning: (1) when to trip auxiliary feedwater pumps and (2) the relative priorities of avoiding the PTS region and maintaining pressurizer level. (Principal Findings #5 and 6)	Region V NRR	Plant Specific B&W Generic Review
The adequacy of the training of nonlicensed operators in the use of valve wrenches, the methods for manually overriding and operating valves, and the use of various indications of valve position. (Principal Finding #9)	Region V	Plant Specific

The adequacy of the annunciator response procedures (e.g., the Annunciator Procedures Manual). (Other Finding #2)

Region V

Plant Specific

The adequacy of the program to insure that all applicable procedures and operator training are reviewed and updated when plant modifications are made. (Other Finding #6)

Region V

Plant Specific

Issue: Adequacy of the radiological control and emergency preparedness program.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the adequacy of the radiological control and emergency preparedness program, including procedures, operator training, equipment availability, and coordination between operating and health physics personnel. (Principal Finding #10 and other finding #5)	Region V	Plant Specific

Issue: Adequacy of the FSAR accident analysis.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Verify the adequacy of the Rancho de FSAR accident analysis, particularly the degree to which credit is given for the nonsafety-related ICS and the nonsafety-related Main Steamline Failure Logic. (Section 9) and (Principal Finding #14)	NRR	B&W Generic Review

Issue: Adequacy of required staffing.

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the adequacy of plant staffing to deal with expected operational transients. (Other finding #7)	NRR	B&W Generic Review

adequacy of the troubleshooting program.

Action

Responsible Office

Category

Region V

Plant specific

the adequacy of the
program to trouble-
shooting equipment in a
thorough and systematic manner
to determine the root cause and
to take corrective actions.
(Attachment #9)



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 05 1985

MEMORANDUM FOR: Harold R. Denton, Director, NRR
James M. Taylor, Director, IE
Robert B. Minogue, Director, RES
C. J. Heltemes, Jr., Director, AEOD
James G. Keppler, Regional Administrator, RIII

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: STAFF ACTIONS RESULTING FROM THE INVESTIGATION
OF THE JUNE 9 DAVIS-BESSE EVENT (NUREG-1154)

An advance copy of the subject report was transmitted to you by memorandum dated July 22, 1985 from the Davis-Besse Team Leader, C. E. Rossi. The report documents the Team's efforts in identifying the circumstances and causes of the June 9, 1985 event, together with findings and conclusions which form the basis for identifying follow-on actions.

You will note from the report that the licensee has not completed troubleshooting and the determination of root causes for all equipment failures or malfunctions. Consequently, the results of future troubleshooting or analysis activities may form the basis for additional follow-on actions. The identification of these additional actions is a responsibility of the normal program office. The responsibility for the followup and reporting on the licensee's continued troubleshooting and determination of root cause for equipment failures is Region III.

The purpose of this memorandum is to identify and assign responsibility for generic and plant-specific actions resulting from the investigation of the Davis-Besse event (documented in NUREG-1154). In this regard, you are requested to review the enclosure which specifies staff actions resulting from the investigation of the June 9 Davis-Besse event. You are requested to determine the actions necessary to resolve each of the items in your area of responsibility and, where appropriate, identify additional staff actions or revisions as our review and understanding of this event are refined. Plant-specific actions required for plant restart should receive priority attention.

Although the NRC Team that investigated the Davis-Besse event did not identify major NRC deficiencies, nonetheless this event provides an opportunity to learn from experience and to feed back the pertinent lessons into our activities. Consequently, all responsible program managers should conduct an in-depth and searching reappraisal of the effectiveness of their programs and the lessons of the Davis-Besse event. In sum, how can we make our programs more effective

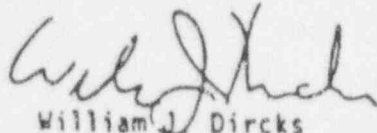
8508090534 10PP

and the NRC a better regulatory agency? For example, what actions are needed when a utility continues to receive low SALP ratings; what impediments or procedures are delaying decisions regarding needed plant upgrades; how can effective corrective action be achieved when plants have a history of maintenance deficiencies; and what should be done when voluntary licensee improvement programs prove less than satisfactory? We need to reflect on these and similar questions and identify further, perhaps more focused actions to gain needed improvements.

In view of the importance of this subject, I intend to have periodic progress review meetings. The first meeting will be in September, and at that time you should be prepared to: (1) discuss the schedule and status of each item within your responsibility listed in the enclosure or that you have identified; and (2) provide a written summary of those actions you have identified for achieving improvements in your program areas. Further, I request that you prepare a written status report on the disposition of your items (and anticipated actions for uncompleted items) within six months. Every effort should be made to dispose of these items promptly.

The enclosure is based directly on the NRC Team's report. Accordingly, it does not include all licensee actions, nor does it cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or possible enforcement items. These items are expected to be defined and implemented in a routine manner. Overall lead responsibility for staff actions relating to facility restart is separate from this effort and rests with NRR. Additionally, NRR is responsible for coordinating and promptly communicating the staff's requirements which must be resolved before operations at Davis-Besse may be resumed. Other offices involved in plant-specific actions are to coordinate their efforts with NRR.

Separately from this action, I will be discussing with you further how we may improve the IIT procedures based upon the experience with the Davis-Besse Team.



William J. Dircks
Executive Director for Operations

Enclosure.
As Stated

cc w/enclosure:
J. Davis, NMSS
T. Murley, RI
J. N. Grace, RII
R. Martin, RIV
J. Martin, RV

STAFF ACTIONS RESULTING FROM THE INVESTIGATION

OF THE JUNE 9 DAVIS-BESSE EVENT

(Reference: NUREG-1154)

1: Adequacy of the licensee's management and maintenance practices.
(Reference: Conclusion Section 8)

	<u>Responsible Office</u>	<u>Category</u>
Evaluate and take action on the licensee's response to findings relating to corrective actions and preventive maintenance problems (including testing, root cause determination of equipment misoperation and operating experience).	NRR	Plant-specific

Evaluate and take action on the licensee's response to findings concerning management practices (e.g., control of maintenance programs and post-trip reviews).	Region III	Plant-specific
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Completion of analyses for loss of feedwater events.
(Reference: Section 7)

	<u>Responsible Office</u>	<u>Category</u>
ate the time margins and quences of alternative nces for a loss of feed-event at Davis-Besse.	NRR	Plant-specific

Adequacy of the Steam Feedwater Rupture Control System (SFRCS).
(References: Section 5.2.2 and Finding 6)

	<u>Responsible Office</u>	<u>Category</u>
the design basis for SFRCS he susceptibility of the SFRCS a) spurious actuations involving items as MSIV closure; and b) e failures.	NRR	Plant-specific

4. Item: Interaction of plant security features and operator actions.
(References: Section 3.6 and Finding 9)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the effect of security features (locked doors, locked equipment, etc.) on the operator's ability to gain prompt access to equipment required to perform safety actions outside the control room in accordance with emergency procedures.	NRR	Plant-specific Generic

5. Item: Availability of the Shift Technical Advisor (STA)
(References: Section 6.1.3 and Finding 14)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the time available and role for STA assistance during complex operating event.	NRR	Plant-specific Generic

6. Item: Reliability of the AFW containment isolation valves and other safety-related valves.
(References: Section 5.2.5 and Findings 4, 5, 6, and 15)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(a) Monitor the licensee's troubleshooting activities.	Region III	Plant-specific
(b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions.	NRR	Plant-specific
(c) Determine if the safety function of the AFW containment isolation valves has been properly specified, i.e., are the valves required to open as well as close for design basis events.	NRR	Plant-specific
(d) Verify that these valves constitute a single failure point for the AFW system for certain design basis events.	NRR	Plant-specific

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(e) Determine that the procedures for adjustments of the AFW isolation valves such as torque switch bypass switches are clear and proper, and that the associated training programs are adequate. Confirm that adjustment settings are consistent with plant procedures.	Region III	Plant-specific
(f) Determine if the engineering basis for the specification of the adjustments for safety-related valves such as the torque switch and torque switch bypass switch settings are adequate for all design basis events.	NRR	Plant-specific
(g) Evaluate the test program for the AFW containment isolation valves to confirm operability for all design basis events.	NRR	Plant-specific
(h) Evaluate whether other safety-related valves in Davis-Besse may be subject to the same type/cause of failure.	NRR	Plant-specific
(i) Conduct a review of failures of safety-related motor-operated valves and provide an assessment of pertinent failure modes affecting valve performance under design basis conditions.	AEOD	Generic
(j) Determine if further generic correspondence, such as an NRC Bulletin, is warranted on this type/cause of failure of safety-related valves.	IE	Generic

7. Item: Adequacy of emergency notifications.
(References: Section 6.1.4 and Finding 12)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(a) Verify the adequacy of the licensee's procedures and training for reporting of events to the NRC Operations Center.	Region III	Plant-specific

- | | | |
|--|----|---------|
| (b) Review the adequacy of NRC guidance for determination of severity levels when plant conditions vary and may be stable when the licensee has an opportunity to report. | IE | Generic |
| (c) Review the adequacy of shift staffing for assuring that knowledgeable individuals will be available for properly implementing the emergency plan during complex and long operational events. | IE | Generic |

8. Item: Reliability of the AFW pump turbines.
(References: Sections 5.2.4 and 6.2.4 and Findings.4, 8, and 15)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(a) Monitor the licensee's troubleshooting activities including possible hot plant operation to confirm failure mode.	Region III	Plant-specific
(b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions.	NRR	Plant-specific
(c) Evaluate the licensee's response and corrective actions relating to the unreliability of the auxiliary feedwater system (including the need for a third pump and turbine trip reset capability).	NRR	Plant-specific
(d) Verify that the AFW system has been adequately tested to confirm system configuration involved with design basis events.	Region III	Plant-specific
(e) Review the implementation of the operator training program to assure proper operator actions, such as resetting of trip throttle valve.	Region III	Plant-specific
(f) Conduct a review of past operating experience and determine the causes for overspeed turbine trips.	AEOD	Generic

- | | | |
|---|----|---------|
| (g) Determine the need for further generic correspondence on this failure mode/cause. | IE | Generic |
|---|----|---------|

9. Item: Reliability of the PORV.
(References: Sections 5.2.8 and 6.2.1 and Findings 10 and 13)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(a) Monitor the licensee's troubleshooting activities.	Region III	Plant-specific
(b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions.	NRR	Plant-specific
(c) Determine the need for a test program to establish reliability.	NRR	Generic
(d) Determine if surveillance tests are necessary to confirm operational readiness.	NRR	Generic
(e) Determine if additional protection against PORV failure is necessary, i.e., automatic block valve closure.	NRR	Generic

10. Item: Adequacy of control room instrumentation and controls.
(References: Sections 6.1.1, 6.1.2, and 6.2.2 and Findings 10, 11, 17, and 18)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(a) Evaluate the adequacy of the SFRCs actuation controls and associated training program.	NRR	Plant-specific
(b) Evaluate the adequacy of the installed control room instrumentation to allow operators to make the necessary and prompt determination for procedure conformance and PORV position.	NRR	Plant-specific

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(c) Determine if NRC requirements should be revised regarding: (1) SPDS availability; and (2) the need for plant-specific simulator.	NRR	Plant-specific Generic

11. Item: Need for isolation of the startup feedwater pump.
(References: Section 5.1.3 and Finding 7)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Reassess acceptability of the provisions which resulted in the inability to place the startup feedwater pump in service from the control room.	NRR	Plant-specific

12. Item: Resolution of equipment deficiencies.
(References: Section 5 and Table 5.1)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
(a) Monitor the licensee's troubleshooting activities.	Region III	Plant-specific
(b) Evaluate the licensee's engineering report on the root cause analysis and corrective action for the equipment listed on Table 5.1 and not addressed by other items in this action plan.	NRR	Plant-specific
(c) Determine the need for generic correspondence on equipment problems.	IE	Generic

13. Item: Adequacy of plant procedures.
(References: Sections 6.1.1 and 6.1.2 and Findings 10 and 17)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Verify that plant procedures involving "drastic" actions are required to be sufficiently precise and clear to ensure prompt implementation.	NRR	Generic

14. Item: Adequacy of safety system testing.
(Reference: Finding 15)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Evaluate the NRC requirements and guidance to assure that safety systems are tested in all configurations required by the design basis analysis.	NRR	Generic

15. Item: Acceptability of current safety assessment methods.
(References: Findings 1 and 2)

<u>Action</u>	<u>Responsible Office</u>	<u>Category</u>
Assess the implications of multiple independent and common mode failures as they relate to departures from design assumptions and specifications used in probabilistic safety analyses.	RES	Generic

ENCLOSURE 3

NRC STAFF PERSONNEL CONTACTED

in Onofre IIT Action Items

Stuart A. Richards, RV
Charles M. Trammell III, NRR
Thomas M. Novak, AEOD
Wayne D. Lanning, NRR
Thomas T. Martin, RI
Aleck W. Serkiz, RES
William Ang, RV
Howard J. Wong, RV
Anthony J. D'Angelo, RV
Phillip H. Johnson, RV
Don G. Marksberry, AEOD
Eric Weiss, AEOD

Sancho Seco IIT Action Items

Stuart A. Richards, RV
George Kalman, NRR
Byron Siegel, NRR
Rick Kendall, NRR
Robert C. Jones, NRR
Frederick Hebdon, NRR

Miss Besse IIT Action Items

Ronald C. Emrit, RES
Ihor M. Jackiw, RIII
Paul M. Byron, RIII
Thomas V. Wambach, NRR
Albert W. DeAgazio, NRR
Carl Johnson, RES
James C. Glynn, RES
Jitendra P. Vora, RES
Louis M. Shotkin, RES
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Thomas M. Novak, AEOD
Wayne D. Lanning, NRR

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Thomas M. Novak, AEOD
Wayne D. Lanning, NRR

POWER REACTOR

EVENT NUMBER: 26877

FACILITY: DRESDEN
UNIT: [] [] [3]
RX TYPE: [1] GE-1, [2] GE-3, [3] GE-3

REGION: 3
STATE: IL

NOTIFICATION DATE: 03/03/94
NOTIFICATION TIME: 19:59 [ET]
EVENT DATE: 03/03/94
EVENT TIME: 18:30 [CST]
LAST UPDATE DATE: 03/03/94

NRC NOTIFIED BY: THRONE
HQ OPS OFFICER: GREG SCARFO

NOTIFICATIONS

EMERGENCY CLASS: NOT APPLICABLE
10 CFR SECTION:
AOUT 50.72(b)(1)(ii)(B) OUTSIDE DESIGN BASIS
NLCO TECH SPEC LCO A/S

PATRICK HILAND RDO

UNIT	SCRAM CODE	RX CRIT	INIT PWR	INIT RX MODE	CURR PWR	CURR RX MODE
3	N	Y	63	POWER OPERATION	63	POWER OPERATION

EVENT TEXT

THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE WHEN AN ANALYSIS DETERMINED THAT TWO STEAM LINE ISOLATION VALVES WOULD NOT CLOSE DURING A HPCI STEAM LINE BREAK.

AN ENGINEERING ANALYSIS DETERMINED THAT THE CRANE MOTOR-OPERATED VALVES WITH CARBON STEEL DISCS AND VALVE GRINDS UTILIZED AS HPCI STEAM LINE ISOLATION VALVES ("3-2301-4", "3-2301-5") WILL NOT CLOSE DURING A HPCI STEAM LINE BREAK. THIS ANALYSIS WAS BASED ON DATA FROM EPRI BLOWDOWN TESTING AND FOUND THAT THE TWO-ROTOR MOTOR OPERATORS CANNOT GENERATE THE TORQUE NECESSARY TO CLOSE THE VALVES. THEREFORE, THE HPCI SYSTEM WAS DECLARED INOPERABLE AND TECHNICAL SPECIFICATION ACTION STATEMENT 3.5.C (7 DAY LCO) WAS ENTERED.

THE VALVES HAVE BEEN CLOSED AND TAKEN OUT OF SERVICE. THE "FAULTY" VALVE OPERATORS WILL BE REPLACED WITH FOUR-ROTOR MOTOR OPERATORS AND A TORQUE SWITCH BYPASS CAPABILITY DURING THE UPCOMING REFUELING OUTAGE (03/12/94). THE EPRI DATA INDICATES THAT THESE OPERATORS GENERATE AN ADEQUATE TORQUE TO ALLOW ISOLATION OF THE HPCI STEAM LINE. UNIT #2 CURRENTLY UTILIZES SUCH VALVE OPERATORS ON ISOLATION VALVES IN THE HPCI SYSTEM. THE ISOLATION CONDENSER AND ALL LOW PRESSURE ECCS SYSTEMS (CORE SPRAY, LPCI) ARE FULLY OPERABLE. THE LICENSEE INFORMED THE NRC RESIDENT INSPECTOR.

POWER REACTOR

EVENT NUMBER: 26884

FACILITY: QUAD CITIES
UNIT: [1] [] []
RX TYPE: [1] GE-3, [2] GE-3

REGION: 3
STATE: IL

NOTIFICATION DATE: 03/05/94
NOTIFICATION TIME: 01:17 [ET]
EVENT DATE: 03/04/94
EVENT TIME: 23:00 [CST]
LAST UPDATE DATE: 03/05/94

NRC NOTIFIED BY: HOUZENGA
HQ OPS OFFICER: TIM MCGINTY

NOTIFICATIONS

EMERGENCY CLASS: NOT APPLICABLE
10 CFR SECTION:
AIND 50.72(b)(2)(iii)(D) ACCIDENT MITIGATION
NLCO TECH SPEC LCO A/S

PATRICK HILAND RDO

UNIT	SCRAM CODE	RX CRIT	INIT PWR	INIT RX MODE	CURR PWR	CURR RX MODE
1	N	Y	87	POWER OPERATION	87	POWER OPERATION

EVENT TEXT

BASED ON AN ENGINEERING EVALUATION, IT WAS DETERMINED THAT THE UNIT 1 RCIC (RX CORE ISOLATION COOLING) CONTAINMENT INBOARD STEAM ISOLATION VALVE (MO-1-1701-16) WOULD NOT CLOSE AS PER DESIGN.

THE VALVE IS A 6" GATE VALVE, MANUFACTURED BY CRANE. BASED ON THE RESULTS OF EPRI TESTING UNDER BLOWDOWN CONDITIONS AT RATED PRESSURES ON THESE VALVES, WHICH DETERMINED VALVE FACTORS TO BE AS HIGH AS .85, THE LICENSEE EVALUATED THE MOTOR SIZING CALCULATIONS AND PREVIOUS TEST RESULTS ON EIGHT SUCH VALVES. THE "16" VALVE IS THE ONLY ONE THAT COULD NOT MEET THE OPERABILITY ACCEPTANCE CRITERIA. THE PREVIOUS ANALYSIS FOR THE "16" VALVE ASSUMED A VALVE FACTOR OF .447.

THE RCIC SYSTEM WAS DECLARED INOPERABLE AND A 14 DAY LCO A/S ENTERED PER TS 3.5.E. ALL ECCS SYSTEMS ARE OPERABLE AT THIS TIME. THE SYSTEM HAS BEEN ISOLATED BY CLOSING AND TAKING OUT OF SERVICE THE "17" VALVE (OUTBOARD CONTAINMENT ISOLATION). THE LICENSEE WILL CONTINUE TO EVALUATE THE SITUATION FOR CORRECTIVE ACTIONS, AND HAS INFORMED THE RESIDENT INSPECTOR.

c/3