VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

February 3, 1988

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

Serial No. 88-009
NAPS/DBR:vlh
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
NRC INSPECTION REPORT NOS. 50-338/87-36 AND 50-339/87-36

We have reviewed your letter of January 4, 1988 which referred to the inspection conducted at North Anna between October 20, 1987 and November 19, 1987 and reported in Inspection Report Nos. 50-338/87-36 and 50-339/87-36. The response to the Notices of Violation are addressed in the attachments.

We share your concern with the number and type of events which have occurred during the Unit 2 outage and startup, and concur with your assessment that the principal common thread throughout these events is inadequate procedures and failure to follow procedures. Each of these events have been evaluated by station management for cause, corrective actions and effectiveness of the corrective actions once implemented. These events have also been assessed in aggregate for common threads and corrective actions to address the common thread issues. In addition to the station management evaluations, we have also incorporated the results of reviews performed under the Human Performance Evaluation System and the Significant Event Review Program. The Significant Event Review Program is conducted with the Vice President of Nuclear Operations, station management and the personnel involved whenever a significant event occurs. Corrective actions dealing with failure to follow procedures (which were discussed with you on January 21, 1988) and status are summarized below:

1. A Station Manager memorandum to all station personnel on Compliance with Station Procedures was issued on January 19, 1987. Attached to this memorandum was a copy of NRC Inspection Report Nos. 50-338/87-36 and 50-339/87-36.

2. Station supervisors have been briefed on the recent events by the Station Manager, Assistant Station Manager for Operations and Maintenance, Corporate Management, and the Senior Vice President of Engineering, Construction and Power Operations. Station personnel are

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being briefed in individual groups with their supervisor by the Station Manager and Assistant Station Manager for Operations and Maintenance. Each briefing is structured so as to cover specific strengths and weaknesses as well as the corrective actions for the common thread issues. Employee attention to detail and rigorous compliance with standards are the key elements of discussion. These briefings were initiated on January 26, 1988.

3. Management attention to and involvement in post maintenance operability and required Technical Specification surveillance testing will be strengthened, especially for startups after major outages. This will be accomplished by improved scheduling of testing, assignment of Operations Coordinators (experienced SROs) to overview operations during startup and to manage the required testing, continuation of the station management required walkdowns prior to each startup, and continuation of the Station Safety Committee review of open safety related work orders prior to each startup.

4. Control of procedures has been strengthened by requiring safety committee review, prior to use, of any change to safety related procedure initial conditions.

5. Mode change checklists are being developed to assist the operators in verifying the status of Technical Specifications required equipment prior to each mode change. These checklists will be developed by March 15, 1988.

6. An assessment will be performed of the methodology and controls currently being used in our Technical Specifications surveillance procedures to determine the need for any necessary changes to specifically address their utilization during outages.

7. The review process for determining acceptability of using periodic test procedures for post maintenance testing is being enhanced to assure procedures are adequate for determining equipment operability and will not adversely impact the unit or other equipment. This review is being directed by the Assistant Station Manager for Operations and Maintenance and will incorporate, as appropriate, INPO Good Practice 87-028 (MA-305), Post Maintenance Testing.

8. The Station Deviation Report program is being revised to augment the evaluation of trends and root causes of events. Enhanced root cause evaluation is a high priority 1988 goal for the station and the Nuclear Operations Department and is being specifically emphasized by the Vice President of Nuclear Operations. Trending of station Deviation Reports involving personnel error is being performed and reported to management as well as being screened for potential followup under the Human Performance Evaluation System.

We have no objection to this inspection report being made a matter of public record. If you have any further questions, please contact us.

Very truly yours,

WLS

W. L. Stewart

Attachments

cc: U. S. Nuclear Regulatory Commission
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Atlanta, Georgia 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

ATTACHMENT

RESPONSE TO THE NOTICE OF VIOLATION REPORTED DURING THE NRC INSPECTION CONDUCTED BETWEEN OCTOBER 20, 1987 AND NOVEMBER 19, 1987 INSPECTION REPORT NOS. 50-338/87-36 AND 50-339/87-36

NRC COMMENT

During the Nuclear Regulatory Commission (NRC) inspection conducted on October 20 - November 19, 1987, violations of NRC requirements were identified. The violations involved inadequate procedures and failure to follow procedures. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1987), the violations are listed below:

A. Technical Specification 6.8.1.c requires written procedures be established, implemented and maintained covering surveillance and test activities of safety related equipment.

   Contrary to the above on October 26, 1987, the 2J Emergency Diesel Generator was electrically overloaded due to an inadequate surveillance procedure 2-PT-83.4, Blackout of Emergency Bus for Shutdown Loads. The procedure 2-PT-83.4 was inadequate in that the test equipment voltage criteria for establishing the diesel load was improperly calculated. This improper calculation resulted in the diesel being loaded to 3100 kw instead of the Technical Specification range of 2900 to 3000 kw. This violation is similar to violation 339/86-04-01 which cited the licensee for overloading the 2J diesel per 2-PT-83.4 in February of 1986, the last performance of the procedure.

   This is a Severity Level IV violation (Supplement 1) and applies to Unit 2.

RESPONSE

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

   The violation is correct as stated. This violation is similar to the violation identified in IEIR 86-04 in that the same diesel was overloaded. However, the root causes of the two events are believed to be different. The earlier event in 1986 was a result of inadequate attention by operations personnel to equipment operating in a mode where output of the diesel could be affected by power grid conditions. Operations personnel had not been adequately trained on the possible changes in diesel output which can result from changes in system grid conditions when the diesel is operated in parallel with the system. Therefore, at that time, the operator assigned to the test had responsibility for all Backboard indications after the test was started and the diesel thought to be operating in a stable condition. As part of the corrective action to address this overload issue, a dedicated operator was assigned to monitor the EDG in the Control Room whenever it is operating.

The most recent event was the result of an inadequate procedure which specified that diesel load be monitored by a voltage meter which was incorrectly correlated to diesel load and which did not require verification of proper loading by use of installed instruments. The operator assigned to the test notified the test engineer that the load indication from the permanently installed meter did not agree with the test meter as diesel load was being increased, but was told that the test meter provided the more accurate indication. When the operator saw that the magnitude of difference in readings was close to 100 KW at the desired load, he again questioned the accuracy of the test meter and notified his supervision. Therefore, it was through the close involvement of the operator that was dedicated to monitor the EDG that the overloaded condition was immediately detected and timely corrective action initiated. The root cause of this event was an inadequate procedure. The corrective action implemented as a result of the 1986 event was effective in identifying the discrepant condition and initiating the appropriate action.

2. REASON FOR THE VIOLATION:

During the performance of 2-PT-83.4, a voltmeter was installed to read load on the emergency diesel generator (EDG). This test meter was used to allow operators to read the EDG load more precisely.

Although the test procedure did not call for monitoring diesel load by using the permanently installed Control Room meter, the operator assigned to the test was closely checking this meter against the test meter as load was increased. The operator questioned the test engineer when he noted that there was a difference in the two readings, but was assured that the test meter was more accurate. When the EDG load reached approximately 3100 KW as read on the installed meter, the operator initiated discussions with the Shift Supervisor. After a brief discussion, it was decided to reduce load to below 3000 KW as indicated by the permanently installed load meter. After additional review, it was discovered that the calculations used to determine the correlation between the voltage reading on the voltmeter and the load in KW was in error. The calculation assumed that the meter read from 0 to 4000 KW, when in fact, the meters full scale was 4200 KW. This caused the test voltmeter reading to show a lower load than actual. The test engineer submitted a plant deviation report identifying that the diesel had been loaded to greater than 3000 KW.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

A procedure deviation for 2-PT-83.4 was written to incorporate the new voltage to load correlation and the procedure was completed without further incident.

The period of time that the diesel was overloaded was estimated to be less than five minutes, but to conservatively evaluate the impact on the diesel, it was assumed to be overloaded for ten minutes. Discussions were held with the EDG manufacturer, Colt Industries, who determined that a ten minute overload would have essentially no effect on the EDG since the EDG is rated for 168 hours at 3100 KW.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

2-PT-83.4 and 1-PT-83.4 will be revised to incorporate the correct correlation for voltage to load and to require using all available indications to verify proper loading.

5. THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:

The procedure revisions will be completed by March 1, 1988. These procedures are not planned to be used until the next refueling outage which is currently scheduled for Unit 2 to begin in November, 1988.

B. Technical Specification 6.8.1.a and c. requires written procedures be established, implemented and maintained covering procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and surveillance and test activities of safety related equipment.

Regulatory Guide 1.33, Appendix A, Section 3, requires procedures for Startup, Operation, and Shutdown of safety-related systems including energizing, filling, venting, draining, shutdown, and changing modes of operation for systems including emergency core cooling, auxiliary feedwater, and nuclear instrumentation.

This violation has five examples:

(1) Contrary to the above on October 14, 1987, the Unit 1 Refueling Water Storage Tank (RWST) level was allowed to drop below Technical Specification 3.5.5 limit. This RWST level reduction was due to an operator failing to follow procedure 1-OP-7.7, Operation of RWST Systems, by performing the restoration valve lineup out of sequence.

(2) Contrary to the above, on October 26, the Unit 2 Pressurizer Power Operated Relief Valve (PORV) was inadvertently lifted due to the operator failing to follow a surveillance procedure 2-PT-211.2. The operator cycled a safety injection valve aligning the discharge of the operating charging pump to the RCS which was solid. The operator failed to verify the initial conditions of the procedure which required all the charging pumps to be secured.

(3) Contrary to the above, on October 31, an unexpected reactor protection system activation occurred on Unit 2 due to an inadequate procedure. The unit was in Mode 4 with the scram breakers closed. The reactor protection trip signal was generated by a surveillance procedure 2-PT-71.4 which was testing the time response of the auxiliary feedwater pumps. This procedure was inadequate in that it did not identify to the operators that a reactor trip signal would be generated with the start signal for the auxiliary feedwater pumps.

(4) Contrary to the above, on October 30, Unit 2 was allowed to continue to operate in Mode 2 with an inoperable power range instrument, N44, without the instrument's protection signals being placed in trip as required by Technical Specification Table 3.3-1. This situation occurred due to inadequate procedures 2-PT-94.0 and ICP-2-N44, which placed the N44 instrument in the inoperable condition but did not require the protection signals to be placed in trip.

(5) Contrary to the above, on October 30, the Reactor Coolant System (RCS) TAVE for Unit 2 was inadvertently decreased below the Technical Specification 3.1.1.5 of 541 degrees F. This inadvertent TAVE reduction was caused by the operator placing the power range instrument in a trip condition without an adequate procedure. The method used to place N44 in trip also generated an open signal to the feedwater regulating bypass valves which resulted in the inadvertent cooldown of the steam generators and consequently the RCS.

This is a Severity Level IV violation (Supplement 1) and applies to both units.

RWST VOLUME DECREASE

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

The violation is correct as stated.

2. REASON FOR THE VIOLATION:

The cause of the violation was failure of personnel to follow procedure. Steps in 1-OP-7.7, Operation of Refueling Water Storage Tank (RWST) Systems, were performed out of sequence. The operator involved was fulfilling his normal watch in the Auxiliary and Fuel Buildings and was also involved in the portion of 1-OP-7.7 that covers make up to the Unit 1 RWST from the Unit 2 blender. When the operator's normal shift was over, he returned to the Control Room/Technical Support Center area for his next assignment. Upon notification that he was to continue his watch station on the next shift, he proceeded to perform the normal watch station rounds. When in the Fuel Building, the operator opened the refueling purification supply valve to the spent fuel pit (SFP). Because this step was performed out of sequence, it established a flow path that allowed water to flow from the Unit 1 RWST to the SFP.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

A Human Performance evaluation of the event was performed. The evaluation concluded the root cause of the event was failure to follow the procedure.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

The Licensed Operator Requalification Program (LORP) will be revised by March 31, 1988 to emphasize the importance of performing valve lineups in a timely manner and performing procedural steps in the specified sequence. Training will be completed by June 1, 1988.

This violation and the response will be placed in required reading for station operations personnel. In addition, the Operation Directives will be reviewed to ensure that the necessary guidance for performing valve lineups is provided.

5. <u>THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:</u>

The LORP training will be completed in accordance with the schedule provided in section 4 of this response.

The violation and response will be reviewed by operations personnel by March 15, 1988. The review and revision, as necessary, of Operations Directives will be completed by April 1, 1988.

PORV LIFT

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

The violation is correct as stated.

2. REASON FOR THE VIOLATION:

The cause of the violation was the failure to follow procedure. 2-PT-211.2, Valve Inservice Inspection (Safety Injection System) was being performed with the Reactor Coolant System primary pressure at 335 psig, primary temperature at 175 degrees F and the system water solid. At the beginning of the test, all initial conditions of the procedure were met (reactor coolant pumps (RCP), high head safety injection (HHSI)/charging pumps, and low head safety injection (LHSI) pumps were secured and the RCS was not water solid). Later, due to the extended time required to complete the test, the initial conditions required for 2-PT-211.2 were not maintained (i.e. one RCP and one HHSI/charging pump were running with the RCS water solid). To compensate for not meeting the initial conditions of the procedure, the alternate charging header was isolated by closing the charging pump discharge motor operated valve to the alternate charging header. Closing this discharge valve isolated 2-SI-MOV-2869A from the operating charging pump. 2-SI-MOV-2869B remained lined up to the charging pump. The inadvertent lifting of the PORVs occurred when 2-SI-MOV-2869B was mistakenly opened instead of 2-SI-MOV-2869A. Opening 2-SI-MOV-2869B caused normal charging flow into the Reactor Coolant System (RCS) via the HHSI flow path. This flow increased primary system pressure and resulted in both pressurizer power operated relief valves (PORVs) opening to relieve the RCS overpressure condition.

The revision of the procedure to address a different lineup based on changed initial conditions should have been completed by the use of a prior to use procedure deviation in accordance with station Administrative Procedure 5.8.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

The Vice President - Nuclear Operations reviewed this event with the individual involved, his supervisor and station management.

An Operations Memorandum has been issued stating that if an initial condition to a procedure must be temporarily changed, then a prior to use procedure deviation will be processed. This memorandum also specified that the initial conditions of a procedure are to be verified upon initial entry and again upon any re-entry into the procedure. A procedure deviation has been processed for station Administrative Procedure 5.8 to require that all deviations to initial conditions of safety related procedures be approved prior to use by the Station Nuclear Safety and Operating Committee (SNSOC).

A Human Performance evaluation of this event has been performed. Corrective actions identified as a result of the evaluation were to revise the labels for 2-SI-MOV-2869A and B to better define the function of the valves and to perform a review for adequacy of other Unit 1 and 2 SI switch and/or indicator labels and make appropriate changes.

2-SI-MOV-2869A and B have been relabelled to designate the appropriate safety injection header.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

The review of Unit 1 and 2 Safety Injection switch labels is in progress and is scheduled to be completed by March 1, 1988.

Administrative Procedure 5.8 is presently under review for a comprehensive re-write to enhance the required review process for procedure deviations to ensure compliance with 10 CFR 50.59 and Technical Specifications. This re-write will include a permanent change to require prior SNSOC approval of deviations to initial conditions in safety related procedures.

5. THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:

Station Administrative Procedure 5.8 will be revised by April 1, 1988. The review of switch labels is an additional enhancement being made to correct a condition which is considered to be a possible contributor. The review and relabelling of SI valves will be completed by March 1, 1988.

INADVERTENT RPS ACTUATION DURING 2-PT-71.4

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

The violation is correct as stated.

2. REASON FOR THE VIOLATION:

The cause of the violation was an inadequate procedure. The purpose of 2-PT-71.4 is to measure the response time of each Auxiliary Feedwater (AFW) pump from a manual start signal and to verify the auto start logic of the AFW pumps. An automatic AFW pump start signal is generated when 2 out of 3 narrow range level transmitters on any one S/G detect less than 18 percent level, and at least one loop stop valve in the respective loop is not fully shut.

During the performance of 2-PT-71.4 with Unit 2 in Mode 4, a S/G low-low level signal was generated by placing two level comparator test switches for the "A" S/G in the "TEST" position. As a result, a reactor trip signal was generated and the AFW pumps received a start signal. The pumps did not start because the motor driven AFW pump breakers were racked in the test position, and the steam supply trip valves to the steam driven AFW pump were isolated.

2-PT-71.4 is normally performed with the Unit in Mode 4, but with the reactor trip breakers open. The requirement to verify that the trip breakers were open, however, was not included as an initial condition in the test procedure. Because of conditions established for another test procedure, the reactor trip breakers were closed when 2-PT-71.4 was performed. When the two level comparator switches for "A" S/G were placed in the "TEST" position, the AFW pumps received a start signal and a reactor trip signal was generated which opened the reactor trip breakers which was not an expected action in 2-PT-71.4. This event was reported in accordance with 10 CFR 50.72.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

A procedure deviation for 2-PT-71.4 was written to add a caution stating that placing two level comparator switches for a given S/G in the "TEST" position would generate a reactor trip signal. An initial condition was also added to verify the reactor trip breakers were open.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

Permanent changes to 1/2-PT-71.4 will be made to incorporate the provisions of the procedure deviation discussed in section 3 of this response.

5. THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:

Procedures 1/2-PT-71.4 will be revised by April 1, 1988. The Unit 2 procedure will be performed next during the refueling outage presently scheduled for the last quarter of 1988.

N-44 INOPERABLE AND NOT PLACED IN TRIP IN ONE HOUR

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

The violation is correct as stated.

2. REASON FOR THE VIOLATION:

The cause of the violation was an inadequate procedure. To perform startup physics testing, the N-44 detector channel was connected to the reactivity computer and placed in trip. To comply with Technical Specification 4.10.3.2, it is required to functionally test (per 2-PT-30.2.4) the power range instrumentation channels prior to the commencement of physics testing. To perform the functional test, the N-44 detector channel was taken out of the trip position. Following functional testing, the N-44 detector channel was not placed back in the trip position since 2-PT-30.2.4 did not require such action. After mode 2 was entered, plant personnel discovered N-44 was not in trip and that the action statement of Technical Specification 3.3.1.1 had been exceeded. Review of procedure PT-94.0, Refueling Nuclear Design Check, revealed that the procedure did not contain instructions to place the N-44 detector channel in trip after connection to the reactivity computer.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

A procedure deviation was processed for PT-94.0 to require placing N-44 in trip per Abnormal Procedure 4.3 when it was connected to the reactivity computer. Just prior to this, it had been identified that the N-44 detector was not in the tripped position, and the current signal had been adjusted to 120 percent power. This action, which was not in accordance with Abnormal Procedure 4.3, Malfunction of Nuclear Instrumentation (Power Range), led to an inadvertent cooldown of the RCS as documented in the fifth example of this violation. To ensure all trip signals associated with N-44 were placed in a tripped condition, the channel was placed in trip using Abnormal Procedure 4.3. The detector current signal was also returned to its actual level.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

PT-94.0, Refueling Nuclear Design Check, will be permanently revised to ensure that the N-44 detector channel is placed in the trip position when the channel is connected to the reactivity computer.

A step will be added to the functional test procedure, PT-30.2.4, requiring that if the N-44 detector channel is being tested prior to physics testing and is connected to the reactivity computer, the channel must be left in trip after the testing has been completed.

5. <u>THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:</u>

1/2-PT-94.0 and 1/2-PT-30.2.4 will be revised by April 1, 1988. The next scheduled startup from a refueling at which time these procedures would be used is December 1988.

COOLDOWN TO LESS THAN 541 F DUE TO OPEN BYPASS VALVES

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

The violation is correct as stated.

2. REASON FOR THE VIOLATION:

The cause of the violation was failure to follow procedure. As discussed in the fourth example of this violation, after discovering that channel N-44 was connected to the reactivity computer but not placed in trip, the channel was incorrectly placed in trip by increasing the test current so that the power range channel would indicate 120 percent power. It was not initially recognized that this action was not the prescribed method to place N-44 in trip. Increasing the current signal does not assure that all output bistables for the channel are in a tripped condition. At the time when the test current was increased, the Feedwater Bypass Flow Control Valve (FCV) controllers for the "A" and "B" S/Gs were in the "AUTO" position. When in "AUTO", the controller for these valves receive a signal from N-44 to anticipate turbine/reactor power mismatch. The apparent reactor/turbine power mismatch resulting from increasing the test current caused the automatic opening of the FCVs which resulted in the cooldown of the RCS to less than 541 degrees F while the reactor was critical.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

To ensure all trip signals were received from N-44, the channel was placed in trip using Abnormal Procedure 4.3, Malfunction of Nuclear Instrumentation (Power Range). The current signal for N-44 was returned to its actual level. Reactor coolant system temperature was increased to above 541°F within 15 minutes to satisfy the Action Statement of T.S.3.1.1.5.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

As discussed in section 4 of the response to the fourth example of this violation, PT-94.0 and PT-30.2.4 will be revised. The revision will require that nuclear instrument channels be placed in trip in accordance with AP-4.3 rather than by adjusting the current output signal. A caution will also be added to AP-4.3 and PT-30.2.4 warning that adjusting the current signal from N-44 will result in changes in feedwater bypass control valve position when they are in "AUTO". AP-4.3 already requires as an immediate action that the Feedwater Bypass FCVs be placed in manual if channel N-44 fails.

5. THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:

The identified changes to PT-94.0, PT-30.2.4, and AP-4.3 for both units will be completed by April 1, 1988. The next scheduled use of the procedures will be during the startup from the Unit 2 refueling outage presently scheduled to begin in November 1988.

C. Technical Specification 6.8.1.c requires written procedures be established, implemented and maintained covering surveillance and test activities of safety-related equipment.

Technical Specification 6.8.3.a allows temporary changes to procedures required by 6.8.1 as long as the intent of the original procedure is not altered.

Technical Specification 6.8.2 requires changes to procedures of 6.8.1 which do not meet the criteria of 6.8.3 for temporary changes shall be reviewed by the safety committee prior to implementation.

Contrary to the above, on October 22, 1987, the licensee changed 2-PT-211.6, Valve Inservice Inspection (Accumulator Isolation MOVs) to allow performance of the test with the Reactor Coolant System (RCS) depressurized. This changed the intent of the procedure which was to only allow performance of the test with the RCS at pressures between 800 and 1000 psig to prevent injection of the accumulator into the RCS. The change was implemented without prior safety committee approval and resulted in the injection of the Unit 2 C accumulator into the RCS.

This is a Severity Level IV violation (Supplement 1) and applies only to Unit 2.

INADVERTENT DISCHARGE OF ACCUMULATOR INTO THE RCS

1. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION:

The violation is correct as stated.

2. REASON FOR THE VIOLATION:

Personnel error in the determination of whether a prior to use deviation was required resulted in the intent of procedures not being followed. The "C" accumulator tank was isolated from its nitrogen supply in order to perform maintenance on the accumulator discharge isolation valve, MOV-2865C. A continuous vent path to bleed off accumulator pressure had been established by tagging open an instrument piping vent valve on one of the two separate accumulator standpipes. Each standpipe can be isolated from the accumulator by two isolation valves. The isolation valves on the standpipe were not tagged open to ensure a continuous vent path on the "C" accumulator tank. As a result, the isolation valves on both instrument standpipes were subsequently tagged closed when maintenance was performed on the instrument piping flanges. The closure of the standpipe isolation valves unintentionally isolated the vent path from the accumulator. This action also isolated all control room pressure and level instrumentation for the accumulator. It was the inappropriate use of instrument piping to vent the accumulator which established the conditions leading to this event. The designed accumulator vent path should have been used.

Post maintenance testing was to be performed on MOV-2865C per 2-PT-212.6, "Valve Inservice Inspection (Accumulator Isolation MOVs)". Unit 2 was in Mode 5 with the RCS drained to approximately 10 inches above vessel nozzle center line. The procedure, however, specified as an initial condition that the reactor coolant system (RCS) be in Mode 3 or 4 with the RCS pressure between 800 and 1000 psig. The initial condition was deviated by the Shift Supervisor without prior SNSOC approval because it was incorrectly judged that the procedure deviation did not change the intent of the procedure. When the test was initiated by cycling MOV-2865C, the "C" accumulator was discharged into the RCS because the accumulator had become pressurized. It was expected that the accumulator pressure would be atmospheric at the time of the test, but the accumulator had become pressurized because the vent path was inadvertently isolated. The most probable cause of pressurization of the accumulator is nitrogen inleakage through the nitrogen isolation valve to the accumulator.

3. CORRECTIVE STEPS WHICH HAVE BEEN TAKEN AND THE RESULTS ACHIEVED:

The Vice President - Nuclear Operations reviewed this event with appropriate station management and operations personnel. Root cause and corrective actions to prevent recurrence were identified.

A Human Performance evaluation was performed with regard to this event and corrective actions identified.

The initial conditions of 1/2-PT-212.6, Valve Inservice Inspection (Accumulator Isolation MOVs), were revised to specify RCS pressure conditions and that venting of an accumulator with inoperable pressure instrumentation is required.

An Operations Memorandum has been issued stating that if the initial conditions of a procedure must be temporarily changed, the procedure deviation must receive prior SNSOC approval. A procedure deviation has been processed for Administrative Procedure 5.8 to require that all deviations to the initial conditions to safety related procedures be approved by the Station Nuclear Safety and Operating Committee prior to their use.

4. CORRECTIVE STEPS WHICH WILL BE TAKEN TO AVOID FURTHER VIOLATIONS:

The event will be reviewed with Operations personnel in Licensed Operator Requalification Program (LORP) to emphasize the importance of proper venting of isolated equipment.

The biennial review of Periodic Test procedures will include a review to ensure that the appropriate initial conditions are included.

The operations standards will be reviewed to identify revisions to the guidance for tagging instructions and practices needed to prevent events of this nature from recurring.

Administrative Procedure 5.8 is presently under review for a comprehensive re-write to enhance the required review process for procedure deviations to ensure compliance with 10 CFR 50.59 and Technical Specifications. This rewrite will include a permanent change to require prior SNSOC approval of deviations to initial conditions in safety related procedures.

5. **THE DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED:**

The review of this event in LORP will be completed by March 31, 1988.

Administrative Procedure 5.3, "Review of Procedures" will be revised by March 31, 1988 to require a review of Periodic Test procedures initial conditions for adequacy. This procedure will also be revised to require the review of procedural steps addressing venting of equipment to ensure that the path designated for venting is the one designed for that purpose.

The operations standards will be reviewed and revisions made, as necessary, by April 1, 1988.

Administrative Procedure 5.8 will be revised by April 1, 1988.