U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No: 50-29/89-16

Docket No: 50-29

Licensee No: DPR-3

Licensee: Yankee Atomic Electric Company
580 Main Street
Bolton, Massachusetts 01740-1398

Facility Name: Yankee Nuclear Power Station

Inspection at: Rowe, Massachusetts

Inspection Conducted: August 1 - September 26, 1989

Inspectors: H. Eichenholz, Senior Resident Inspector
M. Markley, Resident Inspector
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Approved By: A. Randy Blough, Chief, Reactor Projects Section 3A 10-30-89 Date

Inspection Summary: Inspection on August 1 - September 26, 1989 (Report No. 50-29/89-16)

Areas Inspected: Routine inspection on daytime and backshifts by two resident inspectors and two regional specialists of: actions on previous inspection findings; operational safety; security; plant operations; maintenance and surveillance; radiological controls; and, review of event requiring notification of the NRC.

1. General Conclusions on Adequacy, Strength or Weakness in Licensee Programs

The licensee demonstrated a proper safety perspective in conducting the August 17-30 outage to repair the leaking main coolant system bypass line safety valve flange. Operator attention to detail was noteworthy in identifying the leaking bypass line/vent line during plant heatup on August 25. However, operator error during startup on August 29 resulted in a reactor trip similar to one which occurred in 1986. The recurrence of this incident warrants additional management attention (Section 6.4). Additional attention is also warranted to provide improved controls for identifying housekeeping irregularities in the vapor container (Section 4.5). Further, improved guidance to operators for making operability determinations appears warranted. The plant operations review committee (PORC) consistently provided a high level of oversight and review.

2. Unresolved Items

An unresolved item is a matter about which more information is required to ascertain whether it is an acceptable item, a deviation or a violation.

Two unresolved items were identified during this inspection period:

-- Review of licensee actions to provide operator guidance and training for determining operability of plant systems/components (Section 4.6).

-- Review licensee evaluation for adequacy of actions involving radiological personnel contamination incident (Section 8.0).

TABLE OF CONTENTS

Table of Contents

*The NRC Inspection Manual inspection procedure (IP) or temporary instruction (TI) or the Region I temporary instruction (RI TI) that was used as inspection guidance is listed for each applicable report section.

DETAILS

1. Persons Contacted

Yankee Nuclear Power Station

T. Henderson, Plant Superintendent
R. Mellor, Technical Director

Yankee Atomic Electric Company (YAEC)

N. St. Laurent, Manager of Operations

The inspector also interviewed other licensee employees during the inspection, including members of the operations, radiation protection, chemistry, instrument and control, maintenance, reactor engineering, security, training, technical services and general office staffs.

2. Summary of Facility Activities

Yankee Nuclear Power Station (Yankee, YNPS or the plant) operated at 100% rated power until August 4 - 6, 1989, when plant load was reduced to perform condenser leak testing and to replace aged condenser tube plugs. Following the load reduction, the plant continued to operate at full power until August 17-30 when an outage planned for September 1989 was started early to replace the main coolant system (MCS) loop No. 2 bypass line safety valve flange. On August 25, during plant heatup following safety valve repairs, an Unusual Event (UE) was declared and the heatup terminated when a MCS leak was identified on the loop No. 2 bypass line/vent line socket weld. The UE was terminated at 8:25 a.m. the same day when the plant was returned to Mode 5 (cold shutdown). Repairs were effected, the heatup was resumed and the plant entered Mode 2 (startup) on August 29. At 5:38 p.m. the same day, a reactor trip occurred when the control room operator inadvertently turned the main steam non return valve (NRV) Reset/Trip switch to the Trip position. After completing the post-trip review and securing authorization, the plant was restarted on August 29. The plant achieved full power operation on August 30 and continued to operate at 100% capacity through the end of the inspection period.

On September 13, 1989, NRC Commissioner Kenneth Rogers conducted a visit at YNPS. A meeting was held with the resident inspector, and the Plant Superintendent conducted a site tour. Following the tour, discussions were held with site management on NRC and YNPS issues.

During the period of July 31, 1989 to August 4, 1989, NRC Region I (NRC:RI) specialist inspectors completed a routine inspection of the licensee emergency preparedness program (50-29/89-15).

On September 21, 1989, the NRC Systematic Assessment of Licensee Performance (SALP) Board was held to review licensee activities for the period of April 1, 1988 to July 31, 1989 (SALP Report 50-29/88-99).

3. Status of Previous Inspection Findings

3.1 (Closed) Unresolved Item (50-29/87-11-04): This item is related to the Technical Specification (TS) use of the term "associated flow paths" in section 3.7.1.2, Emergency Feedwater System. The limiting condition for operation in this section requires at least two independent emergency feedwater pumps and associated flow paths to be operable. Section 4.b of NRC Inspection Report 50-029/87-11 identified certain licensing issues related to the "associated flow paths" and noted a lack of knowledge by the licensed operators pertaining to this term. The plant organization initially proposed a change to the TS in Service Request No. 87-55 issued to the Yankee Nuclear Service Division (YNSD). Inspector discussions with plant personnel during this inspection indicate that a TS change may eventually be issued based on this Service Request. However, on 6/2/88, the Manager of Operations issued TS Interpretation No. 88-3 which defines "associated flow paths" as follows:

1. The normal flow path via the main feedwater piping in the Turbine Building, and

2. The alternate flow path via the steam generator blowdown piping in the Primary Auxiliary Building.

The inspector interviewed control room operators and operations supervisory personnel to ensure that the definition provided in this interpretation is clearly understood and that the TS limiting condition for operation for the emergency feedwater system requires both the normal and alternate flow paths to be operable. Therefore, based upon review of the TS interpretation of associated flow paths and personnel interviews, this specific item is closed. A new unresolved item was identified by the inspector in conjuction with this review and is discussed in paragraph 4.4.

3.2 (Closed) Unresolved Item (50-29/87-11-05): This item is related to the NRC finding that plant operations personnel had failed to document deficient conditions found during monthly surveillance testing of valve EBF-MOV-557.

During the initial phases of test OP-4211 on July 21, 1987, the valve's motor operator experienced two trips from thermal overload during valve cycling. Maintenance Request (MR) 87-1151 was generated to investigate the malfunction and to repair the valve. Subsequent testing demonstrated that valve EBF-MOV-557 could not be fully cycled within the time limits prescribed by the surveillance procedure. A new torque switch was installed during the repair of this valve. The NRC inspector noted that both of these deficient conditions were encountered during the performance of the surveillance test and that no condition description or maintenance request number had been recorded in the surveillance procedure. The surveillance procedure was completed satisfactorily after the maintenance work was accomplished.

This situation was inadequate documentation of equipment performance problems encountered during TS based surveillance testing which is required to demonstrate system and component operability.

On February 3, 1989 the plant operations department issued memorandum OPS 89-20 and Special Order 89-23. These documents require that all appropriate operations surveillance procedures be revised to add required operator actions in the event that the procedure cannot be satisfactorily completed due to the failure of a system, sub-system, train, component, or device to perform its intended function or due to a lapse of time in excess of the required frequency. These actions specifically include the delineation of the failed surveillance, notification of higher supervision and quality assurance department, review of administrative procedure AP-0008 for possible reporting requirements, and documentation of a maintenance request number, if applicable. The inspector reviewed operations surveillance procedures which were revised to incorporate the requirements for the memorandum and special order. To date, 41 of 66 total procedures have been revised and issued. The remainder are scheduled for revision during the normal biennial review. The licensee stated that all affected procedures are expected to be revised by November 1990. The inspector also noted that all operations and reactor engineering personnel have read and acknowledged these requirements. Licensee actions were adequate and appropriate. This item is closed.

3.3 (Closed) Unresolved Item (50-29/87-11-07): This item refers to an NRC inspector concern that no entry in the control room log was made October 20, 1987, to document that a TS required surveillance test was performed. The quarterly calibration of the vapor container post accident hydrogen analyzer (HV-GA-1) is performed by procedure OP-4623 in accordance with TS surveillance Requirement 4.6.3.1.b. and no indication was provided that this surveillance had been accomplished. Control room log entries provide assurance that important activities, such as TS required surveillance testing, are performed and that they are completed within the required time.

The inspector reviewed procedure AP-2007, Revision 31, and noted that Procedure Section 4., "Operating Log" was amended to require "Whenever a TS required surveillance is begun, the start time shall be recorded in the SS log. When the TS required surveillance is completed, then the completed time shall be also be logged." The log used in the control room during this inspection was reviewed covering an approximate four month period. Control room log entires for surveillance procedures were noted and compared to the operations schedule for their performance. All procedures were performed as scheduled and the necessary log entries were made. Licensee corrective actions were adequate and personnel were responsive to NRC concerns. This item is closed.

3.4 (Closed) Unresolved Item (50-29/87-11-08): This item is associated with an NRC identified concern that performance of I&C procedure OP-4623 did not receive a timely supervisory review. Approximately two days elapsed after the procedure was completed before the I&C department supervisor became aware that the I&C technician performing the procedure had inadvertently left the low range indication for the HV-GA-1 post-accident hydrogen analyzer at a value that exceeded the procedure limit. Late identification of this situation caused the plant operators to delay entry into the technical specification action statement of section 3.6.3.1 as required. The inspector interviewed the I&C department supervisor and determined that the amount of error considered unacceptable in procedure OP-4623 has been made clear in the procedure and to all I&C technicians. He further indicated that I&C technicians were counseled in the need for timely/immediate notification of supervisory personnel upon discovery of out-of-specification conditions. The inspector reviewed approximately 30 I&C surveillance procedures completed after June 1987 and noted that immediate supervisory notifications and reviews had been made and documented as necessary. This item is closed.

3.5 (Closed) Unresolved Item (50-029/88-01-02): This item was in response to a nonconformance with QAD requirements for the control of purchased material, equipment, and services to repair the emergency diesel generators. Licensee dedication of commercial grade components procured from the Power Products Company lacked conformance with administrative procedure AP-0112. The situation arose when the licensee contracted diesel repair services to Power Products after it had been removed from the approved vendors list (AVL), and after post-maintenance test failures were attributed to poorly installed replacement parts. Investigation by the QAD ascertained that Power Products had sufficient internal controls to ensure that only factory authorized replacement parts were installed in the plant's diesels, and that they had provided certificates of compliance to demonstrate that replacement parts were on the approved spare parts list from the diesel vendor. However, the licensee failed to perform receipt inspection of spare parts purchased from Power Products for diesel repairs in 1987. The licensee subsequently performed a receipt inspection of those parts which had been received in the station warehouse and accepted the installed diesel parts based upon satisfactory completion of all post-maintenance testing.

Nonconformance Report (NRC) No. 87-42 was initiated by the QAD to resolve the deficiency and to specify corrective actions. The inspector reviewed the NCR to determine that corrective actions were appropriate and that other actions were taken to prevent recurrence. QC Inspection Report (QCIR) No. 89-18 was also reviewed to ensure that QAD follow-up actions were taken to verify that the corrective actions were properly implemented. The QCIR contains specific inspection items which verify that the NCR corrective actions were properly dispositioned. The QCIR also contains documentation of receipt inspection activities for diesel spare parts; confirmation that the Power Products internal parts control program is adequate; and a report of discussions with maintenance department personnel concerning compliance with AP-0212 procurement and receipt inspection requirements. The inspector further noted that the QAD has placed all emergency diesel generator repair activities on the "Mandatory Inspection List." Corrective actions to prevent recurrence were adequate. This item is closed.

3.6 (Closed) Unresolved Item (50-29/88-02-01): During control room observations for inspection 50-29/88-02, it was noted that the safety parameter display system (SPDS) had failed and its computer was offline for approximately thirty minutes. NRC concerns arose over discussions with control room operators which revealed that there was generally no positive criteria for determining the operability for the "Accident Monitoring Instrumentation" involving the incore thermocouples and reactor head thermocouples. The SPDS computer (twenty-four channels) is the principle display used to monitor this instrumentation from the control room. Backup displays are also available in the control room from four instrument recorders (eight channels) on the front main control board (MCB) and four digital indicators (twenty-four channels) on the back MCB.

TS Section 3.3.3.5 requires that accident monitoring instrumentation channels shall be operable in Modes 1, 2, and 3. It further requires that a minimum of one channel in each of three core quadrants shall be operable, and a total number of eight channels shall be operable. All TS requirements for operability of these instruments can be positively established with any one set of instruments in the control room.

Surveillance Procedure OP-4272, "Accident Monitoring Instrumentation Channel Check," performs a monthly channel check surveillance of the "Primary Display" (SPDS), and the "Secondary Display" (Front MCB). The procedure was not clear what channels or portions of channels were allowed to be out of service and still meet the requirements of TS 3.3.3.5.

The plant operations department issued Service Request NO. 88-72 (8/31/88), requesting that definitive guidelines be provided to the operating staff for use in making operability determinations using the SPDS. The lead I&C engineer response (memo YRP 2236/88) recommended that the four digital indicators on the back MCB be included as part of the backup display, since both primary and backup would contain the same quantity of thermocouples. Thus temporary inoperable status for the SPDS would not need to be addressed under TS Section 3.3.3.5 (providing both displays were positively checked by the surveillance procedure).

The inspector reviewed the revision to surveillance procedure OP-4272. The sections pertaining to the incore and reactor head thermocouples have been substantially revised to clearly designate SPDS as the primary display and the MCB digital instruments as the secondary display. The revision also expressly prohibits using a combination of indications from both displays to attain one operable system. The procedure now exceeds TS Table 3.3-7 requirements for the minimum and total channels required for operability and those instruments which are environmentally qualified.

The licensee also issued procedure OP-4723, "Core Exit and Reactor Head Thermocouple Evaluation." This procedure requires a thorough operability assessment of both primary and secondary displays. All thermocouple data is recorded on a graphic core map which also provides criteria for evaluating the indicated temperature distribution in the core. This procedure is performed, 1) during the weekly reactor engineering department surveillance of core thermocouple indications; 2) when either the primary or secondary displays are declared inoperable; 3) upon request by the plant operations department; and 4) following startup of the SPDS after repair or modification of its source code or data acquisition system.

The above actions resolve operability concerns in this area. This item is closed.

3.7 (Closed) Unresolved Item (50-29/88-05-01): This item concerns a licensee identified surveillance procedure discrepancy and a resulting failure to provide objective evidence that TS surveillance requirements were met to operate the emergency diesel generators (EDGs) for ≥60 minutes while loaded to ≥400KW. The original procedure, OP-4209, required that the diesels be operated for ≥60 minutes while loaded to ≥400KW or ≥500KVA. The procedure also provided that the KVA rating could be substituted for 400KW if the system power factor limits the KW loading to ≥400KW. Thus, the procedure would not ensure satisfaction of the TS section 4.8.1.1.2.d.4 surveillance requirement.

The inspector reviewed revision 21 of procedure OP-4209 and LER 50-29/88-01. Appropriate corrective actions were taken to revise the procedure and to ensure that the EDGs are tested at ≥400KW and

>500KVA. Discussion Section 3, and Procedure Sections 9.f and 10. of OP-4209 all require that the EDG load test be performed for at least 60 minutes at 400 - 420 KW and 500 - 525 KVA. This item is closed.

3.8 (Closed) Inspector Follow Item (50-29/88-05-05): This item refers to the licensee's response to NRC Bulletin No. 87-02, "Fastener Testing to Determine Conformance with Applicable Material Specifications," dated November 6, 1987. At the time of inspection 88-05, the licensee's program in response to the Bulletin was found adequate, but the licensee had not yet revised procedure AP-0212, "Control of Purchased Material, Equipment, and Services," to formalize requirements for random sampling and testing of safety related fasteners to ensure compliance with applicable material specifications.

The inspector reviewed the current version of procedure AP-0212 (Rev. 16) and noted that Requirements Section A.5. and Implementation Section A.2 were revised to state that safety related threaded fasteners will be subjected to hardness testing on a sample basis to verify compliance with the material specification. These sections further specify the fastener sample number per Material Heat number and provide disposition instructions for any out-of-specification fasteners. This item is closed.

3.9 (Closed) Violation (50-29/88-22-02): NRC inspection 50-29/88-22 identified that the licensee was in violation of 10 CFR 50, Appendix B, Criterion XVI, in that "the licensee failed to establish effective measures, in the form of clear procedures to translate corrective action requirements for the proper documentation, and reporting to appropriate levels of management of significant conditions adverse to quality involving design deficiencies, in that, existing administrative controls did not provide: (1) that the QAD be responsible for review of recommendations to prevent recurrence, and (2) that engineering and/or project departments be responsible for review and determination of cause, and provide recommendations for corrective action to preclude repetition." Provisions for establishment of these effective measures were contained in the Yankee Atomic Electric Company Operational Quality Assurance Manual, YOQAP-1-A, Section XVI.

Licensee response letter BYR 89-62, dated March 31, 1989, indicated that the current YNSD Procedure No. WE-001, Administration of the Engineering Manual, directs that reporting of deficiencies or nonconformances to management shall be in accordance with Procedure No, WE-005, "Standard Memorandum." The response further noted that although these procedures do address corrective actions, they do not clearly require the same review of the QA corrective action recommendations required by YOQAP-1-A, Section XVI.

The licensee response letter further prescribes corrective actions taken as follows:

-- The Engineering Design Change (EDC) corrected the original engineering deficiency responsible for the condition adverse to quality.

-- The licensee evaluated the original engineering deficiency and corrective actions were specified to prevent recurrence.

-- Procedure No. WE-109, "Engineering Deficiency Reports," (EDRs) was developed to formalize reporting of engineering deficiencies. The YNSD staff will be trained in processing EDRs.

-- A major revision was initiated to Procedure No. WE-10?, "Engineering Design Change Request."

-- A review of revisions to design changes occurring over the past two years will be performed to determine if engineering deficiencies are responsible for the revisions.

The licensee response letter also prescribed actions to preclude recurrence as follows;

-- The new procedure WE-109 will assure that engineering deficiencies are identified, evaluated, and corrective actions taken to preclude recurrence.

-- Training on WE-109 will be provided to NSD personnel.

The inspector reviewed the licensee's actions identified above and evaluated documentation provided to demonstrate that these actions were adequate and were completed.

The inspector verified that all licensee identified corrective actions for this violation were satisfactorily implemented. This item is closed.

4. Operational Safety

4.1 Plant Operations Review

The inspector observed plant operations during regular and backshift tours of the following areas:

Control Room	Safe Shutdown System Building
Primary Auxiliary Building	Fence Line (Protected Area)
Diesel Generator Rooms	Intake Structure
Vital Switchgear Room	Turbine Building
Cable Tray House	Spent Fuel Pit (SFP) Building
Vapor Container (VC)	

Inspections of the control room were performed on weekends and backshifts as follows: August 2, 4, 10, 12, 13, 18, 19, 20, 23, 26, 27, 28, 29, 30, 31 and September 21, 22, and 25. Deep backshift included: August 25 from 3:00 a.m. to 5:00 a.m. and August 20-31 from 10:00 p.m. to 12:30 a.m.

Operators were alert, attentive, and responded appropriately to annunciators and plant conditions.

Control room instruments were observed for correlation between channels, proper functioning, and conformance with technical specifications. Alarm conditions in effect and alarms received in the control room were reviewed and discussed with the operators. Operator awareness and response to these conditions were reviewed. Control room and shift manning were compared with Technical Specification requirements. Posting and control of radiation, contaminated and high radiation areas were inspected. The use of and compliance with Radiation Work Permits (RWPs) and use of required personnel monitoring devices were checked. Plant housekeeping controls were observed including control of flammable and other hazardous materials. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout and temporary change request logs, and event reportability evaluation requests. Inspections of the control room were performed on weekends and backshifts. Operators and shift supervisors were alert, attentive and responded appropriately to annunciators and plant conditions.

TS and procedure requirements for mode changes were adequately dispositioned prior to changing modes. Operators were continuously aware of ongoing maintenance and surveillance activities. Operators were effective in responding to main control board (MCB) annunciator panalarms. No malfunctioning or nuisance alarms were observed. Operations management was observed to have a strong awareness of control room activities.

One noted exception to the normally excellent control room decorum was observed by the inspector during startup on August 29. The startup was conducted during reactor operator (RO) shift turnover and the beginning of day shift activities. The control room was uncharacteristically crowded and busy. Station and contractor personnel were observed to linger in the operating shift work area casually observing the startup activities. Although access could have been better managed, no degradation in the quality of operator performance was noted. The shift turnover was thorough and of high quality. The operating shift maintained positive control of personnel access to the MCB panels. The operating shift was ultimately effective in restoring proper control room decorum.

Strength was noted in the reactor engineering (RE) and trainee support interface with operations. Reactivity startup data was provided from a location which afforded RE observation of the MCB and promoted effective communications.

Operations management was responsive to inspector concerns. Personnel in other station departments now solicit permission to enter the operating shift area. The inspector observed continued improvement over previous access control practices.

Documentation of shift activities was good. Inspector review of operating logs and turnover sheets indicated good characterization of operating history. Off-normal conditions, surveillances completed, and equipment performance were appropriately documented.

4.2 Safety System Review

The emergency diesel generators, EDG fuel oil, containment isolation and high and low pressure safety injection systems were reviewed to verify proper alignment and operational status in the standby mode. The review included verification that: (i) accessible major flow path valves were correctly positioned; (ii) power supplies were energized, (iii) lubrication and component cooling was proper; and (iv) components were operable based on a visual inspection of equipment for leakage and general conditions. System walkdowns to assess the material condition of the HPSI and LPSI ECCS and the accumulator were performed. Selected accessible valves were verified to be in the correct position and locked when required by plant procedures.

The condition of those system components inspected was found to be good. Leakage from system piping and flanged joints was not observed. No unacceptable conditions were identified regarding ECCS pump lubrication. Local instrumentation was verified to be operational by channel checks with remote indication. No conditions adverse to safety were identified during inspection of this ECCS equipment. No violations were identified.

4.3 Review of Temporary Changes, Switching and Tagging

Temporary change requests (TCRs), which were approved in support of implementing lifted leads and jumper requests and mechanical bypasses, were reviewed to verify that: controls established by AP 0018, "Temporary Change Control," were met; no conflict with the Technical Specifications were created; the requests were properly approved prior to installation; and a safety evaluation in accordance with 10 CFR 50.59 was prepared if required. Implementation of the requests was reviewed on a sampling basis.

The switching and tagging log was reviewed and tagging activities were inspected to verify plant equipment was controlled in accordance with the requirements of AP 0017, "Switching and Tagging of Plant Equipment."

Licensee administrative control of off-normal system configurations by the use of TCR and switching and tagging procedures as reviewed above, was in compliance with procedural instructions and was consistent with plant safety. No unacceptable conditions were identified.

4.4 Operational Safety Findings

During prior NRC inspections and the review of licensee actions that were implemented to close Unresolved Item 50-29/87-11-04 (Section 3.1 of this inspection report), the inspector found that operator response to malfunctioning equipment in the emergency feedwater systems had not been clearly established with respect to system operability requirements for the alternate flow path. A clear and consistent understanding of the system conditions required for operability was lacking. Specifically, operators demonstrated differing levels of understanding of what constituted operability.

Prior NRC observations and findings that relate to this issue were:

-- In NRC inspection 50-29/86-09, it was identified that installation of an undersized trip coil in the circuit breaker on the power supply of the EBF-MOV-557 valve precluded the valve from operating in accordance with its intended design objectives. The EBF-MOV-557 valve, and hence the emergency feedwater alternate flow path, was determined by the NRC to be inoperable. This condition constituted a violation of TS 3.7.1.2. Licensee response letter FYR 86-088, dated September 18, 1986, detailed corrective actions which centered principally upon the engineering and material control issues which led to the installation of the undersized trip coils in the valve. The response did not address the condition of inoperability of the valve or the alternate flow path which occurred from a loss of the valve's motor operator feature.

-- During the conduct of NRC inspection 50-29/87-11, it was identified that valve EBF-MOV-557 had tripped twice on its thermal overload when exercised during the monthly operability surveillance test OP-4211. The valve was also found to be incapable of cycling within the time period required by the test procedure. No entry was made in the control room log as to the status of operability of the system. The valve was subsequently removed from service for maintenance and the control room log entry noted that the alternate flow path was manually operable per TS LCO 3.7.1.2. Control room operators decided that there was not a condition of inoperability in the emergency feedwater system and their decision was supported by the assistant operations manager. NRC concerns were then expressed to the plant operations manager who agreed that the system should be declared inoperable. That action was subsequently taken by the plant operators and a TS action statement was entered based upon the inoperability of valve EBF-MOV-557. The valve was then repaired within the required time period and the system was returned to normal service. NRC inspection report 50-29/87-11 determined that the failure of this valve to perform its intended design function using the motor operated feature from the control room constituted an inoperability in the alternate flow path.

Within the scope of this review the inspector interviewed the plant assistant operations manager, two shift supervisors, and two senior control room operators concerning the required conditions for operability in valve EBF-MOV-557 and the emergency feedwater alternate flow path. No clear or consistent description of required operator actions was provided by these individuals when asked if the alternate flow path should be declared inoperable upon loss of remote (control room) control of valve EBF-MOV-557. Some operators stated that TS action statement would not have to be entered because they considered that local manual control of the valve constitutes the basic condition required for operability and that the intended design function of this valve could still be met. Other operators were unsure of the required operator response but stated that they would not declare the alternate flow path inoperable if they still had manual valve control.

The inspector further determined that the operators were not provided clear and and consistent guidance on criteria to be used for operability determinations for plant systems. Inspector review noted the current practice among operators was to allow for broad flexibility of opinion which depends upon individual operators knowledge of intended system design functions. Based on these discussions, the inspector was concerned that the original system design bases may not be examined for plant systems for determining operability as defined by TS.

Based upon the two previous NRC inspections noted above, and the positions taken by senior licensee operations management, with respect to operability in the emergency feedwater system, the licensee needs to resolve the inconsistent approach to the determination of system operability currently taken by plant operations personnel. This matter can have broader ramifications throughout plant systems and the basis upon which operators would declare them operable or inoperable. The licensee is reviewing the inspector concerns for the need to provide additional guidance and training to aid operability determinations relative to TS requirements. This item is unresolved pending completion of the licensee evaluation and review by the NRC (50-29/89-16-01).

Licensee attention to this matter is warranted to ensure that plant operators have the proper procedural support and guidance to perform their job.

4.5 Vapor Container Housekeeping

Vapor container (VC) housekeeping for the shutdown of August 17 - 30 was generally good. However, during a VC tour on August 21, the inspector noted chains attached to hangers CRB-SH-16 and CRB-SH- 17 which appeared to be providing additional support to the pressurizer spray line. The inspector also observed wiring tied to the loop No. 3 wide range steam generator level instrumentation piping. The wire did not appear to be performing a function.

Subsequent licensee evaluation determined the chains were installed to support the spray lines to accommodate radiological shielding during the 1987 refueling outage. After shielding removal, the independent verification failed to identify that the temporary chains had not been removed. Examination by the maintenance support department (MSD) determined that the chains were not supporting the weight of the spray line. YNSD conducted an engineering evaluation which concluded that the chains had no adverse effect during operation. The RP department conducted a review and walkdown to verify that all temporary shielding and accessories had been removed. No additional anomalous conditions were identified. The shielding procedure was also revised to ensure shielding support equipment was similarly removed.

The licensee was unable to explain why the wire had been attached to the steam generator level instrumentation piping. It was promptly removed.

Inspector review determined the licensee corrective actions to be adequate except that stronger programmatic controls need to exist to identify irregularities of this type. Licensee VC walkdowns failed to identify the chains prior to setting containment integrity following two refueling outages and prior to identification by the NRC. The wire existed on the level instrumentation piping for an undetermined period of time.

Although the shielding removal independent restoration verification was poor, the shielding package was complete with a proper loading analysis performed and documented. MSD and YNSD evaluations were technically sound. The licensee is being responsive to the inspector concerns by indicating that they intend to make program and procedure changes in this area.

5. Security

5.1 Observations of Physical Security

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the security plan and approved procedures. This review included the following security measures: guard staffing, vital and protected area barrier integrity, maintenance of isolation zones, and implementation of access controls including authorization, badging, escorting, and searches. No inadequacies were identified.

5.2 Peaceful Citizens Group Demonstration

At 8:21 a.m. on August 7, 1989, six individuals from a citizen group entered the owner controlled property to conduct a non-violent demonstration. The demonstration consisted of peaceful actions at the door to the gatehouse and the main vehicle entrance gate to the protected area.

The licensee requested assistance from the state law enforcement agency (SLEA) and requested that the demonstrators leave the owner controlled area. Following the arrival of SLEA personnel, the demonstrators were advised that they were in violation of trespassing laws and that they would be arrested if they failed to voluntarily exit the area. Three of the six individuals were subsequently arrested.

The licensee response to this event was conservative and appropriate. Effective coordination with the SLEA was demonstrated. The inspector had no further questions.

5.3 Replacement of Security Equipment

During this inspection period, the licensee continued to upgrade security equipment to improve performance and improve program effectiveness. The inspector verified that proper compensatory measures were employed during the upgrades. Other changes in equipment were identified and corrected through the station design change process. Licensee performance in this area was noteworthy.

6. Plant Operations

6.1 Load Reduction to Leak Test and Replace Condenser Tube Plugs

On August 4-6, 1989, the licensee reduced plant load to approximately 80% rated power to replace the remaining 20% of the aging condenser tube plugs which were not replaced during the July 7-8, 1989, load reduction. Work was completed without incident and the plant was returned to full load at 8:55 a.m. on August 6.

6.2 Plant Shutdown to Replace Bypass Line Safety Valve Flange

On August 17, 1989, the licensee commenced a plant shutdown to perform repairs on the inlet flange of the loop No. 2 main coolant bypass line safety valve (MC-SV-201B) flange. The leak was identified following completion of the core XX refueling outage which ended in January 1989. The licensee assessment determined the leak to be minor in quantity and observed it to be in the form of a steam wisp. Initial monitoring was done from the control room using a camera focused on the flange in the loop room. The camera became inoperable after several months and the licensee performed visual inspections to routinely monitor the steam wisp. During the biweekly containment inspection on August 15, the licensee noted the steam wisp to have disappeared indicating that a possible cell of boric acid crystals may have encapsulated the leaking flange. On August 17, plant load was reduced to 20% rated power to perform a more thorough examination. Boron was removed and the licensee observed boric acid corrosion on two of the four flange fasteners. Following engineering analysis, the licensee decided to shutdown the plant and repair the flange leak.

The licensee completed repairs which involved replacing the flange. A more detailed description of maintenance activities is provided a Section 7.1 of this report. Plant heatup and return to power operation is described in Section 6.3 this report.

Inspector review noted licensee actions to be conservative with regard to nuclear safety. The licensee had previously scheduled a brief outage for September 1989 to repair the safety valve flange. The determination to begin the outage early was based on concerns for reduced margins of safety due to boric acid corrosion and the resultant degraded fastener integrity.

Preliminary licensee measurement and assessment of fastener degradation indicated that main coolant system (MCS) integrity remained acceptable, in that the as-found conditions were determined to meet ASME design requirements and TS requirement for structural integrity.

Inspector review determined the licensee operability and integrity analysis to be technically sound. The licensee demonstrated a good safety perspective in monitoring the valve and implementing corrective measures when degradation was observed. Plant leak rates were maintained within required limits. Operability testing was verified to meet ASME requirements prior to commencement of plant heatup.

6.3 Main Coolant System By-Pass Line Leakage/Declaration of an Unusual Event

During plant heatup on August 25, 1989 at 12:30 a.m., following shutdown to repair the leaking safety valve flange (MC-SV-201B), a reactor operator who was performing a containment inspection identified leakage from insulation around the 5-inch diameter loop No. 2 bypass line. At the time, the plant was in Mode 4 (Hot Shutdown) with MCS pressure and temperature at 290 psig and 304 degrees Fahrenheit, respectively. The leakage was noted to be in the vicinity of the 3/4-inch high point vent line connection. The insulation was removed and the licensee discovered what appeared to be MCS pressure boundary leakage from a defect in the socket weld connection between the vent and bypass lines. The licensee commenced a plant shutdown and declared an Unusual Event (UE) at 1:15 a.m. The States of Vermont and Massachusetts were notified at 1:24 a.m. and 1:26 a.m., respectively. The NRC was notified via the Emergency Notification System at 1:56 a.m. The NRC resident inspector was similarly notified and responded to the site to follow the event.

Based upon extensive discussions by the operations staff, the licensee concluded that the identified leak could be isolated by securing the MCS loop stop valves, and therefore did not initially classify the condition as pressure boundary leakage. A subsequent review at 1:05 a.m. by the Manager of Operations with the Plant Operations Manager and shift operations personnel determined that the conditions should be considered as pressure boundary leakage, which warranted under the licensee's Emergency Action Level Criteria that a UE be declared.

The UE was terminated at 8:25 a.m. on August 25, when the plant achieved Mode 5 (cold shutdown). The vent line was repaired and heatup was initiated on August 28. Specifics of the repairs and failure analysis are discussed in Section 7.2 of this report.

Inspector review noted good operator action in responding to the identification of the leakage. The licensee decision to terminate the heatup was prompt. The cooldown was conducted in a well controlled manner. Reporting of the UE was conservative and licensee management demonstrated a conservative safety orientation by not relying on the main coolant loop isolation valves.

6.4 Inadvertent Reactor Trip During Startup

While in Mode 2 (Startup) on August 29 at 5:38 p.m., the reactor was inadvertently scrammed when the control room operator improperly placed the Train B main steam line nonreturn valve (NRV) trip circuit Reset/Trip switch in the trip position. All systems performed as designed following the trip signal. The event occurred due to personnel error by the operator. An event evaluation was promptly performed and concurrence to restart was provided such that the plant was returned to Mode 2 at 10:30 p.m.

The licensee event reportability evaluation report (ERER 89-55) indicated that the personnel error was caused, in part, due to inadequate procedure guidance. Specifically, Attachment A of procedure

OP-2256, "Operation of the Main Steam System," directed the individual to place the main steam circuitry in "Block" and verify that the panalarm actuates. Review by the licensee determined that the panalarm would not actuate when the Block switch was placed in the Block position at MCS pressures greater than 1800 psig. Pressure at the time of the incident was 2000 psig. The operator had solicited consultation with the shift supervisor and operations manager. When the operator attempted to repeat the procedure, the switch was placed in the wrong position thereby tripping the reactor. Another root cause was identified as the design, which placed the trip and reset functions on the same switch.

Corrective actions to prevent recurrence detailed by the licensee included revising procedure OP-2256 to remove the requirement to place Trains A and B in the block position and to include a caution statement against placing the Trip/Reset switches in the trip position. An operator aid was placed at the NRV control panel to caution personnel accordingly. Also, an evaluation of the control panel switch design was initiated to determine if human factors engineering is appropriate. This was a repeat incident as described in LER 86-13 and NRC inspection 50-29/86-08.

Inspector review noted that operations shift personnel demonstrated a proper safety perspective in responding to the transient. There was no lack of supervisory oversight in that the control room operator erred in the presence of the shift supervisor and operations manager. Although the licensee revised the procedure and placed an operator aid on the control panel, the detailed corrective actions for ergonomic considerations were weak in that the stipulated actions were vague in stating whether or not a human factors engineering evaluation would be performed.

Recurrence of this event indicates a lack of effectiveness for corrective actions taken in response to the 1986 reactor trip. Similarly, direct supervisory oversight was not effective in highlighting potential human factors considerations to the operator. Review of corrective actions to prevent recurrence warrants additional management attention.

6.5 Loss of the Non-Essential Uninterruptible Power Supply (NEUPS)

At 11:12 a.m. on September 7, 1989, plant operators responded to a "MG Set Panel Bus Power Failure" annunciator and found that the NEUPS had tripped. This condition impacted on the following functions: (1) instrumentation for some of the secondary system parameters; (2) process radiation monitoring equipment; (3) commercial telephone switchboard and ENS system; and (4) security features. The NEUPS was restored to operable status at 11:25 a.m. following initial investigation and resetting of three circuit breakers that had tripped. The inspector verified that appropriate compensatory measures were implemented in a timely manner by the security organization in response to the loss of security features associated with the tripping of the NEUPS. No primary or secondary system plant problems resulted. Timely operator actions returned the equipment to service.

The initial licensee investigation found no obvious reason or equipment condition for the trip of the NEUPS. Because the failure occurred approximately 30 seconds following the start of the No. 2 low pressure safety injection (LPSI) pump for a routine surveillance test, the licensee concentrated their attention in this area. As part of the licensee's followup investigation to determine the root cause of the equipment trip, the manufacturer of the NEUPS (Excide Co.) sent a service representative on September 14, 1989 to the site to review NEUPS performance during starting of the No. 2 LPSI pump. No abnormal equipment performance was noted at that time. Additional testing of the battery supply for the NEUPS is planned. This may resolve one of the possible root causes associated with NEUPS trip. One possible scenario was that an electrical transient occurred on the 480 VAC input source to the NEUPS, and that the system's battery was unable to support the NEUPS load when the unit attempted to transfer from a-c to d-c inputs.

The inspector determined that the licensee is appropriately evaluating and investigating this problem. The inspector had no further questions on this item.

7. Maintenance/Surveillance

7.1 MCS Bypass Line Safety Valve Flange Replacement

On August 17, 1989, the licensee commenced a plant shutdown to perform repairs on MCS loop No. 2 bypass line safety valve (MC-SV-201 B) inlet flange. The flange had been leaking since the startup from the core XX refueling outage. An increase in leakage had been observed in July and was characterized as a steam wisp. Visual inspection on August 15 noted the steam to have disappeared and the valve to be encrusted in a boron cell. Following examination on August 17, it was determined the flange fasteners were corroded from boric acid exposure.

The licensee disassembled the leaking flange and determined the root cause of the leak to be upset metal surfaces on the tongue and groove section of the flange. The licensee considered the most likely cause to be improper alignment of the flange during assembly. Licensee review of maintenance history records indicated that the flange had not been worked since original installation in 1974.

Repairs involved replacement-in-kind of the valve flange and carbon steel fasteners. The licensee evaluation of boric acid corrosion rates determined the materials to be acceptable for continued use.

Inspector review verified replacement materials and testing meet ASME and ASTM specifications. Programmatic controls for evaluating MCS leakage and corrosion were found to be adequate. Review by an NRC Region I metallurgy specialist indicated that the corrosion rates, although rapid, were not abnormal in comparison to industry experience. Work packages were technically sound and provided adequate detail for effective implementation. Inspector observation of work activities noted personnel to be conscientious and the work to be of high quality.

7.2 MCS Bypass Line Vent Valve Repairs

During plant heatup on August 25, 1989, a licensee operator identified MCS leakage from the loop No. 2 bypass line high-point vent valve. It was later determined to be a defect in the socket weld connection between the vent valve and bypass lines.

The licensee removed the 3/4 inch vent line at the socket weld joint below the defect area. Preliminary assessment by the YNSD metallurgy specialist determined the apparent failure mechanism to be fatigue. He noted the defect (crack) to follow the fillet toe/tube interface for approximately one inch. He indicated that there was no obvious evidence of secondary cracking. Although he observed some surface pits, no visible evidence of cracking was associated with the pitting. His examination was performed using a 10X magnifier. The licensee plans to have a more detailed failure analysis done in an appropriate hot lab.

Repairs involved welding the vent line to the sockolet. The length of the vent pipe was reduced from 21 inches to 7 inches to minimize pipe moment arm and to limit stress from torquing the vent valve discharge pipe cap. A safety analysis which evaluated this change and associated pipe/weld stresses was completed.

The licensee examined the other loop vent value welds using dye penetrant examination techniques to determine if similar failures had occurred. No indications of weld defects were identified.

Inspector review noted strong YNSD technical and engineering support. The preliminary failure analysis and safety analysis for repairs were technically sound and conservative. The licensee was prudent in examining other vent valves for similar failures. The plant operations review committee (PORC) demonstrated a proper safety perspective in evaluating the safety analysis and concurrence to restart. No concerns for restart were identified by NRC:RI review of as-found conditions and licensee corrective actions.

7.3 Reactor Missile Shield Bolting

During a routine inspection of the vapor container, the inspector found that the reactor missile shield had several missing nuts and bent anchor bolts. This condition was identified to the licensee. The licensee stated they were cognizant of the condition and had previously committed to upgrade the missile shield during a subsequent refueling outage as part of the seismic upgrades being performed in accordance with the Systematic Evaluation Program. They stated that the missile shield was not designed to be a seismic structure at this facility. The inspector found the committed action to upgrade the missile shield appropriate and had no further questions concerning this issue.

8. Radiological Controls

8.1 Observation of Radiological Protection and Controls

Radiological controls were reviewed on a routine basis relative to industry radiological standards, administration and radiological control procedures, and regulatory requirements. Selected work evolutions were observed to determine the adequacy of program implementation commensurate with the radiological hazards and importance to safety. Independent surveys were performed by the inspector to verify the adequacy of radiological controls and instructions to workers.

Radiation Protection (RP) staffing for the August 17-30 unanticipated maintenance outage was weak. Although the overall quality of inplant radiological control of work activities was good, staffing levels were marginal and work-limiting. In order to provide an adequate level of vapor container RP work support; the licensee utilized training, decontamination, and shift technical advisor (STA) personnel to staff radiological access control positions. Because the outage occurred ahead of schedule, no contractor RP support was obtained.

Radiological controls for work activities in the VC was good. The inspector observed effective program implementation. Personnel were observed to conduct activities in a knowledgeable and safe manner. RP personnel provided instructions to workers on an ongoing basis and maintained consistent oversight and control.

8.2 Personnel Radiological Contamination Incident

Radiological controls for maintenance performed on safety valve MC-SV-201B in the hot machine shop on August 20 were poor. Several personnel were contaminated, including the cognizant RP technician. The hot work maintenance shop became contaminated as did areas leading to the radiological control area (RCA) access control point. The licensee was aggressive and effective in controlling the incident upon identification. However, the licensee radiological occurrence report (ROR) and subsequent YSND evaluation was unclear in resolving program implementation and personnel performance issues. Inspector review noted specific root cause conclusions to be lacking regarding the adequacy of the prework radiological survey, undocumented radiological surveys, the adequacy of protective clothing practices, and adherence to procedural guidance for RWPs and ALARA briefings. Discussions with senior station management indicated that these concerns were still being evaluated to delineate comprehensive corrective actions to prevent recurrence. This item is unresolved pending completion of the licensee evaluation and review by a NRC:RI specialist inspector (89-16-02).

9. Review of Events Requiring Notification of the NRC

The circumstances surrounding the following events, which required NRC notification via the dedicated ENS line, were reviewed. A summary of the inspector's review findings follows or is documented elsewhere as noted below:

9.1 At 8:01 a.m. on August 1, 1989, the NRC was notified in accordance with 10 CFR 50.72(b)(1)(v) that at 7:28 a.m. a portion of the Nuclear Alert System was removed from service for scheduled maintenance for an eight hour interval.

9.2 At 9:19 a.m. on August 7, 1989, the NRC was notified in accordance with 10 CFR 50.72(b)(2)(vi) that a peaceful citizens group demonstration occurred at 8:33 a.m., and that the state law enforcement agency was requested to respond to the site. This event is discussed further in Section 5.2.

9.3 A notification to the NRC in accordance with 10 CFR 73.71(b)(1) was made at 2:17 p.m. on August 7, 1989 that an event occurred in the period of July 24 - July 28, 1989. The event involved improperly authorized access of one individual to protected areas, vital areas, and Safeguards Information. The follow-up 30-day Physical Security Event Report (50-29/89-S03) was transmitted to the NRC on September 1, 1989. This event will be reviewed by NRC:RI Physical Security specialists at a later date.

9.4 At 3:50 p.m. in August 8, 1989, the NRC was notified in accordance with 10 CFR 50.72(b)(1)(v) that the radio (KNRY-400) in the plant radio paging system was inoperable resulting in a major loss of off site communications capability. The system was returned to service at 10:30 p.m. following the repair of damaged equipment located at the site of the transmitter and antenna. The damaged equipment appeared to be caused by an electrical storm. In accordance with procedure AP-0711, Rev. 3, "Communications Systems," a Communications Problem Report was initiated to document and track this condition. Recent revisions to procedure AP-0711 were initiated by the licensee to enhance the nature and amount of information available to shift personnel about plant communications equipment and the expected response to events involving communications equipment outages. The action is indicative of the importance that the licensee places on maintaining an effective Emergency Preparedness Program.

9.5 In accordance with 10 CFR 50.72(a)(3), the licensee declared an Unusual Event at 1:15 a.m. on August 25, 1989. This declaration resulted from MCS leakage identified as pressure boundary leakage which was coming from piping associated with the bypass line of the No. 2 main coolant loop. This event is discussed further in Sections 6.3, 6.6, and 7.2 of this report.

9.6 At 6:30 p.m. on August 29, 1989 the NRC was notified in accordance with 10 CFR 50.72(b)(2)(ii) of an automatic reactor scram that resulted from an operator inadvertently initiating a trip signal from the Non-Return Valve's trip logic. This event is discussed in Section 6.4 of this report.

9.7 At 12:11 p.m. on September 7, 1989, the NRC was notified in accordance with 10 CFR 50.72(b)(1)(v) that the ENS Station was inoperable as a result of the loss of the non-essential uninterruptible power supply. Details of this event are contained in Section 6.5 of this report.

9.8 At 5:28 p.m. on September 11, 1989, the NRC was notified in accordance with 10 CFR 50.72(b)(1)(v) that two (KTI-390 & KNBY 400) of the plant two-way radio systems were found to be inoperable. The KNBY-400 radio system is part of the plant emergency paging system. Its inoperability constitutes a major loss of communication and offsite response capability. The inoperability was caused by the failure of licensee contractor personnel responsible for system maintenance to properly return the equipment to operable status at the completion of work on the system earlier in the day. The radio systems were returned to service 7:40 p.m. the same day. The inspector verified that the licensee determined the root cause of the event and instituted appropriate corrective measures to preclude recurrence.

No inadequacies were identified regarding these event notifications.

10. Management Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss the findings. A summary of findings for the report period was also discussed at the conclusion of the inspection and prior to report issuance. No proprietary information was identified as being included in the report.