DCS Nos. 50029 820624 820924 821012
820902 821007 821013
820913 821009 821027

U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-29/82-12

Docket No. 50-29

License No. DPR-3 Priority ________ Category C

Licensee: Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Facility Name: Yankee Nuclear Power Station

Inspection at: Rowe, Massachusetts

Inspection conducted: October 1 - November 15, 1982

Inspectors:	Samuel Collins S. J. Collins, Senior Resident Inspector	12/8/82 date signed
	Samuel Collins, for H. Eichenholz, Resident Inspector, Pilgrim NPS	12/8/82 date signed
Approved by:	for T. C. Elsasser R. M. Gallo, Chief, Reactor Projects Section 1A	12/21/82 date signed

Inspection Summary:

Inspection on: October 1 - November 15, 1982
Areas Inspected: Routine, onsite regular and backshift inspection by the inspectors (182 hours). Areas inspected included previous inspection items; reviews of plant refueling outage operations; plant surveillance observation; inspector review of events; review of Licensee Events Reports; Plant Information Report reviews; and Part 21 followup actions.

Results: No violations were identified by the inspectors.

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DETAILS

1. Persons Contacted

Plant Operations

*H. Autio, Plant Superintendent
E. Begiebing, Maintenance Supervisor
W. Billings, Chemistry Manager
*R. Boutwell, Technical Services Supervisor
E. Chatfield, Training Manager
R. Dobosz, Storekeeper
*B. Drawbridge, Technical Director
D. Fogarty, I & C Engineer
*L. French, Plant Engineer
T. Henderson, Reactor Engineering Manager
*P. Laird, Plant Maintenance Manager
*R. Mitchell, Instrument and Control Supervisor
R. Sedgwick, Security Supervisor
*N. St. Laurent, Assistant Plant Superintendent
J. Trego, Radiation Protection Manager
D. Vassar, Plant Operations Manager
B. Wood, Manager - Administrative Services

The inspector also interviewed other licensee employees during the inspection, including members of the Operations, Health Physics, Instrument and Control, Maintenance, Reactor Engineering, Security and General Office Staffs.

Quality Assurance

*L. Reed, Operational Quality Assurance Coordinater
J. O'Neill, Operational Quality Assurance

Yankee Atomic Electric Company

L. Heider, Vice President of Operations
J. Kay, Senior Engineer-Licensing

*Denotes those present at exit interview on November 12, 1982.

2. Licensee Action on Previous Inspection Findings

a. (closed) Violation (50-29/81-05-01): Concrete Removal During R.V. Support Structure Upgrade Was Not Performed Under Quality Control Procedures. The inspector reviewed IR 50-29/81-05-01, Appendix A, dated May 14, 1981 and the licensee's response contained in letter FVY 81-90, dated June 8, 1981 which detailed corrective actions taken. Actions taken included an evaluation of reinforcing steel and repairs to indications. This item is closed.

b. (closed) Violation (50-29/81-19-01): Procedures not used for Calibration of Safety Related Instruments. The inspector reviewed IR 50-29/81-19, Appendix A, dated December 23, 1981 which identified the violation and the licensee's proposed corrective actions contained in letter FYR 82-08, of January 25, 1982. The licensee has issued the following procedures to address the inspectors concerns:

-- OP-6852, Calibration of the Fuel Oil Storage Tank Level Channel FO-L-1

-- OP-6851, Diesel Driven Fire Pump Fuel Tank Level Indicator FS-LI-2, Calibration

-- OP-6850, Fire Water Tank Level Channel (FS-L-1) Calibration

-- OP-6216, Calibration of Safety Related Pressure Indicators

This item is closed.

c. (closed) Unresolved Item (50-29/81-19-02): Logs of T.S. Items Require Procedural Approval. AP-2011, Operations Department Log Control, Rev. 0, was reviewed by the inspector. The procedure notes that ANSI N18.7-1976, Paragraph 5.3.4.4 requires that log sheets used to monitor safety related parameters must be controlled by use of a procedure. AP-2011 lists the applicable log sheets which contain safety related parameters and others which will be controlled by procedure, and requires that all safety related log sheets will have revisions reviewed by PORC. The inspector verified on October 7, 1982 that the Rowe Station Shutdown Log sheet and Rowe Station Log No. 1 in use by the Control Room Operator were the current revision as listed in AP-2011. This item is closed.

d. (closed) Inspector Follow Item (50-29/81-21-05): The licensee had committed to replace, if required, the Main Coolant Flow under/overcurrent trip relays prior to startup from the current refueling outage. As a result of the licensee's investigation of the existing design and the recent failures encountered with Westinghouse Dual SC-1 Relays, Engineering Design Change Request (EDCR) 82-16, Replacement of Main Coolant Under/Overcurrent Trip System Relays, was generated.

This EDCR replaces the above noted relays with Brown Boveri Electric ITE-50H, solid state, high dropout instantaneous relays. A total of 32 relays will be replaced. The inspector determined that the EDCR was generated, reviewed, and approved for this relay modification in accordance with the licensee's Technical Specifications and established procedural controls.

In addition to the above, the inspector reviewed implementing procedure OP-6000.158, Revision 0, Replacement of Main Coolant Under/Over Current Trip System Relays. This procedure provided the detailed step by step installation instructions and specified the use of OP-4607, Low Main Coolant Flow System A & B (MC Pump Current) Channel Calibration, as the applicable post work test instructions. The inspector viewed the installation of the relays in the Main Coolant Flow Trip System Cabinets A and B located in the Switchgear room.

At this time, the inspector verified the modification would be satisfactorily completed upon the successful performance of OP-4607 and close out of Nonconformance Report No. 82-30, which requires receipt of correct purchasing documentation for 18 of the Model ITE-50H Brown Boveri Electric relays.

The component failures associated with the Main Coolant Flow relays were the subject of repetitive Licensee Event Reports dating back to January 22, 1981 and continuing up to September 2, 1982 for a total of eight events. The repetitive nature and licensee long-term corrective action plans were discussed in the Systematic Assessment of Licensee Performance (SALP), dated August 30, 1982. The modification activities required to replace the subject relays completes the licensee's program to investigate replacement relays and requires no further inspection effort in this area.

For record keeping purposes, the close-out of this Inspector Follow Item completes the review of Licensee Event Reports (50-29/82-18)and (50-29/82-26) describing Main Coolant Flow Over/Under Current Pump Relay failures that occurred on June 24, 1982 and September 2, 1982, respectively.

This item is considered closed.

e. (closed) Follow Item (50-29/82-08-01): Review Use of DP-0441, Special/Temporary Procedures. The inspector reviewed DP-0441, Temporary/Special Procedures, Rev. 5, dated 10/82 and noted procedure changes had been made resulting from the IR 50-29/82-08, Appendix A Violation. Specifically, the use of temporary/special procedures to invoke a "change of procedure or method" is no longer authorized, and a new procedure requirement for all temporary/special procedures to be reviewed by the Security Supervisor prior to distribution has been added. The inspector reviewed the Temporary/Special Procedures File maintained in the CAS and the master file maintained by the Chief of Security for Temporary/Special Procedures No. 82-157, July 19, 1982 through 82-237, October 21, 1982. This item is closed.

f. (closed) Violation (50-29/82-08-02): Failure to Control Spent Fuel Pool Access. The inspector reviewed IR 50-29/82-08, Appendix A, dated September 1, 1982 and the licensee's response contained in FYR 82-94, dated September 27, 1982 which detailed corrective actions. The inspector determined that proper immediate corrective actions were taken. A followup review of Security Temporary Instructions was completed in conjunction with IFI 50-29/82-08-01 followup actions. This item is closed.

g. (open) Follow Item(50-29/82-10-01): Missing Bonnet Flange Bolt on 1¼ in. Motor Operated Valve PS-MOV-191 (Pressurizer Spray Valve). As part of the event followup the inspector reviewed Yankee NSD Memorandum MSG 121/82 which forwarded to the site approved calculation YRC-149 which confirmed that the existing five of the six body to bonnet studs were not over-stressed by operating loads and would satisfy ASME Code Section III criteria. This satisfies the inspector technical concern for system integrity, the licensee intends to document the circumstances surrounding the event in Plant Information Report (PIR) 82-10. This event remains open.

h. (closed) Follow Item (50-29/82-10-03): OP-7200 Instrumentation Tube Minimum Diameter Acceptance Criteria. The inspector reviewed Exxon Nuclear Company letter TJH 378-82, dated October 11, 1982 which documented a subsequent inspection of the Five XN-5 fuel assembly instrumentation tubes utilizing a 0.387 in. diameter plug gage attached to a solid rod. Corrective action has also been taken to insure that the on-site plug gages and testing techniques are identical to that used during fuel assembly inspection at ENC. This item is closed.

i. (closed) Follow Item (50-29/82-10-04): Incorporation of OP-7200 Procedure Change into Document Record Copy. On October 28, 1982 the inspector reviewed the record copy of OP-7200 and noted that the Temporary Procedure Changes reviewed during PORC meeting 82-38 were incorporated into the procedure as described in AP-0001, Plant Procedures and Instructions. This item is closed.

3. Review of Plant Refueling Outage Operations

A. Daily Inspection - The inspector verified the following by direct observation of the activities, tours of the facility, discussions with plant personnel, independent verification, and facility record review:

1. Control room activities were observed to verify proper manning and access control, adherence to approved procedures, adherence to Limiting Conditions for Operation (LCO's), ESF status and selected value confirmation using a unit specific checklist; selected instrument and recorder trace review; nuclear instrumentation (N/I) operability verification; conformance with refueling shutdown margin limits; verification of containment status required for refueling operations; primary vent stack trace review and release followup; verification of onsite and offsite emergency power source availability; control room documents, including: operator logs, maintenance and surveillance documentation, and operator orders were reviewed to note trends, apparent anomolies, routine outage operations, and establish items requiring inspector followup.

2. During daily entry and egress from the protected area (PA) security activities were observed to verify access controls in conformance with the security plan for personnel, packages, vehicles, guard manning and conduct; selected PA barriers, and gates were examined; isolation zone conditions were observed; and licensee monitoring for radioactive materials prior to personnel, materials and equipment release for unrestricted use was monitored during egress from the PA. These checks were performed on the following dates: 10/1, 10/4, 10/6, 10/8, 10/12, 10/14, 10/15, 10/18, 10/20, 10/22, 10/25, 10/27, 10/29, 11/1, 11/3, 11/4, 11/5, 11/8, 11/10, 11/11, 11/12, and 11/15.

No inadequacies were identified.

B. Weekly System Alignment Inspection

Operating confirmation was made of selected piping system trains. Accessible valve positions in the flow path were verified correct. Proper power supply and breaker alignment was verified. Visual inspections of major components were performed. Operability of instruments essential to system performance was verified. The following systems were checked.

-- Lineup and Operation of Alternate Shutdown Cooling System per OP-2162, Operation of the Shutdown Cooling System, Rev. 9, performed on October 23, 1982.

-- Lineup and Operation of the boric acid mix tank (BAMT) per OP-2167, Boric Acid Mix Tank Makeup, Rev. 7, performed on October 12, 1982 and November 11, 1982.

No inadequacies were identified.

C. Biweekly Inspection

1. Portions of the following selected ESF surveillance were observed to verify: that test instrumentation was calibrated; redundant system operability; approved procedures used; work performed by qualified personnel; and acceptance criteria was met:

   -- Setting V.C. Integrity and Operability of the VC and Spent Fuel Pool Ventilation System, per OP-4239, Rev. 5, performed on October 1, 1982.

   -- Surveillance of the Station Power System and the Emergency Diesel Generators, per OP-4207, Rev. 14, performed on October 15, 1982.

   -- Station Battery Equalizing Charge, per OP-2500, Rev. 7, performed October 30, 1982 on No. 1 Station Battery.

   -- Safety Injection Tank Makeup, per OP-2654, Rev. 14, performed on November 3, 1982.

   No inadequacies were identified.

2. A review of the licensees sampling program was conducted by monitoring results of liquid and gaseous samples during the period to verify conformance with regulatory requirements; and boric acid tank (BAT) level and sample results were periodically reviewed for conformance with technical specifications:

   -- Release of No. 2 Test Tank per OP-2379, Rev. 7, performed on October 10, 1982 utilizing Release Permit No. 82-88.

   -- Release of No. 1 Test Tank per OP-2379, Rev. 7, performed on November 7, 1982, utilizing Release Permit No. 82-99.

   No inadequacies were identified.

3. Accessible facility areas were toured to make an independent assessment of plant and equipment. On a sampling basis the following items were observed or verified: condition of selected vital and access controlled barriers; radiation work permit completion and use; protective clothing and where applicable, proper respirator use; personnel monitoring practices; operational status of selected personnel monitors, area radiation monitors and air monitors; equipment tagout sample to verify LCO compliance for equipment out of service; plant housekeeping and cleanliness conformance with approved programs, and communication system operability.

   Inspector tours included the following areas:

   Control room, turbine building, auxiliary boiler room, switchgear room, screenwell house, spent fuel pit, primary auxiliary building, safety injection building, vapor container, pump and heat exchanger cubicles and radwaste handling complex.

No inadequacies were identified.

4. Plant Surveillance Observation

A. The inspector reviewed performance of surveillance testing involving a safety-related system including: review of surveillance procedure for conformance with regulatory requirements; calibration of test equipment; system removal from service and LCO compliance; monitoring portions of surveillance test performance; portions of system restoration to service; test data review for accuracy and completeness; confirmation of licensee test documentation review and discrepancy followup, test result compliance with TS criteria, personnel qualification, and adherence to surveillance schedule.

-- Hydrostatic Testing of Steam Generator and Lines, Steam Generator No. 1, per OP-2251, Rev. 5. Performed on October 15, 1982 at 150 psig to identify SG tube leakage (one tube identified, P-44), and performed on October 29, 1982 at 550 psig to verify SG No. 1 tube integrity following tube inspection and mechanical plugging as discussed in section 5.A.3. of this report.

No inadequacies were identified.

5. Inspector Review of Plant Events

A. Cycle XV-XVI Refueling Operations

During the entire inspection period the reactor plant remained in Mode 6, Refueling. Inspector observations and followup on outage activities is noted below:

1. Licensing Amendment 69 Commitment Verification

On July 22, 1981 the Division of Licensing issued Amendment No. 69 to License No. DPR-3 for Yankee NPS to incorporate Technical Specification (T.S.) changes necessary for core XV operation. Acceptance of several items in the related Safety Evaluation was contingent upon verification of commitments made by the licensee as detailed in letter G.C. Lainas to E.L. Jordan, dated August 30, 1981. During the Core XV refueling, the licensee upgraded the Auxiliary Feedwater Systems (AFW) by addition of two electric motor-driven pumps, with the associated piping and controls. The licensee agreed to perform a 48-hour flow test of the new pumps to verify that the pump operating parameters met the design specifications, the results of the test were to be reviewed by the Resident Inspector. This review of the 48-hour flow test for the new AFW pumps was performed and documented in Inspection Report 50-29/81-13, sections 5.b. and 9.b. Additionally, the Main Steam Non-Return Valves (NRV) were provided with automatic actuation logic and modifications were made to the Reactor Protection System (RPS) to change the reactor low pressure trip circuitry from 1/2 logic to 2/3 logic. These changes required the installation of additional relays which were the best available, but were not environmentally qualified. The licensee committed to replace these HGA relays with qualified relays at the first extended (3 to 4 days) shutdown after the relays were delivered.

The inspector reviewed Job Order Package 81-208, issued August 28, 1981, titled: HGA Relays Installed in the Non-Return Valve System-Unqualified Relays. This package contained various Material Issue (MI) tickets covering the release of qualified parts from the stockroom as the replacement for the non-qualified HGA relays. The inspector noted that included among these MI tickets was No. 19-092 for the release of eight additional HGA-111J2 relays used by the licensee for an additional test circuit (installed in the Control Room CIS cabinet) that was determined to be required and installed in July 1981. The inspector was informed by the cognizant plant I&C Engineer that these relays would not be changed-out as they were determined to be qualified initially, and the MI ticket was actually part of the original Job Order Package 79-250 (EDCR 79-19). The inspector noted that procedure AP-6007 originally provided applicable instructions to replace the eight HGA relays installed in the CIS cabinet, however the procedure was marked-up by the cognizant plant I&C Engineer as not being required to be performed. This area will be discussed further in this report section.

The inspector noted that Maintenance Request 81-959 required relays purchased under Material Purchase Request (MPR) 81-2Q/A23 to be replaced with relays purchased under MPR-81-2Q/A20 and 81-2Q/A34. Additionally, procedure AP-6007, I&C Department Corrective Maintenance, was initiated to specify the instructions for replacement of the unqualified relays and included applicable post-work testing instructions to provide assurance for equipment operability following replacement. The inspector reviewed the QA packages associated with MPR's to ascertain: that replacement parts and materials being used had the proper certification; and determine the licensee's actions relating to disposition of the removed components. As a result of these reviews, the following was noted:

-- MPR 81-2Q/A23: The HGA relays received per this MPR reflected the original non-qualified parts that required replacement per the licensing amendment commitment. A Nonconformance Report (NCR) No. 81-34 was generated by the licensee to reflect the fact that the relays did not have the proper certification. The July 29, 1981 disposition of this NCR was to temporarily accept the relays for operation, remove the hold tags, and have the NCR remain open pending further actions. A total of twenty HGA-111J52 125VDC coil relays and three HGA-111J70 120V ac coil relays were involved. The relays were replaced per licensee commitment and on November 8, 1982, the inspector located these relays in the Plant Stockroom. Two of the twenty HGA-111J52 relays were on a parts shelf with "green" Material Identification Tags reflecting QA material status. The other 18 relays removed from the plant equipment as part of the replacement parts program were untagged awaiting disposition. In discussions with the Plant Storekeeper the inspector noted that Hold Tags and a Material Return slip are required by AP-0213, Material Identification and Control. These actions are necessary to prevent inadvertent use of the material and to provide for its disposition.

On November 9, 1982 the inspector verified that immediate corrective action had been taken by: issuance of the required Material Return slip; and transfer of the twenty HGA-111J52 and three HGA-111J70 relays to Non-Nuclear Safety stock to preclude their use in safety grade or class IE applications.

-- MPR 81-2A/A20: On July 3, 1981 a quantity of twenty 12-HGA-111J2 relays were received on site per this MPR. Eight of these relays were installed in the CIS cabinet in the Control Room for the Non-Return Valve testability circuit as previously noted. NCR 81-43 was issued for these 8 relays due to a lack of documentation and applicable hold tags were attached. The NCR disposition stipulated that a supplement to the MPR be generated (to allow for documentation to be supplied), and to accept the relays for operation. The remaining 12 relays from this shipment had Hold Tags issued due to the nonconformance.

Subsequently, on November 20, 1981, a memorandum was issued by the NSD Cognizant Engineer to describe the fact that General Electric Co. had shipped in error control grade equipment(normally marked 12-HGA-111J52). It was further stipulated, that a supplement to the MPR be issued to have the 20 relays replaced.

Based upon additional review of the purchasing and installation documentation associated with the eight of twenty relays used in the test circuits of the Non-Return Valves located in the CIS cabinet, the inspector was informed on November 9, 1982 that the relays were non-qualified for use and were required to be reviewed and replaced with qualified units. The inspector noted that the original NCR 81-43 was still open and was never modified to reflect the information contained on the memorandum dated November 20, 1981. This precluded the change-out of the nonqualified, incorrectly marked relays as intended by the NSD Cognizant Engineer.

The inspector was told of the licensees plans associated with these 20 relays which consisted of:

1) Issue a new AP-6007 corrective maintenance procedure to remove and replace the 8 relays in question installed in the CIS circuitry.

2) Issue a new Nonconformance Report to reflect the disposition required for the 20 relays, and

3) Ship the 20 relays back to General Electric Co. once the relays are removed from the CIS cabinet.

A replacement of twenty 12-HGA-111J2 qualified relays was subsequently received on site on May 21, 1982. The entire documentation package including the nuclear certification of these relays was reviewed by the inspector and found to provide the proper certification.

-- MPR 81-2Q/A34: An additional quantity of twenty 12-HGA-111J2 relays were purchased as a backup to prior orders of nuclear safety grade qualified units. Some of these relays were utilized in the replacement relay program. The inspector determined, as above, that proper certification was provided and on file at the plant. No outstanding issues exist for this parts order.

Based upon the review of design, installation, test and purchase part infomation, the inspector is satisfied that the replacement of non-qualified with qualified HGA relays will be completed as required and in conformance with the Licensing Amendment 69 commitments.

As part of the verification for conformance to the licensing amendment, the inspector reviewed EDCR 79-19, Automated Non-Return Valves and the installation of the HGA relays. A total of 26 relays were observed in-place in the NRV, CIS, and 5R cabinets located in the Control Room.

The inspector observed the installed qualified HGA relays in the NRV cabinet but without the vendor protective covers. In their place was a strip of plexiglas running across the front of the relays. The inspector was unable to ascertain the condition(s) that caused protective cover removal from a review of the EDCR 79-19 and associated documentation. The inspector was concerned for both the lack of documentation and the unknown aspects that the removal of the protective covers might have on the required environmental qualification of the relays. This situation was discussed with the NSD Cognizant Engineer and the inspector was informed of the conditions that resulted in the necessity to remove the protective covers. A subsequent telehphone conversation between the licensee and a representative of the General Electric Co. indicates that upon written request of YAEC, the G.E. Co. will provide documentation that the protective cover removal will not result in conditions that would void the relay qualifications. Furthermore, the inspector was told that an Engineering Change Notice (ECN) to EDCR 79-19 will be made to provide the basis and document protective cover removal from the 16 relays in the NRV cabinet. This item will remain unresolved pending review of the ECN by the Resident Inspector. Unresolved Item (50-29/82-12-01).

A meeting was held by the inspector with the licensee's senior station management on November 12, 1982 to discuss the various concerns described above.

The inspector had no further questions.

2. Revised Cycle XV-XVI Refueling Boron Concentration

Yankee NPS Technical Specification (T.S.) section 3.9.1 requires that in Mode 6 (Refueling) the boron concentration of all filled portions of the Main Coolant System (MCS) and the Shield Tank Cavity (STC) shall be maintained uniform and sufficient to ensure a K_{eff} of 0.93 or less, which includes a 2% $\Delta K/K$ (reactivity) conservative allowance for uncertainties. The basis for this requirement is to maintain limitations on reactivity to ensure that: 1) the reactor will remain substantially subcritical during Core Alterations, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel.

By memorandum YR.K.01.02.09, dated September 2, 1982, Yankee Nuclear Services Division (YNSD) forwarded to the site the approved (by Manager of Reactor Physics Group) refueling boron concentration calculation for the core XV-XVI refueling as 2125 ppm minimum. This concentration allowed for a 7% $\Delta K/K$ all-rods-in shutdown margin (T.S. 3.9) with allowance for the worst case stuck rod (2.9% $\Delta K/K$) and was based on a beginning of cycle (BOC) core XVI value assuming an end of cycle (EOC) core XV average burnup of 13,000 MWD/MWT.

The approved refueling boron concentration was invoked for operations department personnel by Special Order 686 dated September 7, 1982 which required a MCS and STC refueling boron concentration of greater than or equal to 2125 ppm. This requirement is also contained in approved procedure OP-1100, Dismantling and Reassembly of the Reactor Systems for Core XVI Refueling, Revision 7, step 43.

On November 2, 1982 the site was notified by YNSD that a potential calculation error existed with the Cycle XV-XVI refueling boron concentration and recommended that the required minimum concentration (previously 2125 ppm) be increased to 2225 ppm. This was accomplished by Special Order 694 dated November 2, 1982, which directed the Operations Department to maintain a minimum refueling concentration of 2225 ppm. A meeting was held on November 3, 1982 between site management, the V.P. of Operations, and the inspector to discuss the circumstances surrounding the change in boron concentrations limits. The licensee stated that confirmation calculations were in progress and a preliminary assessment by YNSD indicated that the fuel temperature (doppler effect) correction factor had not been applied to the original Cycle XVI at power boron concentration adaptation to the refueling mode boron concentration. The licensee received confirmation on the evening of November 3, 1982 that the original concentration calculation was in error and a revised calculation, based on the actual EOC Cycle XV average burnup (13,458 MWD/MWT) and proper conditions, shows that a minimum value of 2172 ppm is needed to satisy T.S. requirements. YNSD documented this revised calculation in Memorandum RP-102, dated November 3, 1982.

A review of actual RCS concentration sample results during the period the original refueling boron concentration (2125 ppm) was invoked (September 17, 1982, [Mode VI declaration] 10:21 p.m. to November 2, 1982) indicated that the lowest concentration was 2190 ppm on September 22, 1982, this value is above the corrected T.S. limit of 2172 ppm. The licensee has revised OP-1100, step 43 to reflect the new administrative limit of 2225 ppm and a review of RCS boron concentration sample results (taken from shutdown cooling system) indicated that the concentration has been above 2225 ppm since September 23, 1982.

The licensee stated to the inspector on November 5, 1982 that they would document the circumstances surrounding the above finding in a Plant Information Report pursuant to the requirements of approved procedure AP-0004, Plant Information Reports. The inspector reviewed the licensee's T.S. reporting requirements and reviewed AP-0004 noting that the PIR including corrective actions, is reviewed and approved by the Plant Onsite Review Committee and submitted to the V.P. of Operations.

The inspector will review licensee evaluation of the above event and proposed corrective actions as documented in PIR 82-17 due for issue December 3, 1982. (Follow Item 50-29/82-12-02).

3. Steam Generator Tube Inspection and Repair

a. Steam Generator (SG) Pre-Inservice Inspection Hydrostatic Test

During Cycle XV operations the licensee monitored indications of an apparent primary to secondary leak within No. 1 Steam Generator. Parameters which indicated steam generator tube leakage included deviation from the normal water inventory balance and higher than normal #1 SG (Steam Generator) secondary side activity following a plant transient. The inspector periodically reviewed licensee calculations to confirm adherence to T.S. (Technical Specification) limits concerning allowable main coolant system leakage. Following unit shutdown for Cycle XV-XVI refueling outage on September 11, 1982, the licensee conducted a 150 psi hydrostatic test of No. 1 SG secondary side and confirmed a tube leak at location P-44. This was verified by eddy-current analysis to be a 100% defect on the hot-leg side, approximate 3 - 3½ in. above the tube sheet.

The inspector reviewed the official test copy of OP-2251, Hydrostatic Testing of Steam Generator and Lines-Steam Generator No. 1, Rev. 5, performed October 15, 1982. This procedure was utilized to verify suspected Steam Generator (SG) tube leakage prior to performance of SG inservice inspection (Eddy Current testing). The hydrostatic test was performed with the SG temperature at 80°F, with the SG filled to the main steam line non-return valve, and with the SG pressurized to 150 psig utilizing the electric emergency feed pump. An inspection of the SG primary side was conducted utilizing radiation work permit No. 1958 and it was determined that SG No. 1 had a tube failure at location P-44. This observation was documented in OP-7203, Eddy Current Examination and/or Repair of Steam Generator No. 1, Revision 5, OPF-7203.1 Step A.1.

b. SG Inservice Inspection

Also during the inspection period the licensee contracted for steam generator tube inservice inspection (ISI) services. Eddy current testing of No. 1 SG tubes was performed pursuant to the requirements of T.S. section 4.4.10. The inspector periodically reviewed the operations conducted per OP-7203, Eddy Current Examination and/or Repair of Steam Generator No. 1 and reviewed T.S. 4.4.10. compliance. Results of the tube inspection were as follows:

-- No. 1 S/G Hydro,	1 tube failure (P-44), 100% thru wall, hot leg 3-3½ in. above tube sheet
-- 1st Eddy Current Sample,	200 tubes examined (T.S. minimum=S=193 tubes)
	2 tubes greater than 40% degraded (P-42) 63%, O.D. (outer diameter), hot leg, ½ in. above tube sheet (K-44), 56%, O.D., hot leg, ½ in. above tube sheet
	1 tube marginal, 39% degraded (N-33), O.D., hot leg, 15 in. above tube sheet

-- <u>2nd Eddy Current Sample</u>, 400 tubes examined (T.S. minimum=2S=386 tubes)

1 tube greater than 40% degraded (H-32), 79%, O.D., coldleg, 3 in. above tube sheet

-- <u>3rd Eddy Current Sample</u>, Approximately 942 tubes (T.S. minimum=4S 772 tubes)

4 tubes greater than 40% degraded (H-28), 73%, O.D., cold leg, 3 in. above tube sheet

(H-25), 89% and 95%, 2 small pits, cold leg, 3 in. above tube sheet

(D-30), 89%, O.D., cold leg, 27 in. above tube sheet

(G-26), 65%, O.D., cold leg, 3 in. above tube sheet

Total Tubes Inspected:

Hydrostatic Test = 1602 tubes, full length at 150 psig.

Eddy Current = 200+400+942=1542.

Results = 8 defective and 1 marginal defective = 9 tubes plugged. Degraded tubes greater than 20% but less than 40% = 4.

Present Status of Tubes Plugged in No. 1 S/G:

9 Mechanical
2 welded
41 explosive

Upon review of the licensee's calculations for determining SG tube Eddy Current sample size the inspector noted that the P-44 tube defect confirmed by SG hydro was not included into the total SG tube defect calculation for the 1st inservice inspection sample. The inspector noted that if the defect had been included, the calculation results would have been greater than 1% defective (3 of 200 examined) which equals inspection category C-3 and requires extensive additional inspection of the unit SG's (all of tubes in No. 1 SG and 2S tubes in the other SGs). The inspector reviewed T.S. requirements with NRC Staff (NRR Materials Engineering Branch). It was concluded that exclusion of known SG tube leaks from inservice inspection (eddy current) sample results would be reviewed on a case basis and accepted for Cycle XVI operations with the following qualifications:

-- The licensee should evaluate implementing a program during the next SG ISI to examine the remaining SGs 2-4 cold leg tubes. This is based on the recent No. 1 SG Eddy Current inspection results where 5 of the 9 defects noted were located in the cold leg side of the SG.

-- It is also noted that current SG tube acceptance criteria requirements in T.S. section 4.4.10.4 a.8. defines a tube inspection as an inspection of the SG tube from the point of entry (hot leg side) past the top (fifth) support and, where practical, completely around the U-bend to the top support of the cold leg. Thus criteria which is consistent with Regulatory Guide 1.83, Revision 1 July 1975, seems to exclude inspection of the cold leg above the tube sheet, an area where tube defects are known to exist. This issue has been brought to the attention of NRR Licensing staff who will review the adequacy of current SG tube inspection requirements. (Follow Item 50-29/82-03).

c. SG Tube Repairs

The inspector reviewed licensee contractor operations to plug the 9 identified tubes in No. 1 SG. Previous plugging efforts at Yankee NPS utilized welded or explosive plugging techniques, for Cycle XV-XVI SG tube repairs the licensee contracted for development of a program to design and qualify a mechanical SG tube plug and installation method. This method was originally developed specifically for Model 44/51 SGs; however, the SGs at Yankee NPS differ in tube dimensions and material, therefore a new plug design and all of the qualification efforts were contracted to Westinghouse Electric Corporation for development and application.

The inspector reviewed Procedure No. MRS 2.3.2 GEN-13, Mechanical Plugging of Steam GeneratorTubing and Tube Holes, Revision 13, a vendor procedure prepared by Westinghouse Electric Corporation Nuclear Services Division, which was reviewed and approved by the Yankee NPS PORC Review Meeting No. 82-58 on October 28, 1982. Licensee T.S. section 6.5.1.7 b. and Licensee Operational Quality Assurance Program YOQAP-1-A section III.B.3.b. as invoked by approved procedure AP-0003, Plant Operations Review Committee Responsibilities and Authorities, Revision 6, section B.2. requires that PORC render determinations in writing with regard to whether or not procedures required by T.S. and proposed changes to plant systems that affect nuclear safety constitute an unreviewed safety question as defined in 10 CFR 50.59. The minutes of Plant Operation Review Committee Meeting No. 82-58, File D-02-02-02, October 30, 1982, section 1.c. contains the statement that the Committee reviewed vendor procedure MRS 2. 3.2 GEN-13, Revision 3, Mechanical Tube Plugging of Steam Generator Tubing and Tube Holes, found no unreviewed safety question and recommended approval which was given by Plant Superintendent signature on October 28, 1982. Regulation 10 CFR 50.59 section (b) requires that the licensee maintain records including a written safety evaluation which provides the basis for the determination that the change, test or experiment does not involve an unreviewed safety question. The inspector reviewed MRS 2.3.2 GEN-13, Revision 3 and supporting documentation from the vendor and did not note the presence of this documentation. On November 3, 1982 the inspector conducted a meeting with licensee management and requested that this documentation be provided in order for the inspector to determine the adequacy of the licensee's determination that no unreviewed safety question exists. This item is unresolved pending inspector review of documentation to be provided by the licensee. (Unresolved Item 50-29/82-12-04).

The inspector reviewed LER 50-29/82-38, dated November 10, 1982 which constitutes the 15 day SG ISI report required by T.S. section 4.4.10. 5.a. The LER accurately describes the SG ISI inspection and results.

The inspector had no further questions.

6. Review of Licensee Event Reports (LERs)

A. LERS submitted to NRC:RI were reviewed to verify that the details were clearly reported, including accuracy of the description of cause and adequacy of corrective action. The inspector determined whether further information was required from the licensee, whether generic implications were indicated, and whether the event warranted onsite followup. The following LERs were reviewed.

LER No.	Date of Event	Date of Report	Subject
*50-29/82-18	6/24/82	7/23/82	Main Coolant Flow System Failed Surveillance
*50-29/82-26	9/02/82	10/01/82	Main Coolant Flow System Failed Surveillance
*50-29/82-27	9/02/82	10/01/82	Auxiliary Boiler Room Sprinkler Out of Service Due to Gasket Leak
50-29/82-29	9/13/82	10/13/82	No. 3 SG Low Level Trip Set-point Out of Specification
50-29/82-30	9/24/82	10/22/82	No. 1 and 2 Battery Rooms' Halon Fire Protection Out of Service
*50-29/82-32	10/07/82	10/07/82 10/20/82	Fuel Assembly B-578 Positive Indication During Fuel Sipping
50-29/82-33	10/09/82	11/08/82	Refueling NI Channels Inadvertently De-energized
*50-29/82-34	10/09/82	10/12/82 10/22/82	No. 1 Main Coolant Pump Vent Line Crack
*50-29/82-35	10/13/82	10/13/82 10/26/82 11/05/82	Reactor Protection System Bistables 420 and 423 found to be Calibrated Out of Tolerance Due to a Surveillance Procedure Inadequacy
50-29/82-36	10/12/82	11/11/82	Pressurizer Code Safety Valve SV-181/-182 Out of Tolerance Due to Setpoint Drift
*50-29/82-38	10/27/82	11/10/82	Steam Generator No. 1 Tube Inspection and Repair, 15 Day Report per T.S. 4.4.10. 5.a

Except as noted in section B. below the inspector had no further questions:

B. For the LERs selected for onsite review (denoted by asterisks above), the inspector verified that appropriate corrective action was taken or responsibility assigned and that continued operation of the facility was conducted in accordance with Technical Specifications and did not constitute an unreviewed safety question as defined in 10 CFR 50.59. Report accuracy, compliance with current reporting requirements and applicability to other site systems and components were also reviewed.

A summary of the inspectors review findings follows or is documented elsewhere as noted below.

1. 50-29/82-18 and 50-29/82-26, Main Coolant System Failed Surveillance. Documentation of licensee corrective actions and inspector findings is contained in section 2.d. of this report. These LERs are closed.

2. 50-29/82-27, Auxiliary Boiler Room Sprinkler System Out of Service Due to Gasket Leak. Documentation of licensee corrective actions and inspector findings is contained in section 5.A. of inspection report 50-29/82-10, dated October 8, 1982. This LER is closed.

3. 50-29/82-32, Fuel Assembly B-578 Positive Indication During Fuel Sipping. On October 7, 1982 a positive indication was observed during the sipping of second cycle fuel assembly B-578. Analysis indicated that the assembly was probably the cause for slight reactor coolant activity increase during Core XV operation. The fuel assembly was stored in the spent fuel pool and subsequently examined by remote camera with no visible defects observed. By letter dated October 20, 1982, the licensee cancelled LER 50-29/82-32.

4. 50-29/82-34, Main Coolant Pump No. 1 Vent Pipe Crack.

   On October 11, 1982 the Plant Superintendent notified the inspector that a leak had been found on No. 1 Main Coolant Pump (MCP) vent isolation valve weld. The licensee reported the finding to the NRC pursuant to T.S. section 6.9.4.a(3) (abnormal degradation of Reactor Coolant System, RCS, boundary) and on October 12, 1982 issued LER 50-29/82-34, noting that the weld would be repaired during the current refueling outage.

   On October 22, 1982 the licensee submitted LER 50-29/82-34-01T (14-day followup report) describing the crack as a 3/8 in. indication in the vent pipe heat affected zone (1/2 in. schedule 80 stainless steel welded to a 3000 psi stainless steel coupling) and noting that the indication was verified thru-wall by liquid penetrant inspection (PT). The licensee indicated that the pipe would be replaced and subsequently examined with the inspection results made available, when received, along with further corrective and preventive actions.

   The inspector met with licensee management on October 25, 1982 and requested that LER 50-29/82-34-01T be resubmitted providing the following required information as a minimum:

   -- An accurate description of the problem, i.e., pipe crack in lieu of the terms "indication", "condition", "thru-wall indication", including the normal operating pressure, temperature, and environment of the pipe and whether the system is isolable.

-- The report should be termed "Interim Report" and indicate when the required supplemental information is expected.

The inspector indicated that not only is the above information required by approved licensee procedure AP-0002, Licensee Event Reports, but the NRC must have sufficient bases to evaluate the reported abnormal degradation of the primary system to determine if adequate corrective action has been taken prior to continued power operation.

On November 5, 1982 the inspector was informed by the licensee that during an inspection of the No. 1 vent pipe, a worker had used the pipe as a support for his weight and the vent pipe had broken off. A visual examination indicated that the No. 1 MCP vent pipe and vent valve (VD-V-710) appear to be a material other than stainless steel, possibly carbon steel. The licensee indicated that evaluation was continuing and details would be reported as a supplement to LER 50-29/82-34.

The inspector reviewed site documents and determined the following:

-- Yankee NPS MCPs are Westinghouse Model M-8003-A2. vertical, single-stage centrifugal pumps. Main coolant circulates between the stator and rotor (both canned components) and the motor bearings are lubricated by main coolant. (MCP Technical Manual)

-- Cooling and lubrication of the pump is provided by high pressure primary internal cooling water circulated within the pump. Component cooling water enters the cooling water inlet near the top of the motor circulates in the jacket around the coils (the coils contain the HP primary internal cooling water) and passes out through the cooling water outlet on the motor flange. The average temperature of the primary internal cooling water is about 160°F. The 160°F cooling water is not sealed tightly from the 496°F main coolant and therefore subject to MCS pressure. The MCP vent pipe connection is located on top of the pump unit, is used as the rotor cavity vent, and is indirect contact with the primary internal cooling water. (MCP Technical Manual)

-- The MCP No. 1 vent pipe and valve arrangement as field assembled at Yankee NPS consists of a 3/4 in. pipe connected to VD-V-710 (a 3/4" Jerguson Capped Stem Angle Valve) with a downstream outlet pipe ending with a blind flange. A review of licensee component history records for No. 1 MCP and Misc. Main Coolant Valves and Piping revealed no specific maintenance or replacement entries for VD-V-710. (YNSD Machinery History Records and YAEC Drawing 9699-FM-6A, Revision 29C)

-- The licensee indicated on November 12, 1982 that investigation revealed that VD-V-710 and the pipe nipple downstream of the pump connection were of carbon steel. A review of other 3/4 in. Jerguson valves in the plant was conducted by the licensee to verify proper valve material, no other discrepancies were noted.

The inspector will review licensee followup actions in this area. (Follow Item 50-29/82-12-05). This LER remains open.

5. -- 82-35, Reactor Protection System (RPS) Permissive Bistables; B/S 420 and B/S 423 Found Calibrated Out of Tolerance Due to Surveillance Procedure Inadequacy. On the afternoon of October 12, 1982 the Assistant Plant Superintendent notified the inspector that an error in the RPS permissive circuit calibration procedure had been noted which resulted in bistables (B/S) 420 (Start-Up Rate Permissive) and B/S 423 (At-Power Permissive) being calibrated to actuate the permissive circuit slightly above the required less than 15 megawatt-electric (MWe) or slightly below the required greater than 15 MWe setpoints. At 1:21 p.m. on October 13, 1982 the licensee notified the USNRC Region I Office and issued LER 50-29/82-35 OIT pursuant to the requirements of Technical Specification, Section 6.9.4.a (2). The inspector reviewed the circumstances surrounding this event as noted below:

Yankee NPS License No. DPR-3, Appendix A, Technical Specifications, Amendment 72, section 3/4.3.3.1 requires that, as a minimum, the RPS instrumentation channels and reactor permissive functions of Table 3.3-1 shall be OPERABLE, applicable as shown in Table 3.3-1. During Reactor Permissive Circuit Calibration and Functional Test performance conducted on October 12, 1982 per OP-4613, Rev. 4, plant personnel noted that bistable 420 (Intermediate Range Startup Rate (SUR) Permissive) calibration was out of tolerance with procedural requirements. Bistable 420 is normally closed at low power to implement SUR protection. OP-4613 acceptance criteria required bistable 420 to open at greater than 15 MWe, as measured by thermal converters off the main generator, to remove SUR protection and close at less than 15 MWe to institute SUR protection. A review of T.S. requirements by the licensee revealed that Table 3.3-1 requires I/R SUR protection to be bypassed prior to greater than 15 MWe increasing, and that the bypass be automatically removed prior to less than 15 MWe decreasing. The procedural and T.S. review by the licensee also noted that a similar condition existed for bistable 423 (At-Power Permissive) which is normally open (manually bypassed) at low power levels to bypass Low Main Coolant (MC) System flow trips, [Steam Generator Pressure (ΔP) and Main Coolant Pump (OC/UC)] and Low SG Narrow Range Level scram signals. OP-4613 acceptance criteria required bistable 423 to close at greater than 15 MWe increasing and open at less than 15 MWe decreasing. This is not conservative,in that Technical Specification Table 3.3-1 requires that the bypass is automatically removed (bistable closes) prior to greater than 15 MWe.

The inspector was notified of the finding by the licensee on October 12, 1982,and on October 15, 1982 met with site management to determine licensee intended followup actions. The licensee committed to a review of Yankee NPS accident analysis assumptions to verify that SUR and at-power permissive protective functions were not compromised by the calibration error.
On October 25, 1982 the licensee forwarded letter TAG 82-82, Transient Analysis Group Review of Low Power Trip, from J. Handschuh to R. Berry which documented a review of the Yankee Plant Safety Analysis to evaluate the effect of the changes in RPS trip de-energizing/energizing. In addition to the concerns of LER 82-35 OIT, dated October 13, 1982, it was also noted in TAG 82-82 that the degraded efficiency of the turbine had not been factored into the trip bypass point. This effect results from a higher

primary system (reactor) power necessary to achieve 12 MWe under turbine conditions less efficient than in previous operating cycles. The analysis concluded that the discovery of an inaccurate Low Power Trip Bypass did not modify the results and conclusions of the safety analysis based on a review of the two postulated transients that were examined in the safety analysis from an initial power level of 15 MWe (60 MWth). The Steam Line Rupture (SLR) and Rod Ejection transients were reviewed, and based on current licensing analysis methods which assume a zero MWth initial power level it was concluded that although small changes in trip de-energizing/energizing did occur, operation continued to be banded by the plant safety analysis.

On October 26, 1982 the licensee issued the 14-day LER 82-36-01T-0 which documented the findings of TAG 82-82 and noted that the discrepancy in bistable 420 and 423 trip/reset valves was due to a procedural error which has existed since June 16, 1977. The procedural error was caused by a misinterpretation of the T.S. which was translated into procedure OP-4613 acceptance criteria. The licensee has committed to a RPS circuit modification to facilitate correct RPS calibration and a revision to OP-4613 to reflect T.S. requirements and a review of T.S. related calibration acceptance criteria to ensure a similar condition does not exist in other procedures. These actions will be completed prior to reactor restart.

The above event constitutes a violation of T.S. section 3/4.3.1 in that the minimum number of RPS instrumentation channels and reactor permissive functions of Table 3.3-1 were not operable within the bistable setpoints of T.S. Table 3.3-1 Notation. The item meets the criteria of NRC Enforcement Policy, 10 CFR 2, Appendix C, 47 FR 9987 (March 9, 1982),section IV.A. for a licensee identified item and a notice of violation will not be issued, in that: the item was identified by the licensee, reported pursuant to T.S., section 6.9.4.a. criteria, it constitutes a Violation at Level IV (supplement I.D.); it was not an item that the licensee could reasonably be expected to prevent by corrective action from a previous violation, and the item will be corrected including measures to prevent reoccurence within a reasonable length of time. The licensee identified item remains unresolved pending completion of corrective actions as noted above (Unresolved Item 50-29/82-12-06).

During inspector reviews conducted in conjunction with LER 82-35 followup the inspector noted the following:

-- There is a lack of compiled information on site to facilitate a timely review of the Yankee NPS Safety Analysis. Due to the unique status of the facility Final Hazards Summary Report (FHSR), the reference safety analyses results are documented in each specific core reload analysis which inturn references previous reload analyses for individual accident bases and assumptions. Although it is recognized that the site utilizes the YNSD Transient Analysis Group for specific calculations and overall safety analysis functions, an updated reference document is needed on-site to provide licensee personnel with the background information necessary to perform preliminary evaluations of safety analyses impacts due to operational events. This observation was forwarded to YAEC management during a meeting conducted with site management and attended by the Vice President and Manager of Operations, on October 27, 1982. Site management stated that the need for such documents will be brought to the attention of corporate staff. (Follow Item 50-29/82-12-07).

-- During review of LER 82-35-01T-0, 14 day followup report, the inspector noted that RPS bistable 420/423 activation points as written in the event description appeared to be transposed. This was brought to the attention of licensee management on October 27, 1982. The licensee committed to review the LER and issue a resubmittal to correct the reported data.

On November 5, 1982 the inspector received LER 50-29/82-35-01T-1, dated November 5, 1982 which was a correction to LER 82-35-01T, dated October 22, 1982. The inspector reviewed the submittal for an accurate technical description of the event and had no further questions.

6. 50-29/82-38, Steam Generator No. 1 Tube Inspection and Repair. Documentation of licensee correcitve actions and inspector findings is contained in section 5.A.3 of this report. This LER remains open.

7. Plant Information Report (PIR) Reviews

The inspector reviewed PIRs prepared by the licensee per AP-0004, Plant Information Reports. The inspector determined whether the conditions were reportable as defined in the Licensee Event Reports reporting requirements section of the Technical Specifications (TS) and that the licensee's system of problem identification and corrective action is being effectively utilized The following PIRs were reviewed.

PIR No.	Occurrence Date	Report Date	Subject
82-11	9/13/82	10/13/82	Failure of the polar crane 15 ton hook rectifier and resistor
82-12	9/14/82	10/14/82	Spill from the Reactor Vessel Head Vent while purging loop two with compressed air.
82-13	9/20/82	10/20/82	Fire in the DW-MOV-655 Controller

No inadequacies were identified.

8. Part 21 Report Followup

On November 1, 1982 the inspector received a report dated October 11, 1982 from Conval Inc., to Yankee Atomic Electric Company pursuant to the provisions of 10 CFR 21. The manufacturer had determined that in certain valves supplied with stem material of ASTM A582 416 stainless steel and bonnet material of Nitronic 60 (UNS S 21800), a galling action is possible between the stem and bonnet which could substantially increase operating torque and ultimately result in inability to operate the valve. Conval reported that the onset of galling requires repeated operation of the valve and is most likely to occur at higher temperatures and where significant side thrust is experienced on the stem. The manufacturer recommended that those safety-related valves which contain the 416/Nitronic 60 materials be serviced at the earliest opportunity, and that the stem and/or bonnet be replaced with one of new design. The manufacturer claims that replacement of either part will eliminate the susceptability for galling. According to Conval records the following orders involve valves or parts with the affected material combination in probable safety-related service:

Purchase Order	Conval S.O.	Date Shipped	Items Affected
288901	5447	2/27/81	1. (3) 2 in. 12G2P-105 (8J) with Limitorque SMB-000-5 actuator: REF. YR-EDCR 79-16-52

NOTE: Only 2 valves affected, identified by HEAT CODE "M" on body and serialized 1076 and 1077 respectively.

100232	6479	12/09/81 to 12/28/81	1. (4) 2 in. 12G2-105 (8J) 2. (4) 1½ in. 12G2-105 (7H)

NOTE: Item 1 Body Heat Code "W", serialized nos. 1127-1130
Item 2 Body Heat Code "MF", serialized nos. 1123-1126

The inspector reviewed the licensee's actions taken in response to the Part 21 report as noted below.

The licensee documented receipt of the Conval Part 21 report on October 13, 1982 and assigned review responsibility by means of routing the report thru site management. Maintenance department records indicate the following status of the affected valves:

-- The two 2 in. valves with Limitorque actuators are currently installed and in use as:

EBF-MOV-555, normally open, Elect. Emergency Feed Pump Discharge Isolation

EBF-MOV-557, normally closed, Elect. Emergency Feed Pump Steam Generator Feed Header Isolation

-- The four 2 in. valves are currently installed and in use as:

MS-V-658, normally closed, Main Steam Line No. 1 Atmospheric Steam Dump Isolation
MS-V-669, normally closed, Main Steam Line No. 2 Isolation
MS-V-680, normally closed, Main Steam Line No. 3 Isolation
MS-V-691, normally closed, Main Steam Line No. 4 Isolation

-- The four 1½ in. valves are not installed, presently located in stockroom.

The licensee has ordered upgrade kits from Conval to provide a combination of material that is not subject to galling. The inspector reviewed maintenance request records and verified that upgrading of the valves is scheduled to be accomplished by MR 82-937, dated November 5, 1982. A review of site Stores Department disposition of the valves located in the stockroom revealed that the items were still located in the material bin with other 1½ in. Conval valves. The inspector noted this to department personnel who promptly removed the four 1½ in. valves from the Quality Assurance material storage area, placed them in a Hold Area and attached Hold Tags per AP-0212, Material Receipt.

The inspector noted to site management on November 9, 1982 that although the disposition of the subject Conval Part 21 report was acceptable in that the specific valves were located and timely corrective actions are planned, it appears that a mechanism is needed to formally document the assignment of responsibility and completion of corrective actions. Licensee procedure AP-0020, Operating Information Review provides for the above actions but does not specifically address Part 21 reports.

During a meeting on November 12, 1982 the licensee acknowledged the inspectors comments.

The inspector had no further comments.

9. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable. Unresolved items are discussed in paragraphs 5.A.1, 5.A.3.c., and 6.B.5. of this report.

10. Management Meetings

During the inspection period the following management meetings were conducted or attended by the inspector as noted below:

-- The inspector attended an exit meeting held on October 8, 1982 conducted by a region-based specialist at the conclusion of IR 50-29/82-12, Health Physics Appraisal Followup, on-site inspection.

-- At periodic intervals during the course of the 50-29/82-12 inspection period meetings were held with senior facility management to discuss the inspection scope and preliminary findings of the resident inspector. A summary of findings was also provided by the inspectors on November 12, 1982 (see Paragraph 1 for attendees).