U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-29/85-11

Docket No. 50-29

Licensee No. DPR-3

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

Facility Name: Yankee Nuclear Power Station

Inspection at: Rowe, Massachusetts

Inspection Conducted: May 6 - June 10, 1985

Inspector: for T. C. Elsasser 7/3/85
H. Eichenholz, Senior Resident Inspector date

Approved By: T. C. Elsasser 7/3/85
T. Elsasser, Section Chief, Reactor Projects Section 3C date

Inspection Summary: Inspection on May 6 - June 10, 1985 (Report No. 50-29/85-11)

Areas Inspected: Routine onsite regular and backshift inspection by the resident inspector (141 hours). Areas inspected included: Review of licensee action on previous findings, operational safety verification reviews, bi-monthly safety system walkdown, review of events requiring telephone notification to the NRC, review of plant events, surveillance observations, review of radiological controls, maintenance observations, Plant Operations Review Committee activities, review of the Emergency Planning Drill, review of the potential for overpressurization of ECCS, and survey of the licensee's response to selected safety issues.

Results: No violations were inspector identified; however, two inadequacies involving the ISI Program (Appendix A) and procedure review practices (Section 13) were classified as licensee identified violations. Licensee responsiveness to NRC review and initiative involving a modification to the station battery rooms was considered a notable strength (Section 13). Several areas needing increased licensee attention were: adherence to surveillance review requirements (Section 8) and upgrading of inadequate radiation protection department procedures (Section 8).

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DETAILS

1. Persons Contacted

Plant Operations

B. Drawbridge, Assistant Plant Superintendent
T. Henderson, Technical Director
N. St. Laurent, Plant Superintendent

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

On April 26, 1985, the plant was at full power. A leak was discovered on the No. 1 extraction steam line on April 27, 1985 which resulted in a load reduction to 15 MWe to facilitate repairs. Power was then increased on April 28, 1985, with the load increase being secured at 176 MWe on April 29, 1985 due to a second steam leak which occurred on the extraction steam line to the No. 1 feedwater heater. Following these repairs, the plant was increased to full power. The plant remained at essentially full power until May 4, 1985 when a load reduction to 50% power was initiated to facilitate turbine throttle valve testing and condenser tube cleaning operations. A load increase was initiated on May 6, 1985 and full power achieved on May 7, 1985. For the remainder of the period the plant was at essentially full power, other than minor restrictions that resulted from increased cooling pond water temperature with it's attendant elevated condenser circulating water discharge temperature.

The licensee determined on May 13, 1985 that a control rod movement restriction was required to provide compliance with the performance requirements of 10 CFR 50.46. Details of this event are provided in Section 7 of this report.

At the completion of this inspection period, the plant has been in continuous operation for 206 days.

3. Licensee Action on Previous Inspection Findings

(Closed) Inspector Follow Item (50-29/84-13-03). The licensee was to revise OP-2001, Responsibilities and Authorities of Operations Department Personnel, to clarify the review responsibilities of the Shift Supervisor related to completed surveillances on his shift. The inspector reviewed Revision 15 to AP-2001 and noted that it provided the necessary clarification. This item is closed.

(Closed) Inspector Follow Item (50-29/84-20-05). This item required the licensee to provide the basis for excluding eight valves from a one time demonstration test for Systematic Evaluation Program (SEP) Topic III-10.A, Bypass

of Thermal-Overload Devices. Licensee letter FYR 85-47 to NRC:NRR dated April 25, 1985 specified that the subject overloads were not previously included in the submittal because of their association with other plant modifications. However, they were tested during installation and will be retested during the 1985 refueling outage for completeness of the SEP topic. This item is closed.

(Closed) Inspector Follow Item (50-29/84-20-06). This item involved SEP Topic VI-1, Surface Coating Inspection Program, and required a licensee information submittal to NRC:NRR and verification that the on-site inspection was consistent with the submittal. In the above enumerated licensee letter, the licensee described it's documented paint inspection program. The inspector verified that the onsite inspection was consistent with the described program. This item is closed.

(Open) Inspector Follow Item 50-29/84-20-09) Follow Dose Equivalent Iodine (DEI) levels due to apparent fuel cladding failure in core XVII. During the period of April 27 to June 10, 1985, fluctuations in DEI were noted to vary from 8.8% to 25% of the allowable TS limit. The licensee continues to maintain maximum bleed, purification, and changing flow rates (50 GPM) to maintain the steady state DEI levels at a minimum.

(Closed) Inspector Follow Item (50-29/85-04-02) This item required the licensee to submit a revised Proposed Change to the Technical Specifications to address SEP Topic XV-19, ECCS subsystem leakage limits outside containment. On May 7, 1985, the licensee submitted Proposed Change No. 185 (FYR 85-54) to NRC:NRR, which contained a 50-gal-per-day proposed limit to be included in Technical Specification (TS) 3.5.5/4.5.5. This item is closed.

4. Operational Safety Verification Reviews

a. Daily Inspection

During routine facility tours, the following were checked: manning, access control, adherence to procedures and LCO's, instrumentation, recorder traces, protective systems, control rod positions, Containment temperature and pressure, control room annunciators, radiation monitors, radiation monitoring, emergency power source operability, control room and shift supervisor logs, tagout logs, and operating orders.

(1) On May 14, 1985, the inspector observed that the Pressurizer Wide Range Level indicator, PR-LI-705, was indicating 105 inches. Other indication channels were providing pressurizer level values between 120 and 123 inches. The licensee issued Maintenance Request (MR) 85-158 on February 7, 1985 as a shutdown required item to provide resolution of the apparent indicator anomaly. The onshift licensed operational personnel and I&C foreman could not readily identify for the inspector the applicability of the TSs to the subject instrument channel. Following a review of surveillance procedures, the licensee identified that the instrument channel was one of the two required Accident Monitoring Instruments required for pressurizer water level in accordance with TS 3.3.3.5.

The inspector noted that there was an apparent lack of information and guidance available to the operators on what are acceptable operational discrepancies between the channels that monitor the pressurizer water level. Futhermore, station personnel were unable to identify for the inspector the acceptable error band for the PR-LI-705 instrument channel. The licensee considers the relative indication to be the important information being conveyed by this channel.

To demonstrate that the PR-LI-705 channel response was acceptable, on May 14, 1985 the licensee cycled pressurizer level. The response for the three pressurizer water level channels, including PR-LI-705, demonstrated that there was close agreement between the indicated change on each channel.

No violations were identified.

b. System Alignment Inspection

Operating confirmation was made of selected piping system trains. Accessible valve positions and status were examined. Power supply and breaker alignment were checked. Visual inspections of major components were performed. Operability of instruments essential to system performance was assessed. The following systems were checked:

-- Emergency Diesel Generator (EDG) unit standby verified during tours of the EDG rooms and control room board status review.

-- Charging System verified during control room board status review.

-- Standby status of the Safety Injection Accumulator verified during tour of the accumulator room and control room board status review.

-- Low and High Pressure Injection Systems verified during tours of the Safety Injection Building and control room board status review.

-- Motor driven Emergency Feedwater Pump standby status verified during tour of the Primary Auxiliary Building.

No discrepancies were identified.

c. Biweekly and other Inspections

(1) During Plant tours, the inspector observed shift turnovers; compared boric acid tank samples and tank levels to the Technical Specification; and reviewed the use of radiation work permits and Health Physics procedures. Area radiation and air monitor use and operational status was reviewed. Plant housekeeping and cleanliness were evaluated. Verification of tagouts indicated the action was properly conducted. The inspector identified the following deficiency:

-- During a tour of the Safety Injection Building on May 22, 1985, the inspector observed that various wiring prints were stored in the back compartments of the 480V Emergency Buses. The prints were labeled as uncontrolled documents. Inspector concerns related to the applicability of document control requirements and the potential inadvertent use of out of date documents were discussed with the Plant Maintenance Manager. Licensee action to correct this condition consisted of 1) removal of the wiring prints, and 2) implementing standing instructions to maintenance personnel that when they discover similar conditions, to remove the prints unless they are being controlled in an approved manner. The inspector had no further question on this item.

(2) Observations of Physical Security

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, vehicle searches, and personnel identification, access control, badging, and compensatory measures when required.

No violations were identified.

5. Bimonthly Safety System Walkdown

In lieu of the normal Bimonthly Safety System Walkdown, a special inspection associated with potential overpressurization of Emergency Core Cooling Systems was conducted (see report Section 13). During the review the inspector determined that the Low and High Pressure Injection Systems were operable, with no inadequacies identified as a result of this review.

6. Review of Events Requiring Telephone Notification to the NRC

The circumstances surrounding the following event requiring NRC notification via the dedicated ENS-line was reviewed. A summary of the inspector's review findings follows:

-- At 11:54 A.M. on May 13, 1985, the NRC was notified in accordance with 50.72(B)(1)(ii)(A) that the current Loss of Coolant Accident analysis may not be in compliance with Section I.A of Appendix K to 10 CFR 50 pertaining to axial power distribution assumptions. The subject at this notification is discussed in additional detail in Section 7 of this report.

7. Inspector Review of Plant Events

a. On May 8, 1985, the licensee isolated the Condenser Steam Dump (MS-PCV-402) by closing the upstream isolation valve (AS-V-617). Operation of the plant with the turbine bypass line to the condenser isolated was

initiated to eliminate leakage past the steam dump valve. The Operations Department issued Special Order No. 85-22 to the plant operators which provided procedure changes and the applicable safety evaluation.

The inspector reviewed the safety evaluation and determined that the licensee had provided a written basis in accordance with 10CFR 50.59 (a)(2), that, operating the facility with the turbine bypass line to the condenser isolated is not an unreviewed safety question. Immediately following a plant trip, the isolation valve AS-V-617 will be opened to allow decay heat removal. A licensee evaluation indicates that if the operator action is taken within approximately five minutes after the plant trip, the main steam line safety valve setpoint should not be reached. The inspector had no further questions on this item.

b. On May 13, 1985 at 11:32 a.m., the plant was informed by their Corporate Headquarters that the Loss of Coolant Accident (LOCA) analysis may not be in compliance with the requirements of Section I.A of Appendix K to 10 CFR 50.46 pertaining to axial power distribution assumptions. At 11:54 a.m., the licensee notified the inspector of the identified deficiency, and initiated a call to the NRC in accordance with 50.72 (b)(1)(ii)(a).

To assure that the plant operation stays within analyzed conditions, a more restrictive control rod limit was immediately implemented that restricted Control Rod Group C withdrawal to 84 inches. This limit on Group C operation served to keep power skewed away from the top of the core. Since core power distribution varies with core life, as well as control rod position, a further restriction that limited Group C withdrawal to 83 inches was implemented on June 6, 1985 prior to reaching a core exposure of 11,000 MWd/Mtu. The need for further correction is being reviewed by the licensee. The inspector verified that revised procedural instructions were issued by the licensee providing the necessary actions to assure operators will maintain the plant within analyzed conditions.

c. As a result of the NRC:NRR becoming aware of errors in the Exxon PWR LOCA analysis methods in March 1985, the licensee's Nuclear Services Division (NSD) was requested to evaluate NRC concerns regarding Exxon LOCA Analysis deficiencies. One concern was determined by the licensee to be applicable. This dealt with the acceptability of the axial power distribution study which had been submitted in 1975 as required by Appendix K. Although the NRC:NRR safety evaluation of December 4, 1975 reviewed and approved the LOCA analysis for the Yankee plant which used the chopped-cosine power distribution, the validity of an Exxon assumption that the Westinghouse-derived K(Z) curve was applicable to the Exxon fuel was now being questioned. NRC:NRR has concluded that no axial power shape sensitivity studies have been performed for the Yankee plant which support the use of the maximum linear heat generation rate at all core elevations. Because the NRC:NRR does not have sufficient information to conclude that the plant remains in conformance with 10 CFR 50.46, and because the TSs may not be adequate, a request for information was transmitted to the licensee on May 22, 1985. The information, which is

to be submitted by the licensee by June 28, 1985, is to demonstrate that the current plant LOCA analysis and TSs conform to 10 CFR 50.46. Furthermore, if the licensee cannot provide the required demonstrations, they are to provide plans and a schedule for performing the analysis and TSs review.

The licensee plans on issuing LER 50-29/85-01 to document the event. The inspector will continue to follow the licensee's corrective actions as part of LER followup.

8. Monthly Surveillance Observation

The inspector observed tests and parts of tests to assess performance in accordance with approved procedures and LCO's, test results (if completed), removal and restoration of equipment, and deficiency review and resolution. The following tests were reviewed:

-- OP-4606, Nuclear Instrumentation Channels Functional Test

-- OP-4220, Primary System Water Balance.

-- OP-4207, Weekly Surveillance Test of the No. 1 Emergency Diesel Generator and the AC Power Distribution System.

-- OP-4204, Monthly Test of Safety Injection Train No. 3

-- OP-4674, Process Radiation Monitoring Channels-Electronic Alignment

-- OP-4801, Function Test and Alarm Setting of the Process Radiation Monitoring System

As a result of inspector review in this area, the inspector identified the following items:

(1) Regarding the performance of OP-4220, Primary System Water Balance, the inspector noted on May 13, 1985 that the surveillance records associated with the procedure were being signed by licensed personnel other than the Shift Supervisor (SS) which is contrary to the procedure's instruction. In addition, AP-2001, Rev. 15, Responsibilities and Authorities of Operations Department Personnel, clearly indicates that it is the shift supervisor who is responsible for the review of completed Operations Department procedures on his shift.

A review of completed OP-4220 surveillance records dating back to April 2, 1985 indicates that the identified inadequacy is more than an isolated occurrence. On April 2 and 4, 1985, the review signature of the surveillance was that of a licensed Reactor Operator who is in training for a Senior Reactor Operator's license.

In Inspection Report 50-29/84-13, the inspector identified a similar occurrence and requested the licensee to initiate corrective measures. These measures, identified by the licensee, included stipulating the review responsibility of the SS in procedure AP-2001. A discussion of this issue was held by the inspector with the Plant Operations Manager (POM) on May 13, 1985, who acknowledged the inspector's comments and concerns. Immediately, the POM issued a departmental memorandum to each licensed operator that directs them to adhere to the applicable procedural requirement in AP-2001. The licensee was informed that further recurrence of the identified deficiency could result in enforcement action.

(2) During the performance of OP-4204, Test or Special Operation of the Safety Injection Pumps and determination of ECCS Subsystem Leakage on May 30, 1985, the inspector observed erratic operation of the control room pressure indicator, SI-PI-6, that monitors the discharge pressure of the Low Pressure Safety Injection (LPSI). This indicator is utilized by the control room operator to verify pump performance. The inspector had difficulty in discerning whether the procedural acceptance criteria required by the TS 4.5.2.a.2.b. of greater than or equal to 250 psig was being met by the surveillance. In response to the inspector's questions, the control room operator obtained a value of 260 psig from the local pressure indicator being observed by the auxiliary operator that confirmed the pump was performing acceptably.

The inspector recommended to the POM that consideration be given to utilize local indication for discharge pressures associated with the LPSI pumps if it provides more reliable data. The licensee issued Maintenance Request 85-666 to correct the erratic operation of indicator SI-PI-6. The inspector had no further questions on this item.

(3) On June 6, 1985, the inspector observed the performance of post work operability surveillance in accordance with OP-4801 following repairs to the No. 1 Steam Generator Blowdown Monitor. This testing is necessary to determine operability prior to returning this TS required instrument to service. The inspector noted that the test did not meet the acceptance criteria and an adjustment to the instrument's high voltage was required to achieve acceptable results. When questioned about the instruction utilized to readjust the high voltage, the Radiation Protection (RP) Department Technician informed the inspector that he was complying with verbal instructions from a RP Engineer. The inspector determined that procedure OP-4801 neither prescribed instructions to the technicians for required actions if the acceptance criteria is not met while performing post work testing nor describes appropriate instructions for adjusting the instruments high voltage.

Following identification of the inadequacy, the inspector held discussions with the Radiation Protection Manager (RPM) on the issue. In responding to the inspector's comments and concerns, the RPM concluded that appropriate procedural controls should be developed to provide assurance that off-normal situations, including the event described above, will result in acceptable corrective measures when performing OP-4801. Weaknesses pertaining to poorly written procedures in the RP area were identified in the recent SALP Report (50-29/85-99) as a NRC concern.

According to the Plant's Technical Director, the necessity to obtain additional staffing to aid the ongoing RP area procedure review and upgrading process has been identified, and is currently receiving Senior Operational Management attention. The inspector will follow the licensee's revision of OP-4801 that will prescribe actions required if unacceptable test results are obtained (50-29/84-11-01).

9. Radiological Controls

Radiological Controls were observed on a routine basis during the reporting period. Standard industry radiological work practices, conformance to radiological control procedures and 10 CFR Part 20 requirements, were observed. Independent surveys of radiological boundaries and random surveys of non-radiological points throughout the facility were taken by the inspector.

-- On May 22, 1985, the inspector reviewed the licensee's practices and administrative controls applied to High Radiation Exclusion Area (HREA) keys. These controls are contained in Procedure AP-8010, Rev. 4, High Radiation Area Control. Attachment B to the procedure specifies, in part, that keys to HREAs shall be under the administrative control of Radiation Protection Supervision (Administrative Controllers), and keys may be issued by the administrative controller or any individual designated on APF-8010.3 by the Plant Radiation Protection Manager (RPM).

The inspector noted that form APF-8010.3, Designated Key Issuer Sheet, listed twelve individuals who are assigned access keys; however, four of them have either left the licensee's employ or are no longer in the Radiation Protection Department. Each of the twelve individuals listed on form APR-8010.3 have a numbered key issued to them that provides access to the control point repository which contains a set of HREA keys. The licensee's cognizant Radiation Protection Engineer (RPE) demonstrated accountability for the four errouneously assigned keys, initiated action to issue an updated form APF-8010.3 to reflect current designated individuals who have the authority to use the key to the repository, and issued an updated key issue roster documenting the status of the keys issued to date.

Procedure AP-8010, Attachment C, requires an inventory of keys in the locked respository without specifying a frequency. According to the licensee the inventory is conducted once per week.

TS 6.12 specifies that keys to HREAs shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist. In light of this requirement the inspector questioned the adequacy of the licensee's once per week inventory of the HREA keys and the apparent abundance of keys issued to Radiation Protection (RP) Department personnel for access to the repository at the control point. In response to inspector concerns, the licensee's RPE issued a memorandum to all RP Shift technicians to perform and document an inventory of the repository as part of the shift relief routine. The inspector noted that this practice was subsequently formalized in procedure OP-8042, Rev. 1, Radiation Protection Shift Personnel Duties and Surveillances. Additionally, the licensee is reviewing it's practices and controls associated with the keys to the control point repository. The inspector noted that although Procedure AP-8010 contains instructions, including compensatory measures, detailing actions required for the loss of a HREA key, there are no controls established to deal with the loss of a repository access key.

The acceptability of the licensee's practices and established administrative controls for HREA keys in considered an unresolved item pending further NRC review (50-29/85-11-02).

10. Monthly Maintenance Observation

The inspector observed and reviewed maintenance and problem investigation activities to verify compliance with regulations, administrative and maintenance procedures, codes and standards, proper QA/QC involvement, safety tag use, equipment alignment, jumper use, personnel qualification, radiological controls for worker protection, fire protection, retest requirements, and reportability per Technical Specification. The following activities were included.

-- MR 85-565, No. 1 Component Cooling Pump, Motor, and ACB Routine Inspection.

-- MR 85-568, No. 1 Charging Pump-Excessive Leakage.

-- MRs 85-568 and 668, No. 3 Charging Pump-Excessive Leakage.

-- MRs 85-590, No. 4 Steam Generator Blowdown Monitor.

-- MR 85-643, Trip Valve Position Indication Light Panel - Loss Of Indication.

-- MR 85-666, Low Pressure Safety Injection Discharge Pressure Indicator SI-PI-6 Operates Erratic.

In addition to reviewing the above MRs, additional reviews relating to licensee maintenance activities are contained in Sections 13 and 14 of this inspection report.

No inadequencies were identified.

11. Onsite Review Committee

On May 22, June 4, and June 10, 1985, the inspector observed meetings of the Yankee NPS onsite review committee (PORC) to ascertain that the provisions of TS 6.5.1. were met.

Except for the following items, the inspector had no further comments as a result of reviewing the licensee's activities associated with the onsite review committee:

-- At PORC Meeting 85-20 on May 22, 1985, the licensee identified two situations that involved failure to provide committee review of temporary changes to procedures within 14 days of initiation of the change. Committee review requirements are stipulated in TS 6.8.4 and Station Procedure AP-0001. Following discussions with the licensee's Technical Director and Technical Services Supervisor, the inspector learned that these incidents were identified to be due to administrative oversights. Plant management plans corrective measures to preclude recurrence. Based upon the licensee identified cause of these occurrences, the inspector is treating this matter as a licensee identified violation in accordance with NRC guidance contained in 10 CFR 2, Appendix C.

-- Engineering Design Change Request (EDCR) 84-317, Masonry Wall Modifications Inside the Turbine Building and Switchgear Room Jet Impingement Plate, was reviewed by the PORC at Meeting 85-23 on June 4, 1985. The inspector observed a detailed and meaningful review of the proposed modification by the committee. Significant detailed comments resulted from the pre-PORC review of the EDCR by the Maintenance Support Department. Although the PORC considered the EDCR and Safety Evaluation adequate for preliminary construction activities, these documents were considered to contain insufficient detail for contractor work in and around the battery rooms. Regarding work related to the battery rooms, the committee made cogent comments pertaining to 1) restriction of the contractor to work within one battery room at a time, 2) all shoring placed in the battery rooms to support the pouring of the roof slab to be analyzed and independently reviewed by YNSD Engineering before placement of concrete, and 3) the Safety Evaluation must be revised to address precautions taken while working in the switchgear and battery rooms.

Following the PORC Meeting, the inspector notified NRC: Region I of the licensee's plans to implement the EDCR during normal plant operations. These plans included the pouring of a new concrete roof slab over the existing roof that covers both battery rooms. A concern was raised by NRC:Region I involving the potential difficulties in ensuring a plant

shutdown if a catastrophic failure of the roof over the battery room occurred during modification. This concern was transmitted by the inspector to the Plant Maintenance Manager and Maintenance Support Department Supervisor following the completion of the inspection period, with the recommendation that the licensee consider either pouring the roof over one battery room at a time or consider delaying the pouring of the concrete roof until the plant is shutdown for refueling. The licensee demonstrated a cooperative approach in regard to the aforementioned NRC concerns by considering to review the matter and investigate alternatives.

Subsequently, the Plant Maintenance Manager informed the inspector that the licensee plans to pour the roof slab over one battery room at a time. The inspector had no further question on this matter.

12. Emergency Planning Drill

The inspector participated in the review of the licensee's Emergency Drill which took place on May 15, 1985. This review included three major areas:

-- drill preparation/review of scenario

-- drill observance

-- review of licensees critique/presentation of NRC findings.

The details of the inspector's comments and findings were presented to the NRC:RI team leader and will be described in NRC Inspection Report No. 50-29/85-08.

13. Potential Overpressurization of ECCS

A special limited inspection was conducted during this inspection period due to long-standing concerns of the NRC about the possibility of overpressurizing Emergency Core Cooling Systems. The scope and inspection findings are documented in Appendix A of this inspection report.

14. Survey Of Licensee's Response to Selected Safety Issues-Steam Binding of Auxiliary Feedwater Pumps

An inspection was conducted to determine the actions that the licensee has taken to address the safety issue of steam binding of auxiliary feedwater pumps due to back leakage. This issue has been identified in IE Information Notice 84-06 and in the Institute of Nuclear Power Operations' (INPO) Significant Operating Experience Report (SOER) 84-3. The primary purpose of the inspection is the gathering of information to be used in determining if NRC staff action is necessary on this safety issue. A secondary purpose of the inspection is to determine the actions that licensees at operating reactors are taking in response to recommendations to INPO's SOERs.

The scope of inspection, as contained in Temporary Instruction 2515/67 of the NRC's I&E Manual, includes a determination as to whether procedures to prevent, detect, and correct backleakage have been implemented and whether personnel training has been scheduled. For those items that the licensee has not implemented, an alternate reason or justification was provided if documented by the licensee. All reported licensee actions were those implemented prior to April 1, 1985.

Inspection results are documented in Appendix B to this report and were submitted to I&E for inclusion in their survey results.

15. Managements 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 May 10, 1985 by region based specialists at the conclusion of Inspection 50-29/85-09, Review of the licensee's Environmental and Personnel Monitoring TLD Program.

-- The inspector participated in management meetings associated with the team inspection 50-29/85-08 of the licensee's Annual Emergency Plan Exercise conducted during the period of May 13-16, 1985.

-- The inspector attended an exit meeting held on May 31, 1985 by region based specialists at the conclusion of Inspection 50-29/85-10, Review of the licensee's Radiological Environmental Monitoring Program.

-- The inspector attended an exit meeting held on June 7, 1985 by a region based specialist at the conclusion of Inspection 50-29/85-13, Review of the licensee's construction activities associated with the Safe Shutdown System building.

-- At periodic intervals during the course of the inspection period, meetings were held with senior facility management to discuss the inspection scope and preliminary findings of the resident inspector.

APPENDIX A

INSPECTION REGARDING POTENTIAL OVERPRESSURIZATION OF ECCS

In accordance with the April 22, 1985 Memorandum from R. W. Starostecki, Director, Division of Reactor Projects, NRC-Region I to Resident Inspectors, a special limited inspection regarding potential overpressurization of Emergency Core Cooling Systems was conducted during the current inspection period. The scope of the inspection included documenting the As-Built interface configuration; reviewing surveillance and maintenance procedures; verifying proper application of procedures; and determining if the licensee is addressing related failure experience.

1. The interacing systems reviewed were: Low Pressure Safety Injection (LPSI), High Pressure Safety Injection (HPSI), and the Shutdown Cooling System (SCS). Since NUREG/CR-2069 (Summary Report of a Survey of Light-Water-Reactor Safety Systems) did not contain YNPS's component configuration, the information of the type contained in this document is provided in Attachment 1.

   YNPS was reviewed under the Systematic Evaluation Program (SEP), with the results documented in NUREG-0825, dated June 1983. SEP Topics V-11-A and V-11-B, Requirements for Isolation of High and Low Pressure Systems and Residual Heat Removal System Interlock Requirements, respectively, were covered in the plant review. Details of the concerns and proposed modifications pertaining to potential SCS overpressurization are contained in NUREG-0825. Attachment 1 depicts the valves that normally maintain isolation for each high/low pressure interface. In addition, those valves that can be used to provide the isolation function are identified. Effectively, the LPSI and HPSI headers outside containment are each protected from over pressurization by three check valves installed in high pressure piping within containment.

2. Surveillance activities applicable to the isolation valves in the LPSI and HPSI Systems were reviewed. The requirements for testing are specified in the TSs, and are associated with either containment isolation valve (CIV) leak rate determination (Appendix J) or ASME Section XI ISI Testing. The LPSI and HPSI header outboard isolation check valves (CS-V-621 and SI-V-14, respectively) are the only valves in these systems that are specified in TS Table 3.6-1, CIVs. However, they are exempt from Type C leak rate testing. Inservice Inspection and Testing program surveillance requirements are specified in TS 4.4.9.1 and 4.0.5. All check valves depicted in Attachment 2 are specified as having inservice testing requirements in the program. During refueling outages the licensee conducts flow tests of the ECCS system to verify all check valves open to flow (procedures OP-4208 and OP-4206). The only valves in the program for the subject systems that are required to demonstrate backflow isolation capability are the individual cold leg check valves closest to the loops (SI-V-18,19,20&21), and check valves CS-V-621 and SI-V-14. This testing is specified to be done during refueling, except that SI-V-14 is to have valve closure capability verified on a quarterly basis (actually tested once per week using a differential pressure check per OP-4204).

The ISI Program has taken exception to leak testing requirements for Section XI in favor of the leak testing requirements of Appendix J. This results in no leak testing being performed because of existing Appendix J exemptions. The pressure isolation valves (PIV'S) depicted on Attachment 2 are listed in the current program with a notation that the licensee is evaluating the feasibility of performing leak testing. Currently, the licensee is preparing a program plan revision that takes credit for ongoing pressure monitoring of the high pressure alarming device on the LPSI and HPSI Systems upsteam of these check valves which would alert the operators of a leaky check valve. According to the licensee's ISI Coordinator, this system meets the intent of ASME Section XI, IWV-3421. Based upon the established test methodology, there are no precautions/prerequisites necessary to prevent overpressurization of low pressure piping.

On May 25, 1985, the licensee informed the inspector that the test used to verify closure of check valve CS-V-621 (procedure OP-4204 once per week verification of differential pressure indication following operation of a LPSI pump) was inadequate to demonstrate the once per refueling required test. Although the licensee has identified and documented an alternative test that verified valve cycling and closure which provides ISI program credit, they identified the event as an inadequency in procedural requirements. An LER will be submitted by the licensee to document the event and their prescribed corrective actions. In accordance with the criteria established in 10 CFR 2, Appendix C, this item is classified as a licensee identified violation.

3. The review of maintenance activities and practices that apply to the subject isolation valves resulted in identifying six specific events that occurred between 1974 and 1982. Three of these events involved the LPSI header check valve CS-V-621, which involved valve cover weeping only. Two of the events involved two of the four downsteam cold leg check valves (SI-V-18 on July 1974 and SI-V-21 on July 1977) which involved leakage past the check valve seat. Maintenance consisted of removal of clapper assembly, lapping the clapper and seat, and reassembling the valves. The final event involved internal valve inspection on the HPSI header check valve SI-V-14 per Information Notice 81-30 in 1982. No modifications or design changes of the isolation valves were identified by the inspector to have occurred at YNPS.

From a review of licensee procedures, records, and personnel interviews the inspector could not identify any specified preventive maintenance and component replacement policies for the isolation valves. QC coverage during safety related maintenance activities are involved with document review and/or inspection and/or audit of field documentation packages. Independent verification of maintenance activities is performed, but not on a 100% basis. YNPS utilizes Procedure OP-5104, Safety Related Valve Maintenance, as a routine maintenance procedure for work on the subject isolation valves. Requirements are stipulated for shift supervisor permission for release of equipment and identification of valves being utilized as isolating valves for plant and personnel safety. This procedure specifies that at the completion of maintenance work the valve has been returned to Operations Department control for the performance of post work testing. The Maintenance Request is utilized to document implemented post work testing and results.

4. In approximately 14 years of operation of the current Safety Injection System, there appears to have been no instances of actual or potential overpressurization of low pressure ECCS piping or components.

5. In general, operations and maintenance personnel maintain a strong regard for adherence to surveillance and maintenance procedural controls. Plant operators are formally trained in their surveillance duties associated with the routine testing of the HPSI and LPSI isolation valves. Although a generalized procedure (OP-5104) is utilized by maintenance personnel anytime work is performed on the subject isolation check valves, formal training is provided in the proper use of this procedure. Operators were knowledgeable about the continuous leakage monitoring system, that provides warning of excessive back leakage through the isolation check valves. In response to NRC dissemination of applicable operating experience regarding previous isolation valve problems, the following licensee actions were noted:

   - As a result of Information Notice No. 81-30, Velan Swing Check Valves, the licensee inspected the intervals of the two similar Velan check valves (one of which was SI-V-14: HPSI containment isolation check valve) and concluded that the identified failures would not occur at YNPS.

   - As a result IE Bulletin 79-04, Incorrect Weights for Swing Check Valves manufactured by Velan Engineering Corp., the licensee determined that their installed Velon Check Valves utilized the correct weights.

   - Currently, Information Notice 84-74, Isolation of Reactor Coolant System from Low-Pressure Systems Outside Containment, is undergoing evaluation per Procedure AP-0020, Operating Information Review. The preliminary evaluation has concluded that the information has been adequately addressed at YNPS; however, it was recommended to include the information in Maintenance Department training as a reminder of the importance of procedures, post installation testing and attention to detail.

6. A review was conducted of the following industry wide experience related to isolation barrier failures:

   - IE Information Notice No. 84-74, Isolation of Reactor Coolant System from Low-Pressure Systems Outside Containment.

   - Report to Congress on Abnormal Occurrences 84-8, Degraded Isolation Valves in Emergency Core Cooling Systems.

   - NRC Office for Analysis and Evaluation of Operational Data Engineering Evaluation Report AEOD/E414, Stuck Open Isolation Check Valve on the Residual Heat Removal System of Hatch Unit 2.

No inadequacies or weaknesses in the current facility design or procedures were identified by the inspector that could lead to overpressurization events at the YNPS. However, reasonable assurance of isolation capability could be enhanced by implementing an appropriate preventative maintenance program on LPSI and HPSI header check valves (including the containment isolation valves).

ATTACHMENT 1 - YNPS CONFIGURATION (NUREG/CR-2069)

Plant No.:

Yankee Nuclear Power Station

UNIT DESIGNATION (S)	1
DOCKET NUMBER (S)	29
COMMERCIAL OPERATION DATE (S)	7/1/61

REACTOR TYPE	PRESSURIZED WATER REACTOR
POWER (MWT)	600
NSSS	W
# OF LOOPS	4
ARCHITECT-ENGINEER	S&W
CONTAINMENT TYPE	STEEL SPHERICAL SHELL

INTERFACING SYSTEM LOSS OF COOLANT ACCIDENT **************

INTERFACING SYSTEM	HPCI	
PIPING LOCATION	IN	
NUMBER OF PENETRATIONS	1	Penetration Diameter 3 Inches

```
RCS-MOV-CK-CK-CK-H/L-I-PRV-CK-MOV-MV-CK-P
    LO                        LO  LO
```

LOW PRESSURE (PSIG)	1850	100 Deg. F
HIGH PRESSURE (PSIG)	2300	550 Deg. F
MONITORING	PIND CR-PRESS IND & AL CR	

INTERFACING SYSTEM	LPCI	
PIPING LOCATION	IN	
NUMBER OF PENETRATIONS	1	Penetration Diameter 8 Inches
COMPONENT LINE-UP		

```
                                           CK-MV-CK-P
RCS-MOV-CK-CK-MOV-CK-MOV-I-PRV-CK-MOV
    LO       LO     LO            LO       MOV-CK-ACC
                                           LO
```

LOW PRESSURE (PSIG) 720 100 Deg. F
HIGH PRESSURE (PSIG) 2300 550 Deg. F
MONITORING PIND CR-PRESS IND&AL

INTERFACING SYSTEM	SCS	
PIPING LOCATION	IN	
NUMBER OF PENETRATIONS	1	Penetrations diameter 6 Inches
COMPONENT LINE-UP		

RCS-MOV-MOV-H/L-I-PRV-MV-HX-MV-MOV-CK-P
LC LC NC NC NC
LOW PRESSURE (PSIG) 425
HIGH PRESSURE (PSIG) 2500
MONITORING PRESS IND-PRINL

INTERACING SYSTEM	SCS	
PIPING LOCATION	OUT	
NUMBER OF PENETRATIONS	1	Penetration Diameter 6 Inches

RCS-MOV-MOV-H/L-I-PRV-MV-P
LC LC NC

LOW PRESSURE (PSIG) 425
HIGH PRESSURE (PSIG) 2500
MONITORING PRINL

ABBREVIATIONS

AL	ALARM
ACC	ACCUMULATOR
CK	CHECK VALVE/COMPONENT CHECKING
CR	CONTROL ROOM
H/L	HIGH/LOW PRESSURE INTERFACE
HX	HEAT EXCHANGER
HPCI	HIGH PRESSURE COOLANT INJECTION
I	CONTAINMENT PENETRATION
IN	FLOW TOWARD REACTOR
IND	INDEPENDENT OR INDICATION
LO	LOCKED OPEN
LPCI	LOW PRESSURE COOLANT INJECTION
OUT	FLOW AWAY FROM REACTOR
MOV	MOTOR OPERATED VALVE
MV	MANUAL VALVE
MWT	MEGAWATT (THERMAL)
NC	NORMALLY CLOSED
NSSS	NUCLEAR STEAM SUPPLY SYSTEM
P	PUMP
PIND	POSITION INDICATOR
PRESS	PRESSURE
PRINL	PRESSURE INTERLOCK
PSIG	POUNDS PER SQUARE INCH (GAGE)
RCS	REACTOR COOLANT SYSTEM
RPV	REACTOR PRESSURE VESSEL
SCS	SHUTDOWN COOLING SYSTEM
S&W	STONE AND WEBSTER INC.
VS	VALVES STROKED FOR TESTING
W	WESTINGHOUSE

APPENDIX B

SURVEY OF LICENSEE'S RESPONSE TO SELECTED SAFETY ISSUES

Plant Name and Unit Yankee Nuclear Power Station

Item 03.02a. Steam Binding of Auxiliary Feedwater Pumps

1. Is the discharge or the suction piping of the auxiliary feedwater pumps hot?

   No. Verified by inspector on 5/9/85

2. Is the licensee monitoring and recording the temperature of the auxiliary feedwater system piping once per shift to detect back leakage? If not, how frequently?

   No monitoring is being performed.

3 a. Is temperature readout local or in the control room?

   There is neither local nor control room indication of temperature.

  b. What is the method of monitoring AFW piping temperature? (For example: touching pipe, temperature sensing tape, or pyrometer.)

   None performed.

4. Is the licensee monitoring the temperature of the auxiliary feedwater system piping after each operation of a pump to detect back leakage?

   No.

5 a. Did the licensee determine that procedural changes were needed to assure check valve seating when securing the auxiliary feedwater system?

   None were determined necessary.

  b. Have the changes been made?

   None were determined necessary.

6 a. Have procedural guidance and training in identifying back leakage and returning the system to operability been provided?

   No procedural guidance or training in identifying back leakage has been provided to the plant operators. Procedural guidance exists for returning the system to operability following routine testing.

b. Provide a brief summary description of procedural corrective actions (For example, vent and flush).

No procedural corrective actions have been implemented.

7 a. Is the licensee performing periodic leakage tests of the check valves (or isolation valves if normally closed) in the auxiliary feedwater discharge line? How frequently?

Infrequently.

b. Is the licensee performing periodic inspections of the check valves (or isolation valves if normally closed) in the auxiliary feedwater discharge line? How frequently?

Infrequently.

8. For any items which are not implemented, does the licensee have an alternate reason or justification?

Yes

If so, provide a brief description:

Item 2.

The licensee indicates that operators check equipment in their watchstanding areas, and therefore, would detect backleakage from check valves. Additionally, they indicate that the number of check valves required to leak (at least two) reduces the likelihood of this type of problem. The inspector found no evidence to substantiate the licensee's claim that operators check the piping for temperature to ascertain functioning of the check valves.

Item 4.

The licensee maintains that based upon past experience and normal watchstanding practices no action is necessary. The inspector's interview of Operations Department Personnel, and procedural instructions review, do not support the licensee's justification that credits watchstanding practices.

Item 5.

The licensee has stated that sufficient differential pressure is developed to close the check valves.

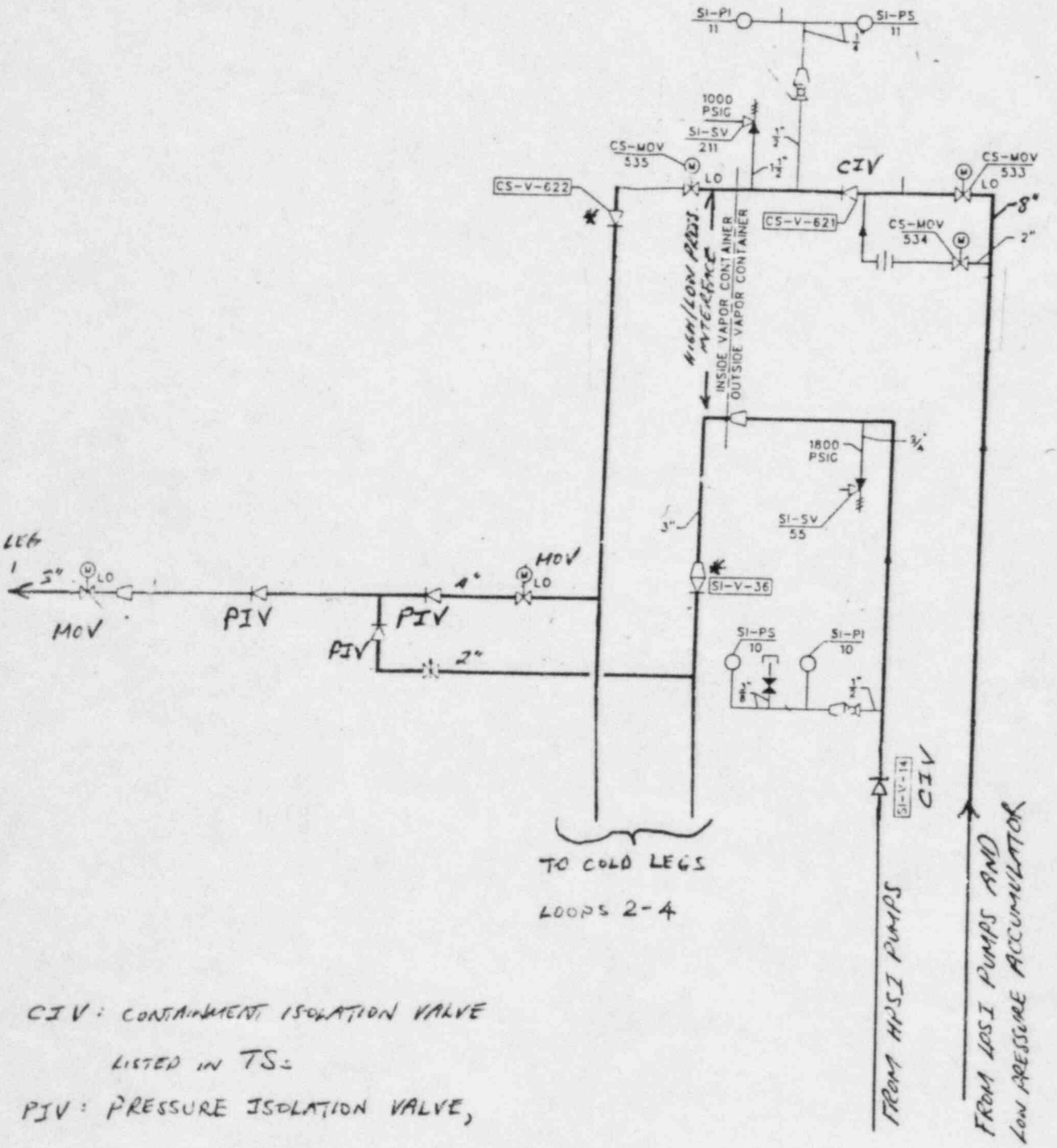
Item 6.

The licensee has indicated that with backleakage not having been a problem at this plant, no actions on this item is necessary. Plant training cover good watchstanding practices and actions which should be taken in the event equipment or systems are not operating normally.

Item 7.

The licensee indicates that flow paths are verified by procedure and therefore operation of the check valves is also verified. Maintenance on the check valves is performed on an as needed basis. No PM program exists for the check valves. When the motor driven emergency feedwater pumps have maintenance performed, the applicable procedure requires an inspection be performed on the discharge check valve of the pump.

ATTACHMENT 2 - IDENTIFICATION OF VALVES THAT MAINTAIN ISOLATION FOR EACH HIGH/LOW PRESSURE INTERFACE.



CIV: CONTAINMENT ISOLATION VALVE LISTED IN TS.

PIV: PRESSURE ISOLATION VALVE, NORMALLY USED TO MAINTAIN ISOLATION FOR EACH HIGH/LOW PR. INTERFACE

* VALVE CAN BE USED TO PROVIDE THE ISOLATION FUNCTION