DCS NO 50-213/84-06-11
50-213/84-06-12
50-213/84-07-21
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U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-213/84-14

Docket No. 50-213

License No. DPR-61

Licensee: Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, CT 06101

Facility Name: Haddam Neck Plant

Inspection at: Haddam, Connecticut

Inspection conducted: July 30 - August 2, August 14-20, and August 29 - October 31, 1984

Inspectors: Paul D. Swetland 11/14/84
Paul D. Swetland, Senior Resident Inspector — Date Signed

Paul D. Swetland for 11/21/84
John T. Shedlosky, Senior Resident Inspector — Date Signed

Thomas P. Johnson 11/29/84
T. Johnson, Reactor Inspector — Date Signed

Approved by: E. C. McCabe Jr 12/3/84
E. C. McCabe, Chief, Reactor Projects Section 3B, Division of Project and Resident Programs — Date Signed

Inspection Summary: Routine resident inspection (352 hours) of plant operations, maintenance, radiation protection, preparation for refueling, refueling operations, licensee events and followup on licensee recovery actions for a refueling cavity seal failure.

Inspector witnessing of cavity seal failure corrective actions identified satisfactory completion of the approved recovery program.

One violation related to the operability of plant fire barriers was identified. (Detail 2.2)

One violation related to the process for review and approval of field changes to design modification packages was identified. (Detail 6.3)

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DETAILS

1. Followup on Previous Inspection Findings

1.1 (Open) Unresolved Item (213/79-20-07) Correct the humidity control problem in the records storage vault. After rework and adjustment, worst case humidity remained about 40%. Documents which would be endangered by long term exposure to such conditions were transferred to the corporate office storage facility. This item is open pending determination of adequacy of procedural controls over storage of such records.

1.2 (Closed) Followup Item (213/83-13-04) The licensee was to restore Xray equipment to operation and improve fire protection of the ammunition storage location. A new Xray device was put into service on October 29, 1984. The inspector verified the satisfactory operation of this equipment. A fire detection system had been previously installed in the ammunition storage area. The inspector had no further questions in this area.

2. Review of Plant Operations

2.1 The inspector observed plant operation during regular plant tours throughout the reporting period. The following plant areas were inspected:

- -- Control Room
- -- Primary Auxiliary Building
- -- Vital Switchgear Room
- -- Diesel Generator Rooms
- -- Control Point
- -- Security Building
- -- Fence Line (Protected Area)
- -- Yard Areas
- -- Turbine Building
- -- Intake Structure and Pump Building
- -- Containment

Control room process instruments were observed for correlation between channels and for conformance with Technical Specification requirements. The inspector observed various alarm condidions which had been received and acknowledged. Operator awareness and proper response to these conditions were reviewed. Control room and shift manning were observed to be in conformance with regulatory requirements. Proper posting and control of radiation and high radiation areas was inspected. Compliance with Radiation Work Permits and use of appropriate personnel monitoring devices was verified. Plant housekeeping controls were observed, including control and storage of flammable material and other potential safety hazards. The inspector also examined the condition of various fire protection systems. During plant tours, logs and records were reviewed to verify that entries were properly made and communicated equipment status/deficiencies. These records included operating logs, turnover sheets, tagout and jumper logs, process computer printouts, and Plant Information Reports. The inspector observed selected aspects of the licensee's security organization including access control, physical barriers, and personnel monitoring. Except as noted below, no unacceptable conditions were identified.

2.2 During a tour of the auxiliary feedpump room on October 30, 1984, the inspector observed two electrical penetrations in the floor of the north end of the building. Penetration No. 801 was temporarily sealed with an approved fiber material. The temporary sealant material in penetration No. 800 had been removed to pull a test instrumentation cable through the penetration and had not been replaced. These penetrations provide access to the safety-related cable vault and are required to be sealed. Upon notification, the licensee promptly replaced the temporary penetration seal. The inspector verified that no fire watch had been established to monitor these penetrations.

Further review of this discepancy revealed that these penetrations had not been previously sealed. The licensee identified the open penetrations while initially running the temporary test cable in early August 1984. The penetrations were temporarily sealed; however, during a subsequent running of the test cable, penetration No. 800 was left unsealed. Technical Specification 3.22F requires all penetration fire barriers to safety-related areas to be functional or be monitored by an established fire watch within one hour. Pursuant to the above, maintenance procedure 8.5-143, Installation, Repair, and Inspection of Fire Barrier Penetration Seals, Revision 0 requires a fire watch to be established and maintained when a penetration is opened. The open penetrations had apparently not been identified previously, because this location was not specifically covered by the above inspection and maintenance procedure. The existence of two unsealed, unmonitored penetrations to the cable vault for more than one hour prior to August 1984, constitutes a licensee identified violation. However, the licensee's corrective actions in response to this condition did not prevent one penetration from again being made inoperable when the temporary sealant was removed for more than an hour and no fire watch was established. These items constitute a violation. (213/84-14-01)

2.3 During this refueling outage, the steam door at the entrance to the control room was kept open to facilitate access. This door is not a security access door. It provides blast protection from a main steam line rupture in the turbine building adjacent to the control room. This protection is not required during cold shutdown conditions. Since this steam door is also a fire barrier, control room personnel were assigned fire watch duties. The licensee normally assigns a dedicated fire watch to compromised fire barriers in accordance with a maintenance work order. In this case, these formal controls were not implemented. Control room personnel were aware of their obligation to shut the door in case of fire, and their fire watch duty was noted in the control room log. This deviation from standard practice for fire watches was found acceptable due to the availability and fire brigade qualification of control room personnel. The licensee committed to formalize guidance for the control of open fire doors to insure consistency in implementation of compensatory action. The inspector will review this guidance in a subsequent inspection (213/84-14-02).

In addition, the inspector noted that the control room and switchgear room blast doors are not self-closing doors. Fire doors are generally required to be self-closing and latching by National Fire Protection Association Code 80. Inspector review of the licensee's fire hazards analysis did not identify that this had been presented or addressed by the NRC's fire protection review. The licensee stated that both these doors are scheduled for replacement with code approved doors. Since the control room is continuously manned and the switchgear room door is verified closed once per shift, the inspector determined that there would be adequate control of these fire barriers until suitable replacements are obtained. This item will be closed upon completion of the door replacement project. (IFI 213/84-14-03)

3. Preparations for Refueling

3.1 New Fuel Receipt Inspection

The inspector reviewed the receipt inspection of 54 new fuel assemblies. These inspections were conducted in accordance with procedure 1.4-3, New Fuel Detail Inspection, Revision 7, during the period May - June 1984. Several discrepancies were identified during these inspections. Vendor repairs were accomplished in accordance with procedure SPL 10.7-218, Rework and Inspection of Batch 15 Fuel, Revision 0. The inspector noted three discrepancies in the completed inspection documents, including incorrect crane inspection dates and missing documentation of inspection finding dispositions. The licensee provided supplemental information showing proper disposition of these items. The inspector noted the need for more critical supervisory review of inspection packages to insure that all dispositions are properly documented. The completeness of receipt inspections will receive routine NRC reinspection.

3.2 Refueling Procedure Review

A review of the refueling and startup testing procedures was performed. This review included verification of procedure adequacy and implementation of plant technical specification requirements. The following procedures were reviewed:

-- FP-CYW-R12 Connecticut Yankee - Refueling Procedure Cycle XII - XIII
-- NOP 2.1-7, New Core Initial Critical Approach, Revision 8
-- SUR 5.3-6, Control Rod Reactivity Worth Measurements, Revision 14
-- SUR 5.3-3, All Rods Out Just Critical Boron Concentration, Revision 8
-- SUR 5.3-2, Hot Rod Drop Time Measurements, Revision 7
-- SUR 5.3-4, Zero Power Flux Map, Revision 4
-- SUR 5.3-17, Operation of the Flux Map System, Revision 8
-- SUR 5.3-5, Isothermal Temperature Coefficient Measurement, Revision 8
-- SUR 5.3-19, Boration Requirements for Reactor Shutdown, Revision 10
-- SUR 5.3-23, Excore-Incore Axial Offset Correlation, Revision 10
-- SUR 5.3-20, Reactivity Balance Procedure, Revision 5
-- SUR 5.3-24, At Power Flux Maps, Revision 5
-- SUR 5.3-26, Excess Reactivity Balance, Revision 4
-- SUR 5.3-39, Hot Rod Drop Time Measurements Using Digital Method, Revision 2

-- SUR 5.3-40, Core Quadrant Power Tilt Determination, Revision 0
-- SUR 5.1-25, Core Power Tilt Determination, Revision 1
-- SUR 5.1-26, Incore Power Distribution Monitoring Axial Offset, Revision 5

Upon completion of individual procedure review, the inspector discussed the refueling and startup test program procedures with a licensee representative. Several minor deficiencies were noted with the procedures, and these were indicated to the licensee for correction.

No unacceptable conditions were identified.

3.3 Core Reload Package Submittal

The inspector reviewed the licensee's submittals for modifications of Technical Specifications, (licensee letters dated May 2 and 25, July 24 and August 31, 1984), based on the Cycle XIII core reload. Amendments 59 and 60 to the plant operating license were approved on October 15, 1984. The inspector verified the implementation of these changes in appropriate plant documents. No inadequacies were identified.

3.4 Outage Control

The inspector reviewed the licensee's preparation for and coordination of refueling outage activities. Management level outage coordinators were assigned to provide 24 hour coverage of outage activities to maintain current critical path status. The licensee employs a full time outage scheduling staff. Performance of plant refueling operations was contracted to a refueling services vendor. Coordination and control of these operations were maintained by the plant maintenance department. The inspector had no further questions in this area.

4. Refueling Activities

4.1 The inspector observed the reactor disassembly process in preparation for refueling operations. On August 21, 1984, prior to commencement of refueling activities, the refueling cavity seal failed, draining the pool to the containment floor. This event was detailed in NRC Inspection Report 50-213/84-23. Plant recovery actions from this event are detailed in paragraph 6 of this report. Refueling operations were resumed after NRC approval of an upgraded seal installation on October 2, 1984. The inspector observed fuel handling operations and accountability in the containment, spent fuel pool, and control room. The following items were reviewed:

-- Core monitoring performed
-- Fuel handling, fuel accountability conducted in accordance with approved procedures
-- Core internals stored to protect against damage
-- Housekeeping and cleanliness conditions acceptable
-- Make-up and qualification of refueling crews
-- Refueling Cavity and Spent Fuel Pool water level as required by the cavity seal failure analysis
-- Boron concentration and make-up (shutdown margin) as required by Technical Specification 3.13

-- Constant direct communication maintained between the Control Room and fuel handling personnel when core alterations were in progress
-- RHR flow maintained to Refueling Cavity as required by Technical Specification 3.13
-- Audible count rate monitoring functioning in the Refueling Cavity Area and Control Room
-- Radiation monitoring requirements
-- Core Reloading Check-off sheets
-- Core Component Verification

Refueling concerns will be pursued under Inspection 50-213/84-23 followup.

5. Maintenance Program

5.1 The licensee's maintenance program was reviewed to determine the adequacy of measures implemented to ensure that equipment failures are evaluated for frequency and root cause, and to identify maintenance errors, their cause and corrective action. The inspector also reviewed licensee record systems to determine if they are organized to support the above functions.

The licensee's failure evaluation program is embodied in management control and corrective action programs. In addition to the initiation of repair activities, component failures result in management review and assignment of corrective action responsibilities. A Plant Information Report (PIR) is written for all significant failures. This report is reviewed promptly by management and establishes reporting requirements and assignment of event evaluation and development of corrective action. These reports receive an initial and final review by the onsite review committee (PORC). It is through this mechanism that the causal analysis and effectiveness of corrective action are evaluated. The inspector reviewed component failure PIR's issued in 1983. The system was found to be effective for management information, assignment of corrective action and tracking completion status. Internal discussion triggered by PORC review and intra-department assignment of responsibilities appeared to be effective in prompt identification of generic failure applicability, although these initial actions were not always well documented.

The activities of the plant maintenance department have been limited to performance of preventive and corrective maintenance and to documentation of these activities. As such, the maintenance department had no formal means nor objective to perform detailed evaluations of corrected component failures. Maintenance supervisors have been effective in updating preventive maintenance schedules based on failure experience. Inspector review of plant maintenance activities during 1983 did not identify evidence of repeated failure events or of similar failures to redundant components. The licensee records reviewed did not detail any component retest failure history. In addition, previous inspection results have identified no significant maintenance problems. The recent 417 day continuous operating period also points to the quality of plant maintenance activities.

Plant maintenance records have recently undergone a significant change. A computerized data system for scheduling, control and documentation of preventive and corrective maintenance was implemented in November 1983. The system provides a capability for trending component failure information which was previously not available, except for personnel recollection. The licensee has not yet implemented any formal trending program for this data base. The inspector also reviewed the charter and procedures for the corporate Nuclear Safety Engineering Group. This organization reviews PIRs to access the need for more generic action, primarily with regard to other licensee and utility facilities. The NSE group has trended PIR failure data.

The inspector concluded that the licensee has implemented an effective maintenance program. Weaknesses in documentation as they apply to maintenance history and trending have been recognized and new systems are being implemented. The inspector had no further questions in this area. Maintenance will receive routine NRC reinspection.

6. Reactor Cavity Seal Failure - Recovery Programs

In response to the August 21, 1984, reactor cavity seal failure, the licensee committed to implement a recovery program prior to resuming refueling operations. This program would review the design of the pneumatic seal, determine its failure mechanism, and implement corrective actions necessary to prevent the recurrence of a catastrophic seal failure and eliminate the potential for draining the spent fuel pool below the top of the active fuel. In addition, a complete evaluation of equipment wetted during the event was performed and corrective actions were completed prior to plant startup.

The inspectors followed these activities and the implementation of the associated design changes and modifications after the licensee had presented them to the NRC Office of Nuclear Reactor Regulation. At all times, the inspectors verified that data, conclusions, and the commitments made to the NRC were accurate and were implemented.

The licensee's programs were defined in three principal documents, Project Assignment (PA) 84-110, "Connecticut Yankee Pool Seal Failure Analysis, Resolution and the Return to Operations" and Plant Design Changes PDCR-672, "Fuel Transfer Canal Waterstop at El. 22'-0" Reactor Cavity" and PDCR-673, "Reactor Cavity Seal Modification". Additionally, Engineering Specifications SP-ME-474 and SP-CE-183 contain the construction specifications for the seal modifications and the fuel transfer canal waterstop, respectively.

Design criteria were established for the new cavity seal and its modifications. These criteria included the requirements that: the failure of a single component does not result in catastrophic consequences; the pneumatic seal be modified to provide substantial margin of safety against failures; a backup seal be provided which is capable of withstanding seismic, hydraulic and air pressure forces, assuming that the pneumatic seal failed; and that, the maximum possible backup seal leak rate allow sufficient time to move all fuel assemblies into safe positions without significant dose rates to the personnel performing the fuel transfer.

6.1 Safety Analysis

Corporate Nuclear Engineering and Operations Procedure (NEO-) 3.12, "Safety Evaluations" implements regulatory requirements for conducting safety evaluations. Revision 0 of this procedure dated March 31, 1983, was reviewed during this inspection and the implementation of its requirements for the above referenced design changes was verified.

The engineering discipline safety evaluations were available to the inspectors who conducted independent verifications of aspects of their content. In addition, the inspectors attended several meetings of the Plant Operations Review Committee(PORC) and a combined meeting of the Nuclear Review Board and PORC on September 13 (NRB 84-15). During those meetings they confirmed that the review functions were being conducted in accordance with the requirements stated in Technical Specification Section 6. There were no unacceptable conditions identified.

The inspectors observed that an in-depth technical audit was being conducted which actively challenged aspects of the analysis and the design.

In accordance with procedure NEO-3.12, paragraph 6.3, the licensee published an Integrated Safety Evaluation. This evaluation was reviewed by the inspectors. Various aspects were discussed with NRC:NRR personnel; however, there are no outstanding issues.

6.2 Design Change Controls

The licensee implements the regulatory requirements applicable to the control of facility modifications through Station and Corporate Procedures. These procedures establish the methods for identifying the need for a modification; the definition of design criteria; the control of construction specifications; the control of design and analysis calculations; and, the preparation of design drawings and drawing changes.

During the inspection of the associated modifications, the inspectors reviewed all or part of the following procedures: NEO-3.03, "Preparation, Review and Disposition of Plant Design Change Requests"; NEO-3.04, "Preparation, Issuance and Control of Project Assignments", Generation Engineering and Construction Division Procedures (GE&C) 2.01, "Preparation, Review, Approval ..., of NUSCO GE&C Division Specifications"; GE&C-2.04,"NUSCO Field Change Authorization"; GE&C-4.03, "Safety Evaluation and Technical Review of Technical Specification Change Requests"; GE&C-4.04, "Preparation, Review..., of Design Analysis, Technical Evaluations and Manual and Computer Calculations"; and, GE&C-5.07, "Processing Nonconformance Reports".

The inspectors reviewed the implementation of these modifications as presented in Specifications SP-ME-474 and SP-CE-183 and design drawings: 16103-51112, Reactor Cavity Backup Seal; 16103-51095, Reactor Cavity Seal Ring and Seismic Support; 16103-51113, Fuel Transfer Canal

Waterstop; 16103-51112, Reactor Cavity Seal Cover Assembly; 16103-20017, Air Supply to Reactor Cavity Seal; 16103-29018, Reactor Cavity Seal, NES/ Selamco Dwg 82053-1; and, 16103-22034, Reactor Cavity Liner. In addition, the following implementing procedures were available and were reviewed: CE No. 18767-RCE-802, "Guidelines for Reactor Pool Seal Reinforcing Pin Installation;" NUSCO Welding Procedure Specifications WPS-103, 105, and 616.

The inspectors noted that the licensee's QC organization had identified several nonconformances associated with the installation of the steel reinforcing pins in the pneumatic seal material. Specifically, station QC nonconformance reports (NCR) 84-191, 192, and 193 documented the findings that a modification to the seals had been made without a PCRC approved procedure or PDCR; the materials used (stainless pins) were not processed through the required receipt inspections to maintain identification and traceability, and work was performed under a non-safety-related work order. These NCR's represent licensee identified violations of NRC and quality assurance program requirements. The inspectors expressed concern to the licensee about this apparent disregard for quality assurance requirements. The licensee stated that these activities had been undertaken on a risk basis pending acceptable resolution of the NCR's. NCR resolution then found the installation acceptable by tracing the pins to supplier receipt records, review of the installation procedure, and inspection of the modified seals. The inspectors had no further questions on this concern.

6.3 Implementation Process

The inspectors observed the modifications to the Reactor Cavity Seal throughout its implementation. They used, as references, the NRC Licensing submittals, engineering specifications, PDCR's, and design drawings.

The installation of the two back-up seal cover place assemblies and seal ring seismic modifications required that component fit-up be made to relatively close tolerances. In the case of the back-up seal, the total fit-up clearance establishes the maximum leakage rate with a failed primary seal. This leakage rate was a basic assumption of the safety analysis.

A number of changes were made to the design in response to problems encountered in obtaining the specified tolerances. This aspect of the work, including seventeen Design Change Notices (DCNs), was reviewed by the inspector.

10 CFR 50, Appendix B, Criterion III requires design control measures to insure that field changes are reviewed by the organization that performed the original design unless another organization is designated to do that. The Northeast Utilities Quality Assurance Program commits to design control measures which meet Regulatory Guide 1.64 and ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants. These standards require that changes to design documents be reviewed and approved by the same groups or organizations that reviewed the original documents. Procedure NEO 3.03 requires design verification,

committee review and management approval for all plant design changes. Procedure GE&C 2.01 specifies a quality assurance and supervisory review of all design specifications. The revision of approved engineering design specifications, drawings and documents is permitted by GE&C-2.01, paragraph 6.6.4 and is controlled in accordance with GE&C-3.05, paragraph 6.6. The inspector reviewed the seventeen DCN's within the series 257-84 through 284-84 to determine that their content and administration met these requirements. In each case the inspector found that the DCN was controlled in accordance with the above procedures; however, the inspector noted that these procedures did not require that field changes receive the same review and approval process applied to the original design. Specifically, the design verification, committee review, and management approval of DCNs were not required, nor were the quality assurance and supervisory reviews required. Although the licensee subsequently incorporated all these DCNs to the modification into a revision of the original modification package and completed the required review cycle, the failure of the design control procedures to implement the requirements of the quality assurance program constitutes a violation (213/84-14-04).

The inspectors reviewed the work in progress on the seal ring to assure that the activities were in accordance with the approved design documents. NUSCO Betterment Construction Special Work Procedure BC-SC-8, "Installation of Reactor Cavity Back-up Pool Seal" along with NUSCO engineering calculation 84-110-315GP "Redundant Seal Leak Rate and Structural Integrity Calculation" were both reviewed by the inspector. These documents provided the detailed instructions for assembly of the back-up seal along with the requirements for measuring the leakage gaps in the back-up seal. Because of problems encountered with excessive clearances in the outer seal ring back-up seal, this procedure was supplemented with DCN 269-84 which attempted to reduce the back-up seal channel to plate gap with 3/16" x 1 3/4" rolled bar. Also, DCNs 279-84 and 281-84 removed certain back-up seal leakage clearances where overlapping closure plate surfaces were sealed with RTV. Upon completion of the back-up seal installation, the licensee confirmed the area for leakage by taking feeler gauge readings of the available gaps between the seal sections and the seating plates. The inspector observed the gap measurement in process and verified the correct transmittal of leak area data. Based on the available gap measurements and the obstruction of certain areas using RTV verified capable of withstanding full cavity hydrostatic pressure, the licensee judged that the minimum leak rate criteria had been established. No unacceptable conditions were identified.

6.4 Quality Control Inspection

The inspector reviewed the data available for the visual weld inspections performed in accordance with procedure QCI-CY-10.02. Weld data cards and NUSCO Construction QC Inspection Reports 84-110-001 through 029 were available for the seismic alignment fixtures, and back-up seal attachments, the seal ring bearing plate, and the fuel transfer canal waterstop field welding. There were no unacceptable conditions identified.

The inspector also reviewed the QC Nonconformance Reports (NCR) issued by the station QC group and those issued by NUSCO Construction QC for these modifications. The following NRC's were reviewed 84-191, 192, 193, 225, 226, 227, 228, 229, and CY-84-075, 077, 078, 079, 080, 083, 084, 086, 087, 088, 090, and 092. The inspector verified that the administrative controls for resolution and disposition of these NCRs were being followed and that the findings for cause and prevention of recurrence appeared accurate. There were no unacceptable conditions identified.

6.5 Wetted Equipment Evaluation

The licensee performed a complete inspection of the affected containment areas to identify all components and structures which had been wetted during the seal failure event. An evaluation of each exposed item was completed and corrective action was completed. This program and the results were detailed in the licensee's submittal to NRC Licensing on October 16, 1984. The inspector reviewed the licensee inspection procedure, NU SAE-1, Revision 0. Independent inspection of the wetted areas and repair work confirmed the results documented in the licensee's submittal. The inspector verified that two open issues, inspection of the containment liner and threaded conduit connections have been incorporated in outage schedules for the next refueling. The inspector had no further questions in this area.

7.0 Licensee Event Reports (LERs)

The following LERs were reviewed to verify that the details of the event were clearly reported, including the accuracy of the description of the cause and the adequacy of the corrective action. The inspector determined whether further information was required, and whether there were generic implications. The inspector also verified that the reporting requirements of Technical Specifications and Station Administrative and Operating Procedures had been met, that appropriate corrective action had been taken and that the continued operation of the facility was conducted within Technical Specification Limits.

- 84-06 Fire Door Latch Inoperable
- 84-07 Degraded Cable Penetration Fire Barriers
- 84-08 Inoperable Fire Door
- 84-09 Total Loss of Offsite Power/Reactor Trip - event detailed in NRC Inspection Report 50-213/84-19
- 84-11 Containment Integrated Leak Rate Test Fialure - event detailed in NRC Inspection Report 50-213/84-25
- 84-12 Containment Local Leak Rate Test Failure - event detailed in NRC Inspection Report 50-213/84-25
- 84-13 Reactor Cavity Seal Failure - event detailed in NRC Inspection Report 50-213/84-23 and in paragraph 6 of this report
- 84-14 Loss of Offsite Power/Emergency Diesel Generator 2A Failed to Pickup Load - event detailed in NRC Inspection Report 50-213/84-22.

8.0 Post Accident Sample System Isolation Valves

During a review of post accident sample system (PASS) operation, the inspector identified a potential problem with the operation of the PASS when containment integrity is established. The containment air sample and return line penetrations are isolated by redundant remotely operated manual valves. In order to take a PASS air sample, these valves must be open. Technical Specification (TS) 3.11 however, requires manual containment isolation valves to be closed when reactor coolant temperature is greater than 200° F.

Although NRC standard technical specifications have allowed manipulation of manual containment isolation valves under administrative control, the licensee had not requested or been granted such relief. The inspector determined that the PASS isolation valves had in fact been opened multiple times for test samples since October 1983, with no identification of the potential violation of TS. Upon notification, licensee management required the valves to be tagged closed until the technical aspects of this problem are resolved. At the conclusion of the inspection the licensee had determined that several submittals to NRC had documented the existing condition of the PASS isolation valves, but no NRC concurrence or revision of TS 3.11 had been identified. Since the isolation valves were opened for short periods under approved procedural control and this practice has been routinely approved for operation of manual containment isolation valves, the inspector determined that there was minimal safety significance associated with this potential violation. The inspector stated that this item would remain unresolved pending the licensee and NRC determination of the correct containment isolation requirement for these penetrations. (213/84-14-05)

9.0 Unresolved Items

Unresolved items are matters about which more information is required in order to determine whether they are acceptable items or violations. Unresolved items identified during this inspection are discussed in paragraph 8.0.

10.0 Exit Interviews

During this inspection, meetings were held with plant management to discuss the findings. No proprietary information related to this inspection was identified.