U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee: Public Service Company of New Hampshire,
New Hampshire Yankee (NHY) Division

Facility: Seabrook Station, Seabrook, New Hampshire

Dates: December 24, 1991 - January 27, 1992

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Approved By: William J Lazarus 2/11/92
William J. Lazarus, Chief, Reactor Projects Section 3B Date

OVERVIEW

The Operations Department responded to equipment failures in a safe manner which minimized the severity of plant transients. The addition of a licensed Senior Reactor Operator to each operating shift enhanced overall plant operation and safety.

Maintenance activities were well supervised and documented. Good communications were exhibited within the department and with other organizations, which enhanced system availability.

Health Physics technicians controlled the spread of contamination, monitored radiation levels, and implemented the ALARA program. The Security Department continued to safeguard the site by maintaining detection equipment, conducting routine patrols, and implementing the Fitness-for-Duty program.

Reactor Engineering installed a new calorimetric program which enabled operators to more precisely control reactor power. Technical Support and Engineering personnel developed contingency plans prior to conducting a containment local leak rate test.

NHY responses to weld issues were appropriate. Present records fully substantiate the adequacy and safety of welds.

The Nuclear Quality Group conducted detailed indepth audits of administrative controls required by the Technical Specifications.

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DETAILS

1.0 SUMMARY OF ACTIVITIES

1.1 NRC Activities

Two resident inspectors were assigned. Backshift inspections were conducted on 12/30, 1/8, 1/9, 1/10, 1/14, 1/15, 1/21, 1/22 and 1/23. Deep backshift inspections were conducted on 1/12, 1/15, 1/20, and 1/23.

On January 13-16, an Emergency Preparedness region based inspection occurred. The results will be documented in Inspection Report 50-443/92-01.

1.2 Plant Activities

The plant operated at 100% until December 23, 1991 when power was reduced to 30% due to an incursion of sulfates into the steam generators. Proper chemistry conditions were restored in the steam generators and power was returned to 100% on December 25.

On January 7, 1992, reactor power was reduced to below 90% due to a loss of plant efficiency caused by tripping of a non-vital electrical bus (principally as a result of loss of feedwater heaters). The electrical bus was energized and power was raised to 100% the same day.

2.0 OPERATIONS

2.1 Plant Tours/Observations of Activities

The inspector observed shift turnovers, discussed operating conditions with the operations staff, reviewed work schedules, and attended Station Operation Review Committee and daily Plan of the Day meetings. The shift turnovers were thorough and included representatives of other departments. The operators were knowledgeable of current plant conditions and planned evolutions; Shift Superintendents and Unit Shift Supervisors conducted independent plant inspections. Staffing changes were implemented that provided 24 hour coverage of the work control supervisor position adding another licensed Senior Reactor Operator to each shift. The inspector considered the staffing changes enhanced overall plant safety by improving coordination of maintenance activities and by increasing the number of licensed operators available to respond to transients.

The inspector determined that the daily planning meetings clearly identified work priorities, provided an understanding of the basis of the priorities, and fostered coordination and cooperation between departments. The extended work control coverage represents a positive initiative.

On December 26, 1991 the "A" service water pump discharge valve stopped moving before reaching the full open position during a pump start. The Operations Department entered the limiting condition for operation of Technical Specification 3.7.4., "Service Water System," which required that the service water pump be restored to service within seven days. The entry into the action statement was correct and timely; the discharge valve is interlocked with the pump controls. Repairs were completed, operability testing performed, and the Technical Specification action statement exited within two days.

2.2 Power Reduction Due to Secondary Water Chemistry

On December 23, 1991 a manual power reduction was initiated due to secondary water chemistry limits increasing to chemistry program action levels. Power level was reduced to 30% in accordance with procedures. The increase in steam generator cation conductivity and sulfate concentration occurred following the shifting of the steam generator blowdown demineralizer trains.

The licensee investigated the cause of the event and concluded the appropriate procedures were followed. A procedure step specified a flush of 1300 gallons followed by a chemistry sample. Personnel calculating the time to flush the return line to the condenser used an incorrect flow rate. After the flush was completed and the chemistry sample indicated acceptable chemistry, the newly regenerated demineralizer train B, was placed into service. Due to procedural ambiguity, flow from the "A" demineralizer was incorrectly included in the flush volume. Therefore there was inadequate flow to completely flush the "B" demineralizer line and sulfates from the regeneration process entered the condenser.

Short term corrective actions included counseling of responsible personnel and implementation of a procedure change which specified the flush of the steam generator blowdown line for a minimum time period assuming a minimum flow rather than a specific volume. Long term actions under evaluation include modification of the steam generator blowdown system hardware and installation of Design Change Request, DCR 91-0002, CA-01 which would add an automatic isolation trip of the blowdown valve when chemistry levels were above acceptable limits.

On February 28 and April 10, 1991, other events occurred in which sulfuric acid was injected into the steam generators. The system modifications and procedure revisions made following these events were not comprehensive enough to prevent the sulfate incursion from this event.

2.3 Electrical Bus Deenergization

Early on January 7, the secondary supply breaker to nonvital bus 13 tripped when generator stator cooling pump A was started as part of a surveillance test. Motor Control Centers (MCC)-131 and 132, which supply power to secondary feedwater heater controls, were deenergized. The affected feedwater heater controls assumed their deenergized positions which reduced the efficiency of the feedwater heat exchangers.

The reactor thermal power increased as a result of the cooler feedwater flow and control room operators initiated a rapid power reduction. The resulting rod insertion caused the Axial Flux Difference (AFD) to be outside of the target band. Power level was reduced to less than 90% power in accordance with the Technical Specification 3.2.1.(a) 2, "Axial Flux Difference."

Plant response was per design. The third air removal pump started after the second air removal pump was deenergized by the loss of Bus 13. Bus 13 and the affected MCCs were manually stripped of all loads and energized within hours of the initiating event. Selected loads were placed on the bus and the unit was returned to full power.

The operations response included rapid assessment of plant conditions and adherence to Technical Specification requirements. An event evaluation team was formed to review the event. A work request was written to determine why the supply breaker to Bus 13 tripped, with initial efforts focused on evaluating the generator stator cooling pump. Concurrently, Technical Support engineers evaluated the use of the cross tie between Bus 13 and Bus 26, including a review of bus loading. The evaluation determined that the use of the cross tie was acceptable for the existing plant conditions.

Later the same morning, the secondary supply breaker to the non vital Bus 13 tripped again when a condenser air removal pump automatically started. The operators responded to the potential thermal power increase by reducing power to 96%. Within three minutes, operators closed the cross tie between Bus 13 and Bus 26 which restored power to the feedwater heater controls and minimized the feedwater temperature transient.

The secondary breaker for Bus 13 was removed and tested. The cause of the breaker trip was determined to be stripped threads on a brass adjustment screw for the "C" phase short term overload device. The stripped threads permitted the screw to turn thus allowing the short term overload setting to drift low. Consequently, the starting current of the pump motors exceeded the "C" phase short term overload setting. The other adjustment screws on the breaker overload device were found to be tight. The overload device was replaced and the breaker returned to service.

The operators' response to the second transient was prompt and illustrated the positive impact of timely Technical Support engineering review and communication with operations. Planning and Scheduling activities were well coordinated during the investigation of the cause of this electrical failure and reflected a strong safety perspective. Activities were postponed that had the potential to impact the loading of Bus 13 or Bus 26, and maintenance was deferred on backup equipment that might be used if the transient reoccurred.

The inspector observed portions of the equipment troubleshooting and operations response. The transient response actions, activities, and priorities were well controlled and prudently implemented. Activities and communications between departments were effective.

2.4 Engineered Safety Feature System Walkdown

The inspector conducted a walkdown of the Solid State Protection System, Primary Component Cooling Water (PCCW) system, and the Engineered Safety Features Actuation System using check lists from EGG-EA-7/94, "Probability Risk Assessment Applications Program for Inspection at Seabrook Station, Draft Report." The inspector verified proper valve and switch alignments for accessible equipment and reviewed the latest valve lineup sheets for proper valve position of inaccessible valves. The inspector verified proper calibration of selected temperature, flow, and pressure detectors on selected instruments on the PCCW system by reviewing the latest calibration procedures. No deficiencies were noted.

The PCCW checklist included the safety related Primary Auxiliary Building (PAB) auxiliary air handling system. The system provides a backup to the normal PAB air handling system which is non-safety related. The systems are designed to maintain temperature in the area of the PCCW pumps below 118°F. A minor modification, MMOD 90-571, installed on June 18, 1991 raised the setpoint of the actuation of the PAB auxiliary air handling system from 90°F to 100°F in order to reduce the run time of the system fans. The calibration procedure for the thermostat was performed when the MMOD was installed. Maintenance procedures were conducted to verify fan motor winding continuity and to visually inspect mechanical components. System actuations occurred in June 1991 when room temperature reached the actuation setpoint. However, no surveillance test existed for verifying system operability.

The inspector reviewed existing calibration and surveillance procedures for the PAB auxiliary air handling system, evaluating MMOD 90-571 and the associated 50.59 review, and held discussions with station personnel. The Technical Support Department planned to evaluate the need for a surveillance test to verify system operability. The inspector concluded that the licensee was taking appropriate actions.

2.5 Management of Overtime - NOV 91-32-01 (Closed)

During the week ending August 14, 1991, two licensed operators did not receive Station Manager's documented authorization prior to working more than 72 hours within a seven day period, which was a violation of requirements in Technical Specification 6.2.2(e). New Hampshire Yankee (NHY) responded to the Notice of Violation in a letter (NYN-92002) issued on January 6, 1992. NHY determined that the reason for the violation was an administrative error by the Operations Manager. The individuals involved were counseled and a review of the shift schedule requirements for the second refueling outage was initiated.

The inspector noted that the use of overtime since the completion of the refueling outage was minimized. The inspector concluded that the long term corrective action of evaluating the refueling shift schedules to minimize the need for overtime in excess of station guidelines was appropriate. This violation is closed.

3.0 RADIOLOGICAL CONTROLS

The inspector reviewed the postings at the Health Physics Control Point, toured the radiologically controlled areas, and verified the status of radiation and contamination controls. Areas toured included the fuel storage building, waste disposal building, primary auxiliary building, and turbine building.

Standing Radiation Work Permits (RWP) for the new calendar year were revised and posted on January 1. The Health Physics personnel used the Plan of the Day meeting and notices at the control point to inform personnel of the changes in Standing RWPs. Health Physics postings were positioned locally so there was no confusion as to the radiation or contamination of the area to be entered. Requirements for entry and exit were clearly stated. Portable radiation monitors and air sampling equipment had current calibration information documented.

On December 24, 1991, a copy of the technical training student handout, Seabrook Station Radiation Worker Qualification, was distributed to radiation workers. The radiation worker requalification training was based on material contained in the handout. The licensee plans to provide the handout to new personnel during initial radiation worker training. The handout contained numerous practical and relevant examples to illustrate good radiological work practices. The Radiation Worker requalification training course was computerized to provide a self-paced review of radiological hazards, procedures, and principles.

A maintenance work request, 91W005983, addressed repair of the fuel pool skimmer pump. The inspector reviewed the associated Radiation Work Request, and observed the radiological and contamination practices associated with the disassembly, cleaning, and temporary storage of the pump components. The pump was disconnected and relocated to the radiologically controlled maintenance shop for decontamination and repair. The Health Physics Technician was knowledgeable of the exposure rates and contamination levels, and maintained adequate controls for the job scope. The maintenance personnel demonstrated good radiation worker practices. The relocation of the equipment to the maintenance facility had ALARA (as low as reasonably achievable) benefits and removed the workers from a high noise environment. The monitoring equipment used was calibrated and periodic monitoring was performed. Temporary storage of components was deliberate and controlled.

The inspector concluded that radiological controls were effective in ensuring a safe working environment and were implemented with ALARA consideration.

4.0 MAINTENANCE/SURVEILLANCE

4.1 Maintenance

The inspector observed work activities and reviewed work request 91W006287 for maintenance on the discharge valve for Service Water Pump A that failed to open fully following the starting of the pump. The work request documentation was comprehensive; the initial electrical maintenance activities were described with as found conditions noted, and a scope change was initiated when a packing adjustment was required. Extensive planning and prestaging were performed in preparation for the mechanical repairs. The inspector observed a portion of the mechanical maintenance effort at the job site; ample maintenance personnel and supervision, including the Maintenance Manager, were involved. The inspector observed the work area after the maintenance was completed, housekeeping was excellent in the area with scaffolding removed.

The inspector concluded that the maintenance activities were well staffed and controlled with sufficient documentation to reconstruct the work activities. The repairs were completed promptly.

On January 10, the Main Control Room received numerous simultaneous electrical alarms associated with DC Bus 11E. Electrical Maintenance and Technical Support personnel diagnosed the failure of a blocking diode transformer on Inverter 2A and a failed capacitor on the underfrequency relay for the "C" Reactor Coolant Pump, both are supplied from DC Bus 11E. Operations tripped the bistable and entered the Technical Specification Action Statement for the underfrequency relay while a replacement was obtained and installed. The loads from Inverter 2A were supplied by the normal maintenance power supply until the transformer was repaired. The inspector confirmed that the tripped bistables and alarms in the Main Control Room were consistent with the equipment failures.

On January 10, while performing the quarterly surveillance procedure OX 1430.04, "Main Steam System Valve Operability Tests," Atmospheric Steam Dump Valve (ASDV) D failed to stroke using the manual/auto controls of the instrument air positioner. After replacement of solenoid (MS-PY-3004-2), which ports instrument air to the bottom of the actuator, the valve was retested. The valve stroked intermittently. Further investigation identified a slide link for the 125 VDC supply to the solenoid was loose and making intermittent contact. The slide link was tightened, the surveillance test was completed, and the valve was returned to service.

The inspector reviewed work request, 92W-000128, studied ASDV control systems, and held discussions with the I&C Department Manager. No records of manipulating the slide link were found prior to April 1990, when the slide link was verified closed by two individuals. ASDV D had passed previous surveillance tests and was able to be positioned using the nitrogen gas backup. The root cause of the failure of ASDV D will be further evaluated.

4.2 Surveillance

The monthly control rod operability test was initiated on December 30 using procedure OX1410.02, "Monthly Rod Operability Check." Data gathering for a previous rod control malfunction immediately preceded the performance of this test with satisfactory results documented on work request 91W005664. Good communication and control was demonstrated in the prejob briefing conducted by the Shift Supervisor prior to the start of control rod manipulations.

During the performance of the test, the expected response was not received for one shutdown bank; the surveillance was halted, and the action statement for Technical Specification 3.1.3.1, "Movable Control Assemblies Group Height," was entered while Instrumentation and Controls (I&C) personnel investigated. When replacement circuit boards did not provide the desired response, I&C identified the individual integrated circuit board component that was faulty, replaced that component, and completed the surveillance test satisfactorily. The licensee returned the replacement circuit boards to the vendor for evaluation.

The inspector observed portions of the troubleshooting and held discussions with I&C personnel. The inspector determined that the actions taken by Operations, Reactor Engineering, and I&C personnel were controlled, and demonstrated excellent communications between departments.

During September 1991 with the plant shutdown for refueling, diesel generator B was tested following an 18-month overhaul. Two valid test failures were reported and the diesel was manually shutdown to prevent equipment damage. The test failures were described in a Special Report submitted on October 11, 1991 to the NRC as required by Technical Specifications 4.8.1.1.3 and 6.8.2. Diesel generator B was tested on a weekly basis in accordance with the diesel generator test schedule in Technical Specification Table 4.8-1. On January 19, 1992, after nineteen successful starts, the monthly testing frequency of diesel generator B was resumed. The inspector noted that the Special Report was submitted in a timely manner and testing was performed as required.

The inspector monitored the unit auxiliary and reserve auxiliary transformers on several occasions. All local indications were operable and within normal bands and no equipment discrepancies were observed. The inspector independently performed procedure OX1446.07, "Containment Loads Circuit Breaker Monthly Verification." All components were verified in the proper configuration. Placards on the breaker compartments specify the applicable Technical Specification.

The inspector concluded that surveillance activities were performed in accordance with the established test schedule to assure safe system performance.

5.0 SECURITY

The inspector toured the protected area boundary, noted that compensatory measures were inplace, observed guards on patrol, and evaluated protected area lighting. During a backshift inspection, safeguards cabinets outside the protected area were verified locked or located behind locked doors. The inspector noted that snow removal was accomplished without affecting protected area security. The inspector reviewed the logs in the Central Alarm Station (CAS), and noted that appropriate compensatory measures were taken. The equipment in CAS and the Secondary Alarm Station was in excellent operating condition. The personnel manning the stations were knowledgeable of job responsibilities, aware of plant conditions, and cognizant of planned activities that had a potential impact on security. The inspector observed checks of the intrusion system; security personnel demonstrated proficiency regarding equipment response and limitations. No deficiencies were noted.

The inspector reviewed the actions taken by NHY as a result of a random Fitness-For-Duty (FFD) test failure by an individual associated with safety-related and security equipment. The individual's site access was suspended for two weeks and the individual was referred to the Employee Assistance Program. The individual's supervisor reviewed the work the individual performed over the previous 90 days and reviewed the performance of the affected equipment. All safety-related work was performed with a second individual or under the direct observation of a NHY Quality Control inspector. No associated equipment problems were noted from either operating performance history or surveillance tests. Neither the individual's supervisor nor security personnel, who interface with the individual, noted any behavioral abnormalities prior to the FFD failure. The inspector concluded that the FFD program was properly implemented and that an adequate review of the individual's work was performed.

6.0 ENGINEERING/TECHNICAL SUPPORT

6.1 Steam Flow Calorimetric

During prolonged operations at 100% power, the electrical output of the turbine generator decreased as the calculated reactor thermal power was maintained at 100%. Based on other plant parameters, industrial experience, and inspections done during the refueling outage, Technical Support engineers determined that fouling of the feedwater flow orifices was resulting in an erroneously higher calculated value of reactor thermal power. The Reactor Engineering Department revised the calorimetric computer program to use either steam flow or feedwater flow for calculating reactor thermal power.

The inspector reviewed the new computer code, the 50.59 evaluation of the new code, and the engineering calculation (SBC-484) for the uncertainty of the steam flow calorimetric. The inspector held discussions with Reactor Engineers, performed independent calculations, and observed the installation of the steam flow calorimetric program.

The computer code contained well documented steps. A function code description and a program description were written. The function code description and the results of over 40 validation test cases, were reviewed by the Reactor Engineering Department Manager. The code was installed in a test bed computer and tested. Once the code was installed in the main plant computer, the steam flow option was not selected until steam flow and feedwater flow calorimetric values were observed for over two days.

The revised code calculated steam flow values based on pressure corrected differential pressure readings which were normalized using feedwater flow and steam flow values derived from the precision calorimetric performed after the refueling outage. The code calculated a reactor thermal power for both feedwater flow and steam flow. Both values can be displayed by the Main Control Room Operators. An automatic feature, which may be selected, displays the steam flow calorimetric value on the digital display whenever the feedwater flow calorimetric value is above 90% power.

The 10 CFR 50.59 evaluation concluded that the calorimetric uncertainty using the feedwater flow was +/- 0.95% of reactor thermal power and using the steam flow was +/- 1.22% of reactor thermal power. The increased uncertainty of the steam flow calculation resulted from the uncertainties introduced by steam pressure corrections used to develop steam flow readings. The evaluation concluded that both calorimetric methods were acceptable since they met the +/- 2% of reactor thermal power uncertainty assumed in the accident analysis in FSAR Section 15.0.3.2(a).

The inspector used an NRC computer program "TPDWR2: Thermal Power Determination for Westinghouse Reactors, Version 2," to independently verify the licensee's calculation. The independently calculated reactor thermal power using feedwater flow prior to the installation of the new code was approximately 0.6% higher than the licensee calculated value. After the installation of the steam flow code the independently calculated reactor thermal power using feedwater flow was approximately 1.5% higher than the licensee calculated value. The inspector concluded that the results of the steam flow code were in agreement with the independently calculated values.

The inspector noted that the four minute average of reactor thermal power stabilized after installation of the steam flow program. The instantaneous values calculated over a two minute period using steam flow varied by less than three megawatts while the values calculated using feedwater flow varied by over forty megawatts. The operators believed the difference was caused by steam flow being steady due to the constant turbine generator demand and the feedwater flow varying due to the oscillations of the feedwater regulating valves in response to the steam generator level control system.

The inspector concluded that the steam flow calorimetric development, documentation, review, and installation were well controlled and resulted in the ability to more finely control reactor power.

6.2 Local Leak Rate Testing

The inspector observed the planning efforts and the conduct of local leak rate testing of the Containment Area Purge (CAP) exhaust valves. The CAP supply and exhaust valves were last tested on October 4, 1991 after they were closed following the refueling outage. The combined leakage of the exhaust valves was found to be acceptable but was increased over the previous leak rate. The exhaust and supply valves are required to be tested every six months on a staggered test schedule.

In preparation for this test, a temporary test flange was installed to enable engineers to quantify leakage through each individual valve. A 10 CFR 50.59 evaluation was performed to support installation of the temporary modification and revisions to procedure EX 1803.003, "Reactor Containment Type B & C Leakage Rate Tests," were approved to support the new test method. A Technical Specification Clarification was approved by the Station Operation Review Committee. The clarification defined the method by which leakage could be determined and the actions required when valve leakage limits were exceeded. A blank flange was manufactured and a seismic analysis was performed to support installation of the flange if both valves failed the leak rate test. The Technical Support Department plans to evaluate the installation of blank flanges in the CAP system after every refueling outage to negate the need to conduct the leak rate tests during power operations.

An effective pretest briefing was conducted by the Program Support Manager. The brief was attended by personnel involved in the test from the Operations, I&C, Health Physics, and Technical Support Departments. The Shift Supervisor and Unit Shift Supervisor understood what actions to take and Emergency Action Levels to declare for all possible outcomes of the leak rate test. The test was well coordinated and performed in an expeditious manner. Test results showed that the leakage was acceptable and was less than the total leakage rate measured after the refueling outage.

The inspector reviewed the Technical Specification Clarification, the 10 CFR 50.59 review for the temporary modification, the changes to test procedure EX 1803.003, and the final test results. The inspector verified the instruments used for the leak rate test were calibrated and observed the engineers in the performance of the test. The inspector concluded that extensive engineering reviews were performed by the Technical Support Department to develop contingencies for the leak rate test results and to ensure the safety of containment integrity.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

7.1 Image Sensitivity of Six Radiographs - NOV 91-12-002 (Closed)

On June 28, 1991, a Notice of Violation (NOV) was issued for six radiographs of safety related field welds that the NRC determined did not obtain the minimum sensitivity required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. New Hampshire Yankee (NHY) replied to the NOV in a letter (NYN - 91106) issued July 8, 1991. NHY questioned the NOV, but committed to perform additional radiographic examinations of the welds in question, review the resulting radiographs, and retain the new records as a supplement to the existing records.

The NRC conducted an independent review of the ASME code acceptability of the supplemental records for the six welds and documented the results in NRC Inspection Report No. 50-443/91-21. The report concluded that the actions taken by NHY in answering and correcting the violation were satisfactory. This violation is closed.

7.2 Weld Radiograph Review Program - NOV 91-33-01 (Closed)

In a letter dated November 25, 1991, the NRC issued a Notice of Violation and a Proposed Imposition of a Civil Penalty to New Hampshire Yankee (NHY) for obtaining insufficient radiographs to furnish evidence that all welds met the quality standards required by ASME code. The licensee and an NRC approved independent Level III reviewer, working as a contractor to

NHY, identified 47 weld radiographs which did not meet the sensitivity and/or density requirements of the ASME code. Three radiographs were rejected for sensitivity, 40 were rejected for density, and four were rejected for both sensitivity and density.

NHY responded to the NOV in a letter (NYN 91196) issued December 23, 1991. NHY did not contest the violation but concluded that the original Pullman-Higgins and YAEC reviews of seven radiographic packages, with respect to sensitivity concerns, were acceptable per the code. The 44 radiographic packages rejected by the Weld Radiograph Reinterpretation Program (WRRIP) with respect to density concerns were not questioned by NHY.

All the weld radiographs addressed by the NOV were performed using a difficult radiographic technique on three inch nominal pipe size or smaller lines where initial Pullman-Higgins radiographer review signatures occurred prior to October 1, 1982. All radiographs which had these common factors were reviewed by the WRRIP. NHY reradiographed and interpreted all 47 of the subject welds promptly following identification. The independent Level III reviewer reviewed all the new radiographs and found no deficiencies.

The NRC inspected the WRRIP and independently reviewed a sample of the new radiographic packages. The inspections and reviews were documented in NRC Inspection Reports Nos. 50-443: 91-19, 91-21, and 91-29. The reports concluded that the responses of NHY were appropriate for the problems that faced them and the program used was acceptable. No further analysis of the radiographs need be undertaken. This violation is closed.

7.3 Missing Radiographic Records - NOV 91-33-02 (Closed)

On December 27, 1990, the NRC senior resident inspector was informed by New Hampshire Yankee (NHY) that radiographic film for one specific weld could not be found during a search of Chemical Volume and Control System (CS) welding records. NHY also found two radiographic record packages that did not contain evidence of a prior Yankee Atomic Electric Company (YAEC) independent review. NHY designed and implemented the Weld Record Reverification Program (WRRP), which identified an additional three missing radiographic record packages.

In a letter dated November 25, 1991, the NRC issued two Notices of Violations (NOVs) to NHY. One NOV was for the four missing radiographic packages and the second NOV was for the two radiographic records which did not contain evidence of YAEC review. NHY responded to the violations in a letter (NYN-91196) issued December 23, 1991. NHY attributed the missing radiographs and signatures to personnel error on the part of Pullman-Higgins records management and YAEC quality assurance overview personnel during plant construction.

The WRRP was completed. The four welds without complete documentation were reradiographed and interpreted in accordance with current NHY program requirements. The two weld record packages that did not receive YAEC review were independently read and interpreted by a YAEC Level III reviewer. Any radiography performed on future modifications will be conducted under the control of the present Operational Quality Assurance Program. NHY is in the process of evaluating the radiographic process and will incorporate program enhancements as a result of lessons learned and experience gained during the first refueling outage. (See section 7.4)

The NRC inspected all aspects of the WRRP and independently reviewed the radiographic packages of concern. The inspections and reviews were documented in NRC Inspection Reports Nos. 50-443: 91-04, 91-09, 91-12, 91-15, 91-19, 91-21, 91-22, and 91-29. The reports concluded that the safety related radiographs now in the record vaults are in compliance with the requirements for radiography and weld integrity. These two NOVs are closed.

7.4 Incomplete Work Request Package

The inspector reviewed Operational Information Report No. 91-057, which determined the reason that radiographic film was not with a completed work package. On October 23, 1991, the Technical Support Department identified that the radiographic film and Radiography Inspection Report (RIR) for a safety related weld performed during the outage were not included with the support documents. The weld was performed during the replacement of a relief valve on the reactor coolant pump seal return line. The radiographic film and RIR were located in the Nuclear Quality Group's fire proof cabinet and all required reviews were conducted and documented before the work package was formally submitted to the Document Control Center (DCC) vault.

The inspector verified that the work package was in the DCC vault and that all required reviews were properly documented. New Hampshire Yankee determined that the lack of timely submittal of the radiographic film and RIR to the Document Control Center was caused by incomplete film review documentation, signoff of hold points based on incomplete information, and a lack of specific requirements on when to submit radiographic film to the vault. The Nuclear Quality Group was assigned the responsibility to recommend revisions to program procedures to clarify radiographic film turnover requirements. The inspector had no further questions.

7.5 Nuclear Quality Group Review

The inspector reviewed the Nuclear Quality Group (NQG) audits with respect to audit documentation for Technical Specification Section 6, "Administrative Controls." Licensee documents considered during the inspector's evaluation are provided as Appendix A. Recent audit reports were reviewed, including the response to and closure status of findings and observations for select audits. Training records were reviewed for a sample of NQG personnel. The inspector discussed the NQG audit program philosophy and implementation with NQG management, auditors, and representatives of groups routinely audited. The inspector attended an NQG Finding Review Board meeting.

Good progress was made in the area of Technical Specification Section 6 audit documentation since the last NRC review documented in NRC Inspection Report 50-443/91-01. The process has continued to evolve to incorporate audit coverage from discrete and global audits with minimum duplication of effort.

NQG Management described the use of the preaudit briefing as a vehicle to establish a dialogue with the group being audited and to provide a forum for concerns or requests from the audited group. The inspector confirmed, through discussion with representatives from the Maintenance and Engineering Departments, that the audited groups concur in the use and value of preaudit briefings.

The depth and scope detailed in the audit reports indicated that the audits were performed by personnel with technical expertise in the subject areas. NQG auditors participated in technical staff continuing training programs and in specialized training courses. The exchange of technical specialists as part of the Yankee In Plant Audit program added unique expertise to the station audits, and provided station personnel an opportunity to broaden their knowledge and experience by participating in audits at other facilities. Management support of the Yankee In Plant Audit program has been excellent.

The NQG group demonstrated good initiative in the area of integrated Refueling Audit implementation. The Refueling Outage Report provided a useful summary for planning and comparing future refueling outages. The NQG group recently initiated the program coordinator concept which assigns responsibility by functional area. The NQG did not duplicate audit efforts that were completed by other groups, for example, the Independent Safety Evaluation Group conducted an indepth review of the Emergency Operation Procedures.

The majority of responses to NQG Audit findings and observations were well developed and timely. Overdue responses to findings were identified to management. Corrective actions were implemented in a timely manner.

Reviews performed as required by Technical Specification Section 6 were thorough. For example, an NSARC subcommittee review identified that procedure ON1034.03,"Condensate System Operation," was implemented in a configuration that may not have met the high energy line analysis of the Final Safety Analysis Report. An engineering review was completed to confirm that the FSAR criteria were met and the results will be reported to NSARC.

The inspector concluded that the NQG organization implemented the audit program for Technical Specification Section 6.0 requirements in a manner to assure and promote overall safety.

7.6 Missing BISCO Seal

On August 5, 1991 during an inspection of safety related seals, a flexible boot seal for a penetration in the cable spreading room was found to have not been installed. The seal provided a tornado barrier. A Station Information Report (91-023) and an engineering evaluation for reportability of the missing seal were prepared. The inspector reviewed the report and evaluation. The inspector determined that proper attention was given to the missing seal and that a report to the NRC was not required.

7.7 Reactor Coolant System Low Flow Trip - LER 91-009 (Closed)

On July 4,1991, with the plant at 100% power, a reactor trip/turbine trip occurred. The reactor trip was in response to a reactor coolant pump motor fault. The reactor trip was described in NRC Inspection Report 50-443/91-19. The Licensee Event Report (LER) was submitted within 30 days of the event and included the relevant information. The LER specified that additional inspection of electrical terminal connector boxes would be completed during the first refueling outage. Inspection Report 50-443/91-22 described the results of the connector box inspection. This item is closed.

7.8 Inoperable Source Range Audible Indication - LER 91-011 (Closed)

On August 10, 1991, during performance of core alterations, the operators discovered that the Source Range Neutron Flux Audible Monitor had not been operating for one hour. The inspector's review of the event was documented in NRC Inspection Report No. 50-443/91-22. The LER presented the required information and was issued within the required time period. The inspector determined that the information was accurate and that the corrective actions were adequate to prevent recurrence. This LER is closed.

8.0 MEETINGS

The scope and findings of the inspection were discussed periodically throughout the inspection period. An oral summary of the inspection findings were provided to the Plant Manager and his staff at the conclusion of the inspection period.

A region-based inspector conducted the following exit meeting during this report period.

DATE	SUBJECT	REPORT NO.	INSPECTOR
1-17	Emergency Preparedness	92-01	C.Conklin

APPENDIX A

NHY DOCUMENTS REVIEWED DURING NUCLEAR QUALITY GROUP REVIEW

"Audit Finding/Observation Update", 7/10/91, 9/4/91, & 1/2/92
"Control Room Command Function Directive", 1/8/92
"Evaluate Updates of EOPS", ISEG Document, 7/8/91
"Finding Review Board Meeting 92FRB01", 1/27/92
"Joint Utility Management Audit (JUMA) - 1991", 9/19/91 and response dated 10/30/91
"Joint Utility Management Program Audit Policy", Rev 4, 9/20/91
"Minutes of NHY Nuclear Safety Audit and Review Committee Meeting # 91-13", 12/24/91
"NHY Policy Concerning Archived Documentation Packages", 10/17/91
"1992 Seabrook Station Quality Assurance Audit Schedule", 7/29/91
"NSARC ELS Meeting No. 91-04", 12/9/91
"Program Coordinator Responsibilities", 1/9/92
"Status Update for QA Audit No. 90-A08-01 Fire Protection", 11/27/90
QAAR No. 91-A03-01, "Emergency Preparedness", 5/14/91
QAAR No. 91-A03-02, "Electrical Configuration/Design Control", 5/29/91
QAAR No. 91-04-01, "Security". 5/17/91
QAAR No. 91-A05-01, "Technical Specifications", 6/11/91
QAAR No. 91-A05-02, "Corrective Action Program", 8/6/91, and responses dated 8/12/91, 9/10/91,& 1/23/92
QAAR No. 91-A06-01, "Generic Letter 89-13", 4/24/91
QAAR No. 91-A07-01, "Performance, Training and Qualification Audit", 12/24/91
QAAR No. 91-A07-02, "Refueling Outage" , 1/3/92
QAAR No. 91-A08-01 "Fire Protection (Annual)", 10/30/91
QAAR No. 90-A09-03, "QA Records", 11/27/90
QAAR No. 91-A10-02. "Radiological Environmental Monitoring and Effluent and Environmental Monitoring", 12/24/91
QAAR No. 90-A10.03, "ALARA", 12/20/90
QAAR No. 91-A11-02, "Corrective Action", 12/18/91

QAAR= Quality Assurance Audit Report