3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents at the point of discharge from the multiport diffuser (see Figure 5.1-3) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents at the point of discharge from the multiport diffuser exceeding the above limits, restore the concentration to within the above limits within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program specified in Part A of the ODCM.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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ATTACHMENT B TO NYE-92005

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

RECEIVED

NOV 07 1991

LICENSING

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Docket No. 50-443

Public Service Company of New Hampshire
ATTN: Mr. Ted C. Feigenbaum
President and Chief Executive Officer
New Hampshire Yankee Division
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

Subject: NRC Region I Inspection 50-443/91-29 (9/10/91 - 10/14/91)

This refers to the above subject safety inspection of open items and the final phases of the first refueling outage at Seabrook. The inspection results are described in the enclosed report and were discussed with Mr. B. Drawbridge of your staff.

Overall, the inspection found that activities associated with the completion of the refueling outage were conducted safely.

However, an apparent violation of NRC requirements is cited in the enclosed Notice of Violation (NOV), for failure to restore a manual valve in the demineralized water system following maintenance resulting in the leakage of reactor coolant system water into secondary systems. Please respond to the apparent violation in accordance with the directions in the NOV.

Thank you for your cooperation.

Sincerely,

James C. Linville, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosures:
1. Notice of Violation
2. NRC Region I Inspection Report 50-443/91-29

9111070003
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ENCLOSURE 1

NOTICE OF VIOLATION

Public Service Company of New Hampshire
Seabrook Unit 1

Docket No. 50-443
License No. NPF-86

During NRC inspection from September 10 - October 14, 1991, a violation of NRC requirements was identified in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR, part 2, Appendix C. That violation is listed below:

> Technical Specification 6.7.1.a requires that the procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, be established, implemented, and maintained. Regulatory Guide 1.33, Revision 2, February 1978, Appendix A, Section 1.c specifies in part that procedures be established for equipment control (e.g., locking and tagging). New Hampshire Yankee procedure MA4.2, Equipment Tagging and Isolation, Section 4.9 requires that tagging order boundary components be restored to their proper positions. Procedure ON1055.01, Revision 4 (Change 20), Fill and Vent of Demineralized Water System, specified the normal position of DM-V301 as closed.
>
> Contrary to the above, on about September 30, 1991, demineralized water system valve DM-V301 was aligned in the open position during system restoration following completion of maintenance on the letdown line radiation monitor resulting in contamination of the demineralized water system.
>
> This is a Severity Level IV violation (Supplement 1).

Pursuant to the provisions of 10 CFR 2.201, Public Service Company of New Hampshire is hereby required to submit a written statement or explanation to the U.S.Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and if applicable, a copy to the NRC Resident Inspector, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report No.: 50-443/91-29 License No.:NPF-86

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

Facility: Seabrook Station, Seabrook, New Hampshire

Dates: September 10 - October 14, 1991

Inspectors:
N. Dudley, Senior Resident Inspector, Operations
A. Cerne, Senior Resident Inspector, Construction
S. Wookey, Resident Inspector
D. Moy, Reactor Engineer
J. Jang, Senior Radiation Specialist
J. Noggle, Radiation Specialist

Approved By: W. J. Lazarus 10/29/91
W. J. Lazarus, Chief, Reactor Projects Section 3B — Date

OVERVIEW

Operations: A violation was cited for use of an uncontrolled valve position database which contributed to contamination of the demineralized water system.

Radiological Controls: The response to the contamination of the demineralized water header was excellent.

Maintenance/Surveillance: Control and conduct of activities improved. Troubleshooting efforts on the air start distributor for diesel generator B were extensive.

Security: Implementation of program requirements were good.

Engineering/Technical Support: Support of activities required to be completed by the end of the first refueling outage was excellent.

Quality Assurance/Safety Verification: Resolution of safety concerns related to welding documentation and Cryofit couplings was supported by station management.

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DETAILS

1.0 SUMMARY OF ACTIVITIES

1.1 NRC Activities

Three resident inspectors were assigned. Three regional inspectors assisted in the inspection of restart commitments and a contamination event. Backshift inspections were conducted on 9/12, 9/16, 9/22, 10/1, 10/2, 10/7, and 10/9. Deep backshift inspections were conducted on 9/22, 9/25, 10/5, 10/6, 10/12, 10/13, and 10/14.

1.2 Plant Activities

The plant was in a refueling outage. Major work included reactor vessel head replacement, repair of diesel generator B air start distributor, replacement of Cryofit couplings, equipment retests, recovery from contamination of the demineralized water header, and repair of the hydrogen seal on the turbine generator.

The plant entered Mode 4, Hot Shutdown, on October 5 and the reactor was taken critical on October 9. Startup physics testing was completed and power was raised above 5% on October 12.

2.0 OPERATIONS

2.1 Plant Tours

The inspector conducted daily control room tours, observed shift turnover, and attended daily plan-of-the-day meetings. The inspector reviewed mode change checklists, containment integrity, compliance with Technical Specification requirements, tagging orders, and valve lineups. No deficiencies were noted.

Routine tours were made of the containment, the spent fuel building, diesel generator building, service water building, switchgear rooms, and the pipe chases. Two valves in the containment, SI-V117 and CGC-V51, were noted to have less than full thread engagement on one body to bonnet bolt. The Technical Support engineers provided documentation of an Engineering Evaluation completed on March 8, 1990, which documented the acceptability of the thread engagement on valve SI-V117. The thread engagement of the bolt on valve CGC-V51 was corrected. Other deficiencies noted by the inspector were verified to have been included on the Maintenance Department closeout discrepancy lists. Other minor housekeeping deficiencies identified to the Operations Department were corrected.

Before containment closeout, the inspector walked down portions of the Safety Injection (SI) and Residual Heat Removal (RHR) systems in the containment and verified that the SI and RHR valves were properly positioned and locked.

Planning for the outage minimized the risks identified in the Shutdown Probability Risk Assessment developed by New Hampshire Yankee. At least one protected train of Emergency Safety Features equipment was maintained throughout the outage. After completion of the four hour operability test of diesel generator (DG) A following an 18-month overhaul, train A became the protected train and disassembly of DG-B was commenced. The next day a Nuclear Quality Group (NQG) inspector found that one of the brush riggings on DG-A was cocked and scraping the slip ring. Even though DG-A was operable, DG-B was reassembled and tested for operability so that the cocking of the brush rigging on DG-A could be corrected without entering a Technical Specification Action Statement. The decision to maintain one diesel operable at all times and to immediately correct the identified deficiencies reflected a proper safety perspective.

On September 12, with the level of water in the refueling cavity below 23' above the reactor vessel flange in preparation for reactor vessel head installation, the Shift Superintendent was informed that the Inservice Inspection (ISI) of the RHR-B heat exchanger girth weld showed a potential weld defect. Train B of RHR was in operation. The Shift Superintendent entered Technical Specification 3.4.10, "Structural Integrity," which required that the structural integrity of ASME Code Class 2 components be restored prior to increasing reactor coolant temperature above 200 degrees F. Also, Technical Specification 3.9.8.2 "Residual Heat Removal and Coolant Circulation, Low Water Level," was entered based on the assumed inoperability of RHR train B. Preparations for the reactor vessel head replacement were stopped. RHR train B remained in operation, an evaluation of the indications of a weld defect was initiated, and the water level in the reactor cavity was maintained. After further review of the weld and ISI data, the Technical Support Department determined that the initial indications were questionable and that RHR train B should be considered operable. The Shift Superintendent exited both Technical Specifications.

The next day, following discussions between the inspector, Operations personnel, and Licensing personnel, Technical Specification 3.4.10 was re-entered pending final resolution of the indication of a defect in the heat exchanger girth weld. The indication was determined to be a result of the ultrasonic technique used and the coarse grain boundaries in the interior of the weld. Further details of the ISI aspects of this event are contained in NRC Inspection Report 50-443/91-27.

The inspector reviewed the Technical Specifications and their bases, discussed the ISI findings with Technical Support personnel, and observed the discussions between Operations, Licensing and Technical Support personnel. The inspector concluded that the decision by the Shift Supervisor to declare RHR train B inoperable and enter Technical Specification 3.9.8.2 was not necessary. In addition, the inspector noted that the action statements were not fully implemented. The decision to initially exit Technical Specification 3.4.10 was premature but was properly reevaluated by the Operations and Licensing Department. The inspector concluded that the requirements of the Technical Specifications were met.

2.3 Operator Aids

During the plant walkthrough portion of operator licensing examinations, several weaknesses were noted by the NRC examiners in regards to operator aids and procedures. The Operations Department took actions to address these weaknesses. The inspector reviewed the implementation of the corrective actions.

Emergency tools required for manually overriding the main steam isolation valves (MSIV) were available and labeled at each MSIV. The inspector page checked selected controlled operating and surveillance procedures and found that all inserted pages identified by a page number followed by an "a" were present. Since June 1991, Station Procedures require that a revision of a procedure must renumber all pages. An audit of selected working procedures in the file cabinet in the main control room determined that the appropriate revisions of the procedures were on file.

The inspector reviewed the administrative controls over five operator aids available in the main control room. Four were controlled by the "Operator Aids Control" book which indicated the aids on the index and all but one of the aids in the tabbed section of the book. The Operations Department corrected the minor administrative deficiency. The fifth aid was controlled by procedure OX 1408.02, "Boron Injection Flow Path Monthly Valve Alignment Check", and updated monthly.

The inspector concluded that the identified weaknesses were adequately addressed.

2.4 Demineralized Water System Contamination

On September 30, the demineralized water system became contaminated due to leakage from the Reactor Coolant System (RCS). The leakage was initiated when the pressure in the demineralized water system dropped (as a result of filling of the steam generators) below the pressure in the letdown radiation monitoring system causing the solenoid operated flushing water valve, RV-6520-02, to come off of its seat. A demineralized water manual isolation valve, DM-V301, was improperly left open completing the flow path to the demineralized water system.

The flow path remained open for approximately six hours and an estimated 1250 gallons of RCS water entered the demineralized water system. Activity levels due to Co-58 and Co-60 ranged from 10^{-3} μC/cc at the eye wash station closest to the flushing line to 10^{-7} μC/cc in the Condensate Storage Tank. Most of the contamination remained in the demineralized water system. Contaminated water which reached the secondary oil/water separator sump No. 2 and the turbine building sump, was pumped through hoses to the circulating water discharge structure. The unplanned radioactive effluent release and the radiological conditions created by the event are discussed in Details 3.1 and 3.2, respectively.

The inspector reviewed the Technical Support Department's Event Evaluation, station logs, and alarm printouts. The inspector attended SORC meetings and held discussions with plant staff concerning the event.

The main control room received a high radiation monitor alarm from the turbine building sump discharge monitor some time after the RCS leakage began. The alarm terminated the discharge from the sump and the sump was sampled. No activity was found. The radiation monitor setpoint was adjusted based on the background radiation levels and the discharge was reestablished.

The Volume Control Tank level chart recorder in the main control room was inoperable and level was being monitored from computer points which were trended on a visual display terminal. Level was noted to be decreasing at a higher rate than normal, which was confirmed by several automatic RCS makeups. The control room operators began reviewing all work in progress in an attempt to identify potential sources of RCS leakage. The leakage to the demineralized system was identified after maintenance workers became contaminated and a second alarm was received from the turbine building sump radiation monitor. A second sample from the sump indicated activity. Checks by Health Physics (HP) technicians of the radiation dose rate from the demineralized water piping helped identify and terminate the source of the RCS leakage.

The licensee's Event Evaluation Team determined that the solenoid valve functioned as designed. An earlier Engineering Evaluation determined that the design was adequate since there was no identified plant conditions that would develop the differential pressure needed to unseat the valve.

The manual isolation valve, DM-V301, was opened on September 19, 1991, in accordance with an incorrect tagging restoration sheet following maintenance on the CVCS letdown line radiation monitor. The error was the result of specifying the restoration valve lineup solely by reference to valve position information in a developmental, computerized tagging database. This database was not intended to be a replacement for system alignment procedures, schematics, or P&IDs, but as an informational tool to supplement them. Procedure MA4.2, Section 4.9 states that components not identified on P&IDs, electrical schematics, or in the procedures will be restored using other documentation. The operating procedure for the demineralized water system, ON1055.01A, contained a system lineup checklist which indicated that DM-V301 should be closed. A complete valve lineup for the demineralized water system was not conducted at the end of the outage because no work had been performed on the system and it was not recognized that the work on another system that interfaced with the demineralized water system had affected the valve lineup. Failure to restore the demineralized water system in accordance with procedures MA4.2 and OP1055.01, after maintenance on the letdown line radiation monitor, is a violation of Technical Specification 6.7.1.a (NV4 91-29-01).

The Operations Department completed a valve lineup of the demineralized water system after the event and conducted a review of the tagging restorations produced by the tagging computer to verify proper valve lineups. No additional errors were noted in the computer database.

During subsequent flushing of the demineralized water system, some contamination occurred in the secondary chemistry lab when a drain system backed up. The secondary chemistry lab was decontaminated and a work request was written to unclog the drain system. During filling of the condensate storage tank, 200 gallons of water spilled on the ground. The concentration of radioactivity in this water was well below the level that qualified for unlimited release. The slightly contaminated soil was placed in 55 gallon drums and guidance was provided to maintain the level in all tanks below the high level alarm setpoint.

3.0 RADIOLOGICAL CONTROLS

3.1 Assessment of Offsite Effects from the Contamination of Demineralized Water System

Following the inadvertent contamination of the Demineralized Water System, the licensee's effluent and environmental radiological assessment were reviewed by a region based inspector. The inspector toured areas of the letdown radiation monitoring system, turbine building, auxiliary boiler, demineralized water storage tanks, condensate storage tank, oil/water separator vault No. 2, settling basin, and storm drainage systems.

Approximately 45,000 gallons of slightly contaminated water was released from the turbine building sump and oil/water separator vault No. 2 to the ocean. A hose had been connected to transfer nonradioactive water from the oil/water separator vault No. 2 to the discharge building during the outage. During the event, the water in vault No. 2 became contaminated and was discharged via the hose to the discharge structure. Subsequent to the discovery of this event, the licensee identified two small leaks from the hose near a storm drain. This storm drain was connected to the settling basin from which the licensee controls the water released to the Browns River. Also, an estimated five gallons of contaminated water escaped to the auxiliary boiler building roof through the auxiliary boiler system.

The licensee took grab samples from the following areas to assess radiological impact to the public health and safety and to the environment, as well as to determine the plant contamination status.

- Turbine Building Sump
- Oil/Water Separator Vaults No. 1 and No. 2
- Auxiliary Boilers A and B
- Auxiliary Boiler Building Roof
- Condensate Storage Tank
- Demineralized Water Storage Tanks

- Primary and Secondary Component Cooling Water
- Reactor Makeup Water Storage Tank
- Storm Drain Manholes (six locations)
- Settling Basin Water
- Browns River Water
- Ocean Water
- Potable Water
- Associated Plant Systems

The inspector reviewed all results of radioactive measurements and radiological dose assessment and noted that Co-58 was the dominant gamma emitting radionuclide. All measurement results for environmental samples; storm drain water, settling basin, Browns River, and ocean (sample collected in Salisbury, Massachusetts) were less than the lower limits of detection (LLD) with exception of the soil collected in the vicinity of the leaky hose near the storm drain. Radiological analytical results of contaminated soil samples collected from the leaky hose appeared to be localized and Co-58 at a concentration of 1.53+/-0.06 x 10^{4} µCi/gram was the only identified plant-related radionuclide in the soil samples. Even though the activity of the contaminated soil was low (almost same activity of Cs-137 in soil due to fallout, and about 3% of natural background dose), the licensee treated the contaminated soil as radwaste. The soil was removed and stored in 55-gallon drums.

The licensee performed a dose assessment using the Offsite Dose Calculation Manual methodology for the turbine building sump and oil/water separator vault No. 2 release. The licensee did not know the total volume of release, therefore, the licensee enveloped the calculated amount of release based on time of the beginning of the event. The results of the radiological dose assessments for the whole body and organ doses were 4.76×10^{-3} mrem and 2×10^{-4} mrem, respectively to a member of the public involved in shore activities such as swimming and boating. Those projected doses were less than one percent of monthly Technical Specification limits.

Based on the review of the licensee's measurement techniques, analytical results, and actions, the inspector determined the following.

- The licensee has an excellent capability to accurately measure gamma emitters.

- The licensee's actions to monitor the possible leakage to the environment was excellent.

- The licensee has the capability to perform the necessary radiological dose assessment.

The inspector concluded that there was no negative impact on the environment or to the public health and safety as a result of this event.

3.2 Assessment of On-Site Effects From the Contamination of the Demineralized Water System

The licensee's onsite radiological control and handling of the event was reviewed by a region based inspector. The inspector witnessed the operations and technical support recovery team in planning and beginning the recovery activities. The inspector toured the new Radiological Controlled Areas (RCA) created by the event, reviewed radiological surveys, and interviewed licensee personnel in assessing the licensee's response to the event.

On September 30, 1991 at around 5:00 p.m., the first contaminated worker associated with this event was discovered. Within the next hour, two more workers were discovered to be contaminated, all apparently associated with demineralized water use. Around 6:00 p.m. the licensee deduced the possibility of a contaminated demineralized water source. The source of water was posted as a contaminated area and a water sample was taken for radiochemical analysis. The water was found to be contaminated.

Around 7:20 p.m. the process radiation monitor on the turbine building sump alarmed. At 7:22 p.m. the control room made a station wide announcement over the public address system for personnel to refrain from using the demineralized water system until further notice. By 8:30 p.m. the Health Physics Department made a preliminary determination that the contaminated demineralized water system extended beyond the bounds of the normal RCA and extended contamination monitoring and exposure monitoring controls to include the turbine building and auxiliary boiler building. All entrances and exits from these buildings were roped off and posted resulting in two access points at which whole body contamination monitoring instrumentation were installed.

Informational flyers were produced and distributed to plant personnel entering the station informing them of the new RCA and associated monitoring requirements. Continued surveying throughout the night and next morning resulted in the posting of four contamination areas in the turbine building and one contamination area in the auxiliary building with the addition of one radiation area closely associated with the Secondary Component Cooling (SCC) Header Tank located at the top of a vertical ladder above the top floor level of the turbine building. The new RCA was completely surveyed about noon on October 1, 1991 which was approximately eighteen hours following identification of the event. Portable Eyewash Stations were positioned in the Primary Auxiliary Building at those stations supplied by demineralized water. The inspector concluded that the licensee response to the event was appropriate and timely.

Exposure Impact

From a review of the post-event survey data, the following radiological environment in the new RCA was found:

LOCATION	DOSE RATES	CONTAMINATED AREAS
Turbine building 21 ft.	0.05-0.1 mR/hr	4 areas (5,000 dpm/100 cm^2)
46 ft., 50 ft.	0.01-0.03 mR/hr	
75 ft.	0.01-0.02 mR/hr	
SCCW Head Tank 75 ft.	2.5 mR/hr	
Aux Boiler Building	0.1-1 mR/hr	1 area (5,000 dpm/100 cm^2)

These general area radiation levels were the initial conditions before any system decontamination flushes were performed. An average new RCA dose rate was estimated to be 0.08 mR/hr. A worker continually exposed to this radiation field for 520 hours per calendar quarter could theoretically receive 42 mRem. 10 CFR 20.202 does not require personnel monitoring below 312 mRem per calendar quarter. Therefore, the external exposure hazard was minimal.

The contamination level of 5,000 dpm/100 cm^2 was a low concentration of activity which was not expected to pose any internal exposure hazard. The licensee instituted an air sampling program in the turbine building to verify the lack of an airborne radioactivity hazard. The plant's potable water supply was repeatedly sampled during the event with no trace of radioactivity detected.

The inspector concluded that the radiological conditions found as a result of the event, posed no significant additional external or internal exposure hazard to the station personnel. A preliminary estimate suggests that there will be an increase in the radioactive waste generated in the form of spent ion exchange resins and Dry Active Waste (DAW) resulting from the cleanup of the various contaminated areas.

Nonradioactive System Controls

The inspector reviewed the licensee's program for routine sampling or monitoring of the various nonradioactive systems of the plant with respect to IE Bulletin 80-10 and IE Circular 80-14. IE Bulletin 80-10, "Contamination of a Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to the Environment" recommends a routine sampling/analysis or monitoring program be established for those nonradioactive systems that have interfaces with radioactive systems to allow identification of contaminating events. This same guidance also states that if a nonradioactive system becomes contaminated, the affected system should be restricted in its use, the cause of the contaminating event identified, and the affected system components decontaminated. The guidance continues by suggesting a formal 10 CFR 50.59 safety evaluation be performed immediately if the normally nonradioactive system is to be operated in a contaminated condition. The inspector verified current sampling/analyses had been performed according to the following schedule.

Nonradioactive System	Sampling Frequency
Service Water	Weekly
Demineralized Water Header	Monthly
Instrument Air System	Monthly
Sewage Treatment Sludge	Annually
Settling Basin Sludge	Annually

After acknowledgement of the event, the licensee restricted the use of the demineralized water system and began a 10 CFR 50.59 safety evaluation to identify any unreviewed safety concerns. The licensee formed a round-the-clock team to prevent any further releases to the environment, to identify the cause of the event, and to develop an operational recovery plan to effect a methodical system flush or decontamination. The inspector concluded that licensee actions were taken in accordance with stated regulatory guidance.

Event Recovery

The licensee added five Health Physics technicians to its staff to assist in the long term monitoring and contamination control of the contaminated secondary systems. A weekly radiation and contamination surveillance program was initiated for the new RCA plant areas. Additional radiological work controls were implemented in new areas requiring Radiation Work Permits (RWP) for any work involving breaching of a contaminated secondary system and for working inside any of these contaminated areas. Additionally, all vehicles exiting the station and any dumpsters leaving the station during the first few days after the event were surveyed.

The licensee provided the following future recovery guidance. After the secondary system flushing is completed in October, the radiological conditions will be reassessed. The decision to remove posted signs and controls from potentially contaminated secondary areas will be made based on dose rates less than 0.6 mR/hr, contamination levels not measurable above background, and the potential for detecting uncontained radioactive material in the areas.

3.3 Control of Contamination in Containment

Prior to drain down of the reactor cavity in preparation for reactor vessel head installation, personnel worked in most areas of the containment in work clothes. After the drain down of the reactor cavity, contamination from the cavity became airborne and was spread throughout the containment. Access to the containment was restricted until the airborne problem was eliminated. However, efforts at decontamination of the containment were unsuccessful and all personnel entering the containment were required to wear full anti-contamination clothing. The Health Physics Department expected the contamination, comprised primarily Co-58 which has a half life of 70 days, to decay away prior to the next refueling outage.

The inspector concluded that contamination control in the containment was good until the cavity was drained. The Health Physics Department plans to review the cavity drain down process to improve contamination control for the next refueling outage.

4.0 MAINTENANCE/SURVEILLANCE

4.1 Plant Tours

The inspector observed work in progress in the containment, the primary auxiliary building, the pipe chases, the diesel generator rooms and the turbine building. In general, the control of materials and processes in containment were stronger than controls in the pipe chases. Controls and processes improved as the outage progressed. The organization and control of diesel generator B 18-month maintenance overhaul appeared strengthened as a result of the lessons learned during the 18-month maintenance overhaul of diesel generator A. Housekeeping in non-safety related areas was adequate.

The program to paint and label the doors in the protected area was completed. In addition to the information denoting a fire or security door, decals were posted for hearing protection and other precautions associated with the area about to be entered. The use of the newsletter to communicate to workers the status of outage conditions or requirements was a good initiative.

4.2 Diesel Generator Air Start Distributor

On September 11, 1991 diesel generator (DG) B was shutdown due to excessive vibrations and overheating of the right air start distributor. The diesel had run for over two hours as part of a monthly surveillance test to establish operability after an 18-month maintenance overhaul. Several air line fittings from the air start distributor were cracked and the rotor was worn due to contacting the housing. The vendor, Colt-Pielstick sent consultants to assist. The alignment of the camshaft to which the rotor is attached was checked, and the rotor and housing were replaced.

On September 16, DG-B was again shutdown during performance of the monthly surveillance test due to an oil line failure. The oil line was repaired and the surveillance test was restarted. The diesel was later shutdown due to high noise in the area of the air start distributor. Two air lines were found severed on the air start manifold, and the rotor was damaged due to contact with the housing. A new housing and rotor from a Unit 2 air start distributor were installed. The oil mister system which provides lubrication for the air start rotor during initial startup was inspected and cleaned. Blockage was found in the oil supply lines and the vent hole in the original air distributor housing was found to have been clogged. The oil mister system on DG-A was also inspected and an oil pump was replaced.

The air start control valve solenoids and air lines were instrumented and vibration monitors were placed on both air start solenoids. A series of test runs of DG-B were conducted without recurrence of the high vibrations. A 24 hour operating test at full load was conducted. After 21 hours of operation, the vibrations appeared and DG-B was manually shutdown on September 21, 1991.

A design modification to install an alignment bearing on the end of the air start distributor housing was developed by New Hampshire Yankee and the parts were manufactured on site. The modification was never installed since a realignment of the coupling between the camshaft and the air start distributor rotor resolved the problem. The final realignment specifications were provided to New Hampshire Yankee by Colt-Pielstick on September 24. The vendor stated that each air start distributor housing was unique and an alignment was necessary each time the housing and rotor were replaced.

A review of DG-B vibration data indicated an increased level of engine vibration over the last year. An inspection of the foundation and foundation bolts of the diesel identified a cracked washer and apparent holes in the grouting. The washer was replaced and the foundation was grouted. An epoxy adhesive was injected into the holes. The inspector reviewed the procedure (MS91-1-19) and the licensee's controls of the vendor performing the grouting. The responsible technical support engineer was knowledgeable of the process and monitored the vendor's performance. Technical Support will monitor future diesel starts to determine if the vibration has reduced as a result of the foundation grouting.

On September 25, DG-B failed to start due to an electrical ground caused by a temporary coaxial instrumentation lead. The ground activated the voltage imbalance relay which caused the voltage regulator to transfer to manual before full field voltage was established. The less than full field voltage caused the potential transformers to produce less than full control voltage signals resulting in the governor positioning the fuel racks to minimum position. As a result, the diesel was starved for fuel and stopped. The coaxial lead on the temporary instrument was replaced with an alligator clip lead, the diesel was rolled with air, and a successful 24 hour run was conducted.

The inspector noted that the vendor was consulted early in the troubleshooting process and that New Hampshire Yankee independently developed design changes when implementation of initial vendor recommendations were unsuccessful.

4.3 Diesel Generator 18-Month Surveillance

The inspector observed preshift briefings and the conduct of portions of the 18-month surveillance tests for DG-A. The preshift briefings were extensive, and copies of the procedures were available for all personnel involved. No problems were encountered during the conduct of the test. The inspector noted an improvement in the preshift briefing and test conduct over the previously conducted test on DG-B.

5.0 SECURITY

During routine plant tours, the inspector noted security practices which included personnel access control, compensatory actions taken for impaired security barriers, the conduct of routine patrols, and operation of the Central Alarm Station and Secondary Alarm Station. The inspector reviewed actions taken for Fitness-for-Duty failures and the extent of the licensee's review of previous work activities of the individuals who failed the testing. No deficiencies were noted.

6.0 ENGINEERING/TECHNICAL SUPPORT

6.1 Feedwater Check Valve Bolting Modification

On April 1, 1991, feedwater header check valves were found to have broken cap screws that retain the internal valve dashplate. The design of these check valves is unique to Seabrook, but is similar to smaller check valves used in other facilities. The reason for the failures was determined to be over-stressing the cap screws during valve opening transients. The check valves were modified by increasing the number of cap screws on the dashplate from eight to sixteen. The modification was reviewed in NRC Inspection Reports 50-443/91-04 and 50-443/91-15. On April 23, 1991, Edward Valves Inc. issued a 10 CFR Part 21 report which recommended use of larger cap screws, a wider lower locking ring, and dashplate modifications. On September 25, 1991, the NRC issued a safety evaluation of the installed short-term modifications and the design modifications recommended by Edward Valves, Inc. The NRC staff concluded that the implementation of the design changes would reduce the likelihood of failure of the check valves. The inspector reviewed the completed modification package and the associated 50.59 safety evaluation analysis.

No transient loads were identified that could produce a sufficient load to cause elongation or failure of the sixteen 3/8 inch bolts. However, for conservatism, the dash plates and locking rings were redesigned to withstand the system design pressure and the maximum differential pressure across the dash plate in the valve opening direction. The inspector noted that the original design calculations did not evaluate the stresses on the dash plate during the opening of the check valves and that later calculations based on complex design considerations were indeterminate as to the stresses that were placed on the bolts.

During receipt inspection of the redesigned dashplates, New Hampshire Yankee determined the bolt holes were improperly drilled and returned the dashplates to the vendor for correction. After the installation of the larger bolts, the Technical Support engineer determined that the clearance between the thicker bolt heads and the underside of the valve piston head was insufficient. The piston head was machined to provide adequate clearance.

During the August 1991 outage, the check valves were inspected. No elongation of the bolts were noted. Design Change, DCR 91-035, which replaced the sixteen 3/8 inch cap screws with 5/8 inch cap screws and installed redesigned dashplates and locking rings provided by Edwards Valves, Inc. was implemented.

The inspector reviewed the repairs to main steam isolation valve MS-V-92, which was supplied by Edwards Valves, Inc. and had experienced several failures. The valve was originally supplied by Rockwell International, Flow Controls Division which was later acquired by Edwards Valves, Inc. Two solenoid valves and a check valve on the hydraulic actuator were determined to be leaking by October 28, 1990. These valves were replaced with a package which included a new air driven hydraulic pump. The original air driven hydraulic pump was functioning properly. Later on June 27, 1991, the new air driven hydraulic pump seized and was replaced.

The solenoid valves were manufactured by Keane Controls Corporation of Fullerton, California and the air driven hydraulic pumps were manufactured by Haskel Engineering and Supply Company of Burbank, California. The inspector determined that the failures of MS-V-92 were caused by different components which were supplied by different subcontractors. New Hampshire Yankee conducted an inquiry into an industry reliability data system and found no problems with Edward Valves, Inc. main steam isolation valves reported by other utilities.

The inspector concluded that the failures of MS-V-92 did not indicate a generic problem with faulty or inadequate equipment being supplied by Edwards Valves, Inc. The inspector concluded that the initial design of the main feed water isolation valves were inadequate for the application, as indicated by the Part 21 Notification. Corrective actions were taken which resolved the design deficiency. The problem was determined to be unique to Seabrook based on the size of the check valves; however, the reason for the design deficiency has not been resolved between Edwards Valves, Inc. and NHY. This issue is closed.

6.2 Thimble Tube Thinning Bulletin No. 88-09

Bulletin No. 88-09 was issued on July 26, 1988 and requested the implementation of an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. The inspector reviewed New Hampshire Yankee's incore instrument thimble tube inspection and monitoring program, RS0727, Rev.0, which was developed in response to the NRC Bulletin 88-09. No deficiencies were noted.

The first incore instrumentation thimble tube inspection was successfully completed by Cramer and Lindell Engineers on September 15, 1991. A previous inspection attempt had been unsuccessful due to developmental problems associated with the test probe. The thimble tubes are of a dual wall construction with fixed incore detector wire between the walls. The diameter of the inner tube is 118 mils compared to the standard incore thimble

tube diameter of 188 mils. The newly developed probe was of a smaller diameter and provided eddy current data on both the inner and outer walls. A final report on the results of the eddy current test was scheduled for submittal to the NRC in October.

6.3 Startup Preparations

The inspector reviewed the following plant startup operating procedures in preparation for observing the startup.

-- RS1737.1, Post Refueling Initial Criticality
-- RS1737.2, Post Refueling Low Power Physics Testing
-- RS1738, Power Ascension Testing
-- RN1736, Reactivity Measurement, Rev.2
-- RN1732, Incore Analysis, Rev.1
-- RN1733, Flux Mapping System Operation, Rev.1

For each of the above procedures, the inspector verified that the acceptance criteria, precautions and initial conditions were adequately stated. The inspector found the procedures to be consistent with the requirements of ANSI/ANS19.6.1-1985, "Reload Startup Physics Test for Pressurized Water Reactor." No deviations were noted.

The inspectors observed the conduct of portions of Startup Physics Testing. Reactor Engineering and Quality Control personnel were stationed in the Control Room working with the licensed operators during the physics testing. Pre-task briefings were held before critical steps and communications between the groups were excellent.

6.4 Core Exit Thermocouple High Temperature Alarm

The implementation of design changes recommended by NRC Generic Letter 88-17 were initially reviewed in NRC Inspection Report 50-443/91-06. The core exit thermocouple temperature high alarm was not installed at the time of the inspection. The inspector verified that the alarm was installed in the control room and was set at 200°F as specified in procedure OS1000.12, "Mid-Loop Operation." The inspector also reviewed procedure OS1213.02, "Loss of RHR While Operating at Reduced Inventory on Mid-Loop Conditions." No deficiencies were noted.

6.5 Diesel Generator Jacket Water Temperature Control

The inspector reviewed Design Change Request DCR 90-0049 that modified the existing pneumatic control loop which modulates the diesel generator jacket water temperature control valve. The inspector discussed the setting for the diesel generator high temperature trip with Engineering personnel. The inspector verified installation of the derivative unit addition to the existing control loop and the addition of protective covers to the setpoint adjustment knobs for the pneumatic temperature and differential pressure controllers.

Operating experience during diesel generator surveillance testing had shown that jacket water temperature, which is controlled by a three-way temperature control valve, may overshoot the trip setpoint following a diesel start. The addition of the derivative unit to the control circuit provided a smoother transition from initial startup to steady state operation with minimal overshoot and avoided approach to the high temperature trip setpoint.

The inspector noted that the design change was extensively reviewed prior to implementation. A 10CFR50.59 evaluation and an Interdiscipline Review and a Risk/Reliability Review were completed. Seismic qualification of the derivative units was documented. The addition of the protective covers on the controller setpoint adjustment knobs satisfied the corrective action statement included in a September 9, 1988 Diesel Generator Special Report from the licensee. DCR 90-0049 included minor system enhancements such as adding vent, drain, and instrument test connections, changing a setpoint, and modifying an alarm circuit. The safety evaluation included a review of the system enhancements.

The inspector concluded that the design change was beneficial to safety since the occurrence of spurious high jacket coolant temperature trip may be avoided.

6.6 Westinghouse Relays and Magnetic Contactors - Information Notice 91-45 (Closed)

As described in Inspection Report 50-443/91-22, there are 66 similar relays in use at Seabrook Station which may malfunction due to an epoxy compound becoming semifluid when the coil is energized for extended periods. Four of the relays are in the reactor protection system. The remainder are used in alarm or computer circuits. During the refueling outage, the licensee removed and inspected the four reactor protection system relays. One relay showed visual evidence of flowing compound before testing. A spare relay and the four reactor protection system relays were energized at 138 volts dc for two hours. At the end of the test period, each relay was deenergized, disassembled, and inspected. Four of the relays tested satisfactorily, the one with "as found" flow exhibited obvious flowing of the compound. Westinghouse was informed of the test results and noted the results during a meeting at NRC Headquarters on September 18, 1991.

The licensee reinstalled the four relays that satisfactorily passed the test into the reactor protection system. The relays used in alarm and computer applications will remain in service since they are used in a low temperature environment and are not normally energized. The licensee plans to return spare relays to Westinghouse for testing prior to use.

6.7 Steam Generator Level Channels Filter Card Addition

NRC Inspection Reports 50-443/90-15 and 90-17 provided background on a feedwater isolation based on erroneous steam generator level signals following a reactor trip. The reports addressed the short term actions taken prior to the implementation of a design

change. The inspector reviewed Design Change Request 90-0041 and the completed work packages which installed a Westinghouse recommended lag circuit. Selected wiring diagrams that were changed as a result of the design change were reviewed.

The design change involved installation of circuit cards configured for a lag function in the output of the narrow range steam generator level transmitters. The design change included a modification to permit routine testing of the steam generator level bistables without jumpers and removed an unused load dependent level setpoint circuit identified by Westinghouse as a potential cause for a multiple loop feedwater malfunction event.

The inspector noted use of industry experience in designing this change. The licensee developed station procedures for testing the filter cards in accordance with ISA Standard S67.06 in addition to Westinghouse recommendations. The inspector acknowledged the positive initiative to eliminate jumpers for routine testing and to eliminate the potential common mode failure. The work packages were comprehensive and were completed in accordance with procedural guidelines.

7.0 QUALITY ASSURANCE/SAFETY VERIFICATION

7.1 Evaluation of Contamination of Nonradioactive Systems

The inspector reviewed the 10CFR50.59 evaluation for operation of certain nonradioactive systems which became contaminated as a result of a reactor coolant system leak into the demineralized water system. The evaluation assumed that all systems which directly, or indirectly, interfaced with the demineralized water system, would operate with some level of internal contamination until activity levels decayed to less than detectible values. Some systems are normally radioactive and were not evaluated. Other normally non-radioactive systems had predetermined actions under which operations could continue if they became radioactive and were not evaluated. The remaining systems were evaluated as a potential release path to the environment.

The evaluation considered (1) expected off-site liquid doses resulting from the Demineralized Water System decontamination and cleanup operations, including maximum potential impact based on the total source term transferred from the primary system, (2) the maximum hypothetical and potential airborne release via secondary system components such as the Auxiliary Boiler aerated vent, and system safety relief valves, and (3) a postulated accident leading to the failure of the outdoor Demineralized Water Storage Tanks. All potential, as well as expected offsite doses to members of the public, were determined to be less than the allowable dose limits established in Section 3.11 of the Station's Technical Specifications. These dose limits were taken from both 10CFR50, Appendix I for routine operations to keep doses to the public "As Low As Reasonably Achievable", and the 40CFR190 total dose standard for combined dose contributions from all uranium fuel cycle sources.

The inspector verified that the calculated maximum hypothetical radioactivity releases did not exceed Technical Specification effluent release limits or the offsite dose limits. The inspector concluded that the evaluation supported continued operations of the contaminated systems. The evaluation was conservative, thorough, and well supported by engineering tests and analysis.

7.2 Reactor Coolant System Unidentified Leakage LER 91-010 (Closed)

On September 11, 1991 the licensee staff presented NRC Regional personnel an overview of the investigation and corrective action for the Cryofit tube coupling failure which resulted in reactor coolant system leakage. Details of the meeting are provided in Attachment 1. The licensee submitted a Licensee Event Report (LER) Supplement on October 2, 1991. The event review and the original LER response are reviewed in NRC Inspection Reports 50-443/91-19 and 91-22.

The inspector discussed the scope of the investigation, testing program, and hardware changes with the Instrument and Control Engineering Supervisor. The inspector verified the removal of Cryofit tube couplings in the containment in instrument lines exposed to a high temperature and high hydrogen environment. The inspector reviewed the New Hampshire Yankee Cryofit Coupling Verification Program Engineering Evaluation and the associated procurement documents.

A chemical analysis was performed to verify the material in use was the alloy specified. Instrument tubing that was installed as a result of the cryofit removal was formed using the original tubing as a template which resulted in a reduced number of conventional tubing connections. The scope of the piping replacement was extensive and included all areas of the reactor coolant boundary within containment at risk for the failure mechanism.

The licensee expended extensive resources and significant man rem exposure in the effort to address the potential safety implications of the cryofit failures. The inspector concluded that the Coupling Verification program was well defined and implemented in a timely manner.

The LER supplement update accurately describes the scope of the safety related tubing replacement. This item is closed.

7.3 Unlocked Circuit Breaker - LER 91-005 (Closed)

During performance of procedure OX-1446.07, "Verification of Locked Valves," the breaker for the rod control cluster change fixture (1-FH-RE-12) in motor control center MCC-111 was found open and unlocked. Technical Specification 3.8.4.1, "Containment Electrical Penetrations," requires that the breaker be opened and locked. This event was evaluated in NRC Inspection Report 50-443/91-15.

The inspector reviewed LER 91-005 and noted that the identified corrective actions appropriately addressed the event. The inspector reviewed the implementation of corrective actions and held discussions with Engineering Department personnel. Signs were placed on all Technical Specification 3.8.4.1 related circuit breakers. Procedure OS1090.05, "Component Configuration Control," was revised to reflect the reduced number of locked valves resulting from an engineering analysis. All locked components identified in OS1090.05 were listed in the tagging computer. The inspector noted that the failure to control the listing of components in the tagging computer contributed to the contamination of the demineralized water system. This LER is closed.

7.4 Missing Radiographic Record for a Safety-Related ASME Field Weld - Unresolved Item 90-24-02 (Open)

By letter (NYN-9115) dated September 17, 1991, New Hampshire Yankee (NHY) submitted its final reply to the NRC Notice of Violation issued in conjunction with NRC Inspection Report (IR) 50-443/91-12. Along with the documentation of the final corrective action taken in response to the violation, NHY provided a completion report on the Weld Radiograph Reinterpretation Program (WRRIP) instituted as a result of NRC findings identified and documented in NRC IR 50-443/91-21. NRC inspectors returned to Seabrook Station on September 23-24, 1991, as a continuation of IR 91-21 inspection activities, to conduct additional radiograph reviews incident to the licensee WRRIP completion.

By letter dated September 24, 1991, the NRC requested additional information from NHY regarding the WRRIP conclusions and supporting data. NHY responded by letter (NYN-91157) dated September 27, 1991, attaching supplemental contractor (i.e., Hellier) report information and specific data and explanations to address all remaining NRC questions regarding radiographic record and film adequacy. Subsequently, the inspector was notified that one weld number in Enclosure 3 to NYN-91157 had been incorrectly documented. The incorrect weld number, 1-CS-318-04, F0204, should have been listed, relative to the Table 3 reference, as weld number 1-CS-318-02, F0204. The inspector determined that this typographical error was inconsequential to the finding and conclusions of the licensee WRRIP Completion Report.

The inspector reviewed the licensee's letters and the WRRIP data and findings, and evaluated this information relative to the overall results of the programmatic efforts taken to reverify the Pullman-Higgins field weld records. This Weld Record Reverification Program (WRRP), planned and initiated in March 1991 at NRC request, was completed in August 1991 with the final NHY status report (NYN-91134) submitted to the NRC on August 30, 1991. The most significant WRRP findings were the identification that four weld radiographic record packages could not be located in the NHY records vault. The specific details for each missing record were inspected and are documented in previous inspection reports addressing this open item. Subsequent corrective action, to include the re-radiography of all four welds,

was spot-checked by the inspector including the witness of re-radiography in progress, and the final radiographs were independently reviewed by qualified NRC film reviewers (reference: IR 91-21).

Other WRRP findings and WRRIP inspection items, relating to record deficiencies, were documented on NHY Corrective Action Requests (CARs) and reviewed by the NRC as each CAR was closed out. The identified record deficiencies represented issues of minor safety significance and were deemed appropriate to be processed and corrected by the licensee corrective action program. During this inspection, the inspector reviewed the last five CARs dealing with the deficiencies identified relative to the radiographic records review process.

These five Corrective Action Reports (CARs 91-036 thru 91-040) all relate to record discrepancies and not weld quality problems. The inspector spot-checked the disposition of each CAR, noting confirmation of the completion of all corrective measures by the NHY QA Surveillance Supervisor with final acceptance of corrective action verification by the Nuclear Quality Manager. Where calculations were included in validating the acceptability of radiograph quality geometric unsharpness determinations, the inspector independently checked the numerical results to confirm compliance with ASME Code, Section V criteria. The inspector identified no problems with the licensee conduct or documentation of corrective action for the identified records deficiencies.

NRC overview of the licensee WRRP and WRRIP activities and resolution of all corrective actions relative thereto is essentially complete. The results of the NRC team inspection record and film quality reviews are documented in NRC IR 50-443/91-21. No additional safety concerns or unacceptable findings that had not already been documented by the licensee were identified. No evidence that inadequate final welds were accepted at Seabrook Station has been identified. NHY has responded to all NRC questions and met all commitments to the NRC regarding the adequacy of Pullman-Higgins field weld and related radiograph quality. Therefore, the NRC plans no further technical review or inspection followup regarding this matter at Seabrook. However, NRC enforcement action for the identified violations is under consideration. This unresolved item remains open until the NRC enforcement deliberations and final documented actions are completed.

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 meeting was held at the Region I office at King of Prussia, Pennsylvania, on September 11, 1991. At the meeting, New Hampshire Yankee presented the results of their review of the failed Cryofit coupling in the pressurizer steam space sample line. The list of Attendees and the slides used during the presentation are provided as Attachment 1.

Region-based inspectors conducted the following exit meetings during this report period.

DATE	SUBJECT	REPORT NO.	INSPECTOR
9-13	ISI: Eddy Current Testing	91-27	R. McBrearty
9-24	Welding Records	91-21	M. Modes
9-27	Startup Testing	91-29	D. Moy
10-2	Radioactive Effluent	91-29	J. Jang
10-3	H.P. Controls	91-29	J. Noggle
10-4	Operator Licensing	91-30(OL)	J. D'Antonio

ATTACHMENT 1

NEW HAMPSHIRE YANKEE PRESENTATION OF
CRYOFIT COUPLINGS
NRC REGION I OFFICE, KING OF PRUSSIA, PA
September 11, 1991

LIST OF ATTENDEES

U.S.NRC

R. Barkley, Project Engineer, RI
J. Durr, Chief, Engineering Branch, RI
W. Haass, Vendor Inspection Branch, NRR
H. Kaplan, Mat's Engineer, RI
W. Lanning, Deputy Director, DRS, RI
J. Linville, Chief, Projects Branch 3, RI
A. Lohmeier, Reactor Engineer, RI
S. Wookey, Resident Inspector, SB

NHY ATTENDEES

B. Beuchel, I&C Engineering Supervisor
P. Brooks, Division Manager Metals Division
R. Deloach, Executive Director - Engineering & Licensing
B. Drawbridge, Executive Director - Nuclear Production
T. Harpster, Director Licensing Services
G. Kline, Technical Support Manager
A. Pelton, Technical Manager, Metals Division
J. Vargas, Manager of Engineering
K. Willens, YAEC Principal Materials Engineer

New Hampshire
Yankee

CRYOFIT COUPLINGS

New Hampshire Yankee Presentation to NRC

September 11, 1991

PRESENTATION PARTICIPANTS

- *Bruce Drawbridge* - NHY Executive Director, Nuclear Production
- *Jeb DeLoach* - NHY Executive Director, Engineering and Licensing
- *Terry Harpster* - NHY Director of Licensing
- *Joe Vargas* - NHY Manager of Engineering
- *Gary Kline* - NHY Technical Support Manager
- *Bruce Beuchel* - NHY I&C Engineering Supervisor
- *Ken Willens* - YAEC Mechanical Engineering Group
- *Alan Pelton* - Raychem Corporation
- *Peter Brooks* - Raychem Corporation

PURPOSE AND OVERVIEW

- *Jeb DeLoach,* Executive Director of Engineering and Licensing

PURPOSE

- Show understanding of the cause of failure
- Selective replacement appropriate and conservative
- Remaining couplings suitable for the application

CRYOFIT COUPLING

- Raychem Corporation tradename
- Joins tubing in sample and instrument lines
- Alloy called TINEL
- Shape memory effect

New Hampshire
Yankee

OVERVIEW

- Failure mechanism
- Corrective action
- Conclusion

DETAILED DESCRIPTION

- *Bruce Beuchel* - NHY I&C Engineering Supervisor

APPLICATION

- Sample and instrument sensing lines
- 1/4", 3/8", 1/2"
- Safety-related and balance-of-plant
- Over 3000 originally installed

INITIAL QUALIFICATION

- Raychem Qualification Package
- Raychem Report EDR-5116

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Raychem Qualification Package

- ANSI B31.1 up to 2"
- ASME III up to 1"
- Mechanical Performance
  - Burst Pressure
  - Tensile Strength
  - Fatigue, vibration, and shock
- Elevated temperature
- U.S. Navy approval
- Vendor-identified test programs

Raychem Report EDR-5116

- Temperature rating upgraded for pressurizer
- Elevated temperature tests
  - Long term temperature soak
  - Thermal cycling
  - Stress relaxation studies

INITIAL PROCUREMENT

- UE&C specification for instrument piping
- UE&C specification for cryogenic couplings
- Traceability
- Not fraudulent material or bad lots

NHY ACTIONS

- Immediate Actions
- Literature Search
- Review Third Party Testing
- NHY Program

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Immediate Actions

- Root Cause Analysis
  - Test samples from pressurizer gas space sample line
  - Form hypothesis
- Replace selected couplings
- Perform system walkdowns
- Develop selection criteria for additional tests
- Informed rest of nuclear industry

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Literature Search

- EPRI
- INPO
- DOE
- NASA
- US Navy
- AECL
- NSSS Vendors

Review Third Party Testing

- B&W
  - PWR primary and secondary (AVT) chemistry
  - U-bend specimens
  - Applicable to Seabrook
- Framatome
  - PWR primary chemistry
  - Different alloy of TINEL
  - Confirmed hypothesis

NHY Program

- Cryofit Locations
- Sample Selection Process
- Test Specifications
- LOCA Test
- Replacement Criteria
- Cryofit Replacement Summary
- Long Term Corrosion Study

Cryofit Locations

• Main Steam and Feedwater	~ 1400
• Sampling (primary, secondary, other)	~ 1000
• Reactor Coolant	~ 300
• Chemical and Volume Control	~ 150
• Blowdown	~ 125
• Hydrogen Gas	~ 50
• Others	~ 250
• Stores	~ 215

Sample Selection Process

- Both RCS and AVT chemistry
- Various process temperatures
- Various hydrogen concentrations
- Flow conditions (flowing / stagnant)

Test Specifications

- Macroscopic visual inspection
- Functional tests
  - Leak test
  - Burst test
- Metallography
- Hydrogen pickup
- Mechanical properties

Summary of Cryofit Test Results

Sample	Burst Pressure (psi)	Hydrogen Pickup (ppm)	Metallography	Mechanical Properties
Virgin Cryofit Coupling	> 24,000	< 10	No Cracks	No Degradation
Pressurizer Gas Sample				
- Inside Containment	> 24,000	700 - 2000	Cracking	Significant Degradation
- Outside Containment	> 24,000	< 10	No Cracks	No Degradation
Pressurizer Liquid Sample	> 24,000	10 - 20	No Cracks	No Degradation
RCS Hot Leg Sample	> 24,000	10 - 20	No Cracks	No Degradation
Pressurizer Level Instrument				
- Reference Leg Close to PZR	> 24,000	50 - 70	Minute Cracks	Slight Degradation
- Reference Leg Close to Transmitter	> 24,000	< 10	No Cracks	No Degradation
- Variable Leg Close to PZR	> 24,000	10 - 50	No Cracks	No Degradation
Waste Gas System Instrument	> 24,000	< 10	No Cracks	No Degradation
Charging System Instrument	> 24,000	10 - 15	No Cracks	No Degradation
S/G Blowdown Sample Lines	> 24,000	10 - 15	No Cracks	No Degradation
Main Steam System Instrument	> 24,000	< 10	No Cracks	No Degradation

Replacement Criteria

- Degraded couplings
- Couplings in lines with RCS chemistry which are:
  - RCS pressure boundary and not remotely isolable
  - Required for RPS / ESFAS
  - Required for ECCS operability

LOCA Test

- 24 hours w/ 4% Hydrogen
- Virgin Cryofit couplings
- Cryofit couplings removed from plant
- Results
  - No significant hydrogen absorption
  - No change in mechanical properties
  - No cracking

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Yankee

Cryofit Replacement Summary

- RC Sample Lines in Containment
  - Pzr gas space ~ 50
  - Pzr liquid space ~ 30
  - RCS loop 1 hot leg ~ 12
  - RCS loop 3 hot leg ~ 25
- Instrument Lines
  - Pzr level/pressure ~ 70
  - RCS flow ~ 150
  - RTD bypass flow ~ 20
  - Reactor vessel level 5
  - RCP #1 seal Δp ~ 25
  - Charging pump flow 1

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Long Term Corrosion Study

- Confirm 40 year life of Cryofit couplings
- Periodic sampling during outages

CONCLUSIONS

- Root cause evaluation indicates design application issue
- Implemented a thorough and effective program to resolve
- Remaining couplings can perform their function
- Periodic sampling program will ensure validity of test results