U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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EXECUTIVE SUMMARY

Seabrook Generating Station, Unit 1
NRC Inspection Report 50-443/96-10

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7-week period of resident inspection.

Operations:

- The licensee response to the NRC identification of the presence of temporary shipping caps in the spare conduit ports of several pressure transmitters has been excellent. The caps have been in place since original transmitter installation and are inconsistent with vendor manuals. However, due to the specific transmitter applications and environmental qualification classifications, instrument operability was not effected. The licensee decision to establish a task force to verify or establish proper configuration for a significant sample population of currently installed transmitters reflected sound issue scope and initial corrective action perspectives.

- In sharp contrast to the corrective actions perspectives evidenced above, the licensee failed to take prompt or comprehensive initial corrective actions to concerns initially raised by the Nuclear Safety Audit and Review Committee (NSARC) in March 1994 regarding the lack of performance of a 10 CFR 50.59 safety evaluation for a procedure revision that established new operational and configuration parameters for the emergency feedwater (EFW) system. Ultimately, the licensee properly evaluated the concern, determined the EFW system configuration established by the procedure revision to be outside the design- and licensing-bases of the station, and reported the occurrence to the NRC in accordance with 10 CFR 50.72 and 50.73 reporting criteria. However, because initial corrective actions were neither prompt nor adequate, the failure to perform an adequate evaluation when the subject procedure was revised is cited as a violation of the requirements of 10 CFR 50.59.

- Control room operators alertly identified initial indications of potential primary reactor coolant system to secondary system leakage. Chemistry samples verified the presence of a minor leak of approximately 1.0 gallons per day (gpd) in the "C" steam generator (SG), which is a small fraction of the technical specification limits of 500 gpd through any one SG and 1.0 gallons per minute (gpm) through all four SGs. Throughout the report period, the leakage rate varied from less than detectable levels to approximately 1.0 gpd. Station management developed a well-coordinated response to the initial leakage, and established conservative administrative response actions in the event of increased leakage.

Maintenance:

- System engineers effectively evaluated the potential adverse effects of oil intrusion into the emergency diesel generator (EDG) barring device micro-switches.

Continuing evaluation identified a component upgrade to make the switches less vulnerable to foreign material intrusion.

- The work plan for the attempted leak sealant repair of a steam line instrument root valve was effectively supported by engineering evaluation that considered appropriate American Society of Mechanical Engineers (ASME) Code requirements and valve design and application information. Good independent quality oversight of the work effort was evidenced when a quality control (QC) inspector identified that a drill stop had been improperly set. Field supervision properly terminated the repair attempt when it was determined that physical interferences would prevent successful completion of the intended repair. A subsequent repair plan was developed and successfully implemented.

- A Licensee Event Report (LER) and its supplement (LER 96-03 and 96-03-01) regarding EFW pump mechanical seal failure properly addressed the reporting criteria of 10 CFR 50.73.

Engineering:

- Diagnostic motor operated valve (MOV) testing of containment building spray (CBS) valve, CBS-V-43, was well controlled. The replacement of the actuator gear set to restore assumed operational performance and design-bases margin was well controlled and effectively implemented.

Plant Support:

- Radiological protection program controls were observed to have been properly implemented. Workers were noted to be complying with established procedures and controls. Chemistry personnel effectively supported the initial verification and subsequent monitoring of the minor primary to secondary system leakage.

- A previously issued safeguards violation (50-443/96-02-03) regarding the lack of a continual behavior observation program for contractors with unescorted access who are absent from the station for extended periods (greater than 30 days) was closed following a review that determined appropriate corrective actions had been implemented to the access authorization and control processes.

Assurance of Quality

- Overall, station management demonstrated good oversight of issues important to safety as well as to corrective action processes. Response to the initial indications of primary to secondary leakage were prompt and well coordinated. Increased monitoring intervals and administrative limits were established that were significantly more conservative than existing station procedural or regulatory requirements.

- The management decision to form a task force to verify or establish appropriate instrument transmitter configurations reflected sound corrective action practices.

- Additionally, controls established to ensure proper conduct of an in-plant temporary leak sealant repair effectively identified and ensured correction of improper tool settings prior to the initiation of the field work.

- However, station management did not aggressively ensure the timely technical resolution of the NSARC concern initially raised in March 1994, regarding the implementation of a revision to the startup feedwater pump (SUFP) operations procedure without performance of a 10 CFR 50.59 safety evaluation. It took approximately two years of persistent effort on the part of the NSARC to heighten the concern sufficiently before a detailed safety evaluation determined the procedure revision had authorized the establishment of an EFW configuration that was outside the original design basis of the system.

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Summary of Plant Status

The facility operated at approximately 100% of rated thermal power throughout the inspection period with routine minor power reductions performed to support instrument calibrations and turbine valve testing. On November 25, 1996, initial indications of primary to secondary leakage were identified.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, routine operations were performed in accordance with station procedures and plant evolutions were completed in a deliberate manner with clear communications and effective oversight by shift supervision. Control room logs accurately reflected plant activities and observed shift turnovers were comprehensive and thoroughly addressed questions posed by the oncoming crew. Control room operators displayed good questioning perspectives prior to releasing work activities for field implementation. The inspectors found that operators were knowledgeable of plant and system status.

O2 Operational Status of Facilities and Equipment

a. Inspection Scope (71707, 62707)

The inspectors routinely conducted independent plant tours and walkdowns of selected portions of safety-related systems during the inspection report period. These activities consisted of the verification that system configurations, power supplies, process parameters, support system availability, and current system operational status were consistent TS requirements and UFSAR descriptions. Additionally, system, component, and general area material conditions and housekeeping status were noted.

O2.1 Transmitter Configuration Control (IFI 50-443/96-10-01)

b. Observations and Findings

On October 8, 1996, during an independent tour of the service water (SW) pump house, the inspectors noted that plastic foreign material exclusion (FME) shipping caps were installed in the spare conduit ports on SW system pressure transmitters, 1-SW-PT-8272 through 8274 and 1-SW-PT-8282 through 8284. This observation was brought the attention of plant management who directed appropriate engineering personnel to review this condition. Engineers subsequently verified that temporary shipping caps were installed in the subject transmitters and ACR 96-1002 was generated.

The SW pump house area where the subject transmitters are located, environmental zone SW-1, is classified as a mild environment with respect to the station environmental qualification program required by 10 CFR 50.49. This classification does not specifically require that moisture accumulation at the terminal side of the transmitter housing be prevented by means such as a qualified environmental seal. Nonetheless, the vendor manual for the subject transmitter model, Rosemount 1153 series-manual FP73280, directs that unused conduit ports be closed off with a stainless steel pipe plug, with threads sealed by pipe thread sealant. Therefore, work request, WR 96W002012 was initiated to replace the plastic FME shipping plugs with stainless steel pipe plugs per the transmitter manufacturer specifications. The ACR evaluation was expanded by the licensee to inspect and verify or establish as necessary the proper configuration of a population of 523 currently installed process transmitters. The licensee concluded this action was appropriate due to the diversity of transmitter models and designs that could be installed in various service applications. A dedicated task force was established, with projected milestones of December 15, 1996 for the completion of in-plant inspections and January 13, 1997 for project completion and task force report issuance.

The inspectors have met with the task force leader several times to verify project scope, assurance of quality, incorporation of potential lessons learned into proper program processes, and to maintain status of observations. At the conclusion of the inspection report period, the task force had not identified any transmitter discrepancy that effected instrument operability or environmental qualification. However, several minor deficiencies such as loose junction box door clamps, missing qualification tags, and missing end caps for instrument drain line tubes had been identified.

c. Conclusions

The shipping caps that the inspector identified installed in the spare conduit ports of the SW pressure transmitters have been in place since original installation. Although these caps do not conform with vendor specifications, their installation in this specific application does not effect instrument operability or qualification.

The licensee demonstrated excellent causal and corrective action analysis by establishing a dedicated task force to provide positive verification of the design-bases configuration requirements for a significant sample population of transmitters currently installed in the station. Review of the final task force report and any subsequent corrective actions will be tracked by an inspector follow-up item (IFI 50-443/96-10-01).

O2.2 Turbine-Driven Emergency Feedwater Pump Exhaust Line Painting

b. Observations and Findings

On October 11, 1996, during a system walkdown with the system engineer, the inspector identified that the turbine-driven EFW (TDEFW) pump steam exhaust piping had been painted blue. The color code procedure, SM 7.2, "Station Labeling

Program," indicates that this piping should be painted white. More significantly, the inspector questioned if the paint was qualified to the temperatures expected to be experienced in the exhaust line during design-bases operation of the TDEFW pump. ACR 96-1040 was initiated.

The licensee provided a turbine specification sheet that indicated full load exhaust temperature would be 235° Fahrenheit (F). Vendor specifications for the paint indicated the primer coat was temperature resistance rated to 350° F and the finish coat of colored paints are rated to 250° F.

c. Conclusions

The incorrect paint color coating for the TDEFW pump turbine exhaust line was of negligible significance. Additionally, the inspector verified that the labeling program had properly considered process piping temperatures in determining the chemical qualities for the paint products. The inspector had no further questions regarding this issue.

O2.3 Indication of Primary to Secondary System Leakage

a. Inspection Scope (71707)

On November 25, 1996, control room operators noted indications of potential primary or reactor coolant system to secondary system leakage. At approximately 3:00 a.m., operators were preparing to regenerate a blowdown system cation filter bed when the "C" SG blowdown radiation monitor alarm briefly went into the alert range. The alarm quickly cleared, however at approximately 8:00 a.m., the alarm again went into alert and remained at slightly above normal readings. The operators promptly notified chemistry and entered the off-normal procedure OS 1227.02, "Steam Generator Tube Leak." Chemistry initiated secondary system sampling with increased frequency in accordance with procedure, CS 0905.08A. The inspector reviewed Technical Specification (TS) 3.4.6.2, associated with reactor coolant system leakage, procedure OS 1227.02, NUREG/CR-6365 "Steam Generator Tube Failures," and held discussions with Chemistry Department personnel and senior station management.

b. Observations and Findings

Chemistry confirmed the presence of a primary to secondary leak in the "C" SG by use of gaseous isotopic analysis and determined the leak rate was very small (approximately 1.0 gpd). TS 3.4.6.2 limits reactor coolant system leakage to 1.0 gpm total reactor-to-secondary leakage through all Sgs and 500 gpd through any one SG. The licensee promptly and aggressively began a thorough evaluation of the indications of SG tube leakage. Senior station management provided strong oversight of and involvement with evaluation of the tube leak and development of corresponding response strategies and contingencies. The station sought the most current industry experience on SG tube leaks and contacted SG experts at Westinghouse. The licensee established sampling and operational controls that were significantly more conservative than both general industry standards that had been established by the Electric Power Research Institute (EPRI) guidance (150 gpd total leakage, rate of change of greater than 60 gpd in one hour shutdown criteria) and TS requirements (500 gpd). An administrative limit of 60 gpd for initiating a plant shutdown was imposed. Additionally, contingency repair plans were under development to repair the source of the leakage during the next refueling outage or sooner if conditions should sooner necessitate a forced shutdown. Frequent briefings were conducted to inform station management of the status. At the conclusion of the inspection report period, the SG tube leak had stabilized at approximately 0.2 gpd.

c. Conclusions

The inspectors determined the licensee response to the tube leak was prompt, thorough, and conservative which demonstrated sound safety perspectives. The licensee used recent industry information when developing response strategies, which was considered very prudent given the industry experience with SG tube leaks and the sometimes rapid leak rate increase from a few gpd to several hundred in a short period of time. The operators alertly responded to the radiation data management system (RDMS) alarms. The inspectors observed good coordination between the chemistry, health physics, and operations departments. Senior station management involvement was strong and ensured appropriate station response. The inspectors had no further questions.

07 Quality Assurance in Operations

07.1 (Closed) LER 96-04, "Emergency Feedwater System Valve Closure" (VIO 50-443/96-10-02)

a. Inspection Scope (71707,40500)

On July 24, 1996, the licensee submitted LER 96-04 which documented that evaluations, completed on June 27, 1996, had concluded that the alignment of the startup feedwater pump (SUFP) to the steam generators (Sgs) via the emergency feedwater (EFW) flow control valves for non-emergency operation was an unanalyzed condition that significantly compromised safety. On June 27, 1996, a one-hour non-emergency notification was made to the NRC operations Center in accordance with the requirements of 10 CFR 50.72.

The inspector reviewed numerous documents that recorded the development of this issue from the initial identification of a concern in 1991 through issuance of LER 96-04. Documents of significant import included:

- Station Information Report (SIR) 91-036, dated December 6, 1991
- Root Cause Analysis for SIR 91-036, No. 91-013, dated January 31, 1992
- UFSAR Section 6.8, "Emergency Feedwater System"
- TS Section 3.7.1.2, "Auxiliary Feedwater System"
- ON 1034.03, "Condensate System Operation" Rev 3, change 13

- OS1035.02, "Startup Feed Pump Operation," Rev 7, change 01-04
- TS Clarification, TS-148, "Emergency Feedwater System Operation"
- ACR 95-511, dated December 13, 1995
- Yankee Atomic Memo, "Review of ACR 95-511," dated May 24, 1996.

Additionally, several Nuclear Safety Audit and Review Committee (NSARC) meeting minute reports and correspondence to and from the NSARC and plant staff were reviewed.

b. Observations and Findings

On April 2, 1991, feedwater check valve, FW-V-330, did not stroke freely with the plant maneuvering in Mode 3 (HOT STANDBY). The normal condensate and feedwater flow path was required to be isolated in order to facilitate repairs to the valve. Rather than placing the reactor in a shutdown condition, the licensee established a configuration in which the SUFP was isolated and the steam generators were fed by the condensate pumps via the EFW system, with level being maintained by the EFW flow control and isolation valves. The configuration was established by a revision (Change 13 to Revision 3) to station procedure ON 1034.03, "Condensate System Operation." The plant remained in this configuration for approximately three days until repairs to the feedwater check valve were completed and the normal feedwater flow path was restored. During this period of time, the plant was in the action statement of TS 3.7.1.2, applicable in Modes 1,2,and 3, which limits continued operation in this configuration to 72 hours.

On December 6, 1991, SIR 91-036 was generated due to questions regarding the potential for the off-normal feedwater alignment to be in conflict with high energy line break evaluations. A root cause analysis evaluation for SIR 91-036 (91-013, dated January 31, 1992) concluded that the procedure revision that authorized the off-normal feedwater alignment lacked an adequate 10 CFR 50.59 evaluation and that inconsistencies existed between various station manuals and procedures that govern the performance of 10 CFR 50.59 evaluations and procedure revisions. In part due to SIR 91-036, Technical Clarification (TC), TS-148, was issued May 28, 1992 and revised August 21, 1992, that further supported operation of the EFW system non-emergency SG level control in the off-normal feedwater alignment.

On January 27, 1994, station procedure, OS 1035.02, "Startup Feed Pump Operation," was revised (Change 01 to Revision 07) to authorize operation of the SUFP and EFW flow control and isolation valves to maintain SG water level during normal plant operations. TS-148 was referenced as technical support for the procedure revision. Shortly after, on March 30, 1994, the NSARC Operations Subcommittee (NSARC OC) questioned the adequacy of the procedure revision evaluation at NSARC OC meeting No. 94-01, as documented in NSARC OC meeting minutes memorandum dated April 8, 1994. Specifically, the NSARC OC expressed concern that the procedure revision was implemented without the performance of a 10 CFR 50.59 evaluation. Further, the NSARC OC expressed concern that the revision used a TC to support the revision. Licensing management subsequently issued memorandum, LIC 94-0396, dated June 16, 1994, that clearly defined the controls, instruction, and limitations of a given TC.

As a result of continued apparent differing opinions between the NSARC OC and plant staff regarding the need for a 10 CFR 50.59 evaluation for the newly developed EFW system operation, the NSARC elevated the issue to the attention of the Station Manager for SORC resolution in the August 23, 1994 NSARC meeting (94-11), as documented in the meeting minutes report dated August 23, 1994.

Two more NSARC meetings passed without resolution of the concern. It was not until April 13, 1995, that the licensee confirmed the intention to perform a 10 CFR 50.59 evaluation. The evaluation was performed in June 1995, however the NSARC assessed the evaluation as weak, lacking in technical as well as design- and licensing-bases information. Due to the apparent lack of a success path on this issue, NSARC issued ACR 95-511 on December 13, 1995.

Ultimately on May 1, 1996, at the direction of the Senior Site Officer acting as the NSARC Chairman, Yankee Atomic conducted a comprehensive evaluation of ACR 95-511 as well as the historical documents on the procedure change. The evaluation, "Review of ACR 95-511," dated May 24, 1996, concluded that the procedure revision placed the EFW system in a configuration that was outside the design- and current licensing basis for the plant. The evaluation determined that; operation of the EFW system in non-emergency modes contradicted past licensee statements and supporting licensing documents; operation with the EFW flow control and isolation valves throttled for non-emergency SG level control was not evaluated with respect to the UFSAR; and, TS 4.7.1.2.1.a.1 requires that the EFW flow control and isolation valves be open at all times during non-emergency plant operations. The specific vulnerability involved a scenario in which the EFW flow control and isolation valves would be shut to support the starting sequence of the SUFP coincident with the receipt of a station blackout.

Licensee evaluation concurred with the Yankee Atomic review of ACR 95-511, and on June 27, 1996, the licensee notified the NRC in accordance with 10 CFR 50.72 of the unanalyzed EFW alignment. Immediate corrective actions included deleting use of the EFW system for non-emergency SG level control. Other corrective actions were directed at clarifying 10 CFR 50.59 requirements, issuing standing orders that would reinforce that EFW flow control and isolation valves should not be throttled, and revision to TS-148.

c. <u>Conclusions</u>

UFSAR Section 6.8.1 states that the design basis for the EFW system is to provide the capability to remove heat from the reactor coolant system during <u>emergency conditions</u> when the main feedwater system is not available, including small LOCA cases. Additionally, for station blackout, the turbine-driven EFW pump will operate during the four-hour coping period to cool down and maintain the secondary side pressure at about 250 psig. This section further states that for all other modes of plant operation, including startup, hot standby, and normal operation up to full power load, the EFW system is depressurized and has zero flow.

Additionally, the UFSAR Section 6.8.2 description of the EFW system states that the open position of the flow control valves for system limiting conditions will be set to insure the minimum flow of 470 gpm to three Sgs and a minimum total flow of 650 gpm to four Sgs with one EFW pump operable. Further, the UFSAR Section 6.8.3 safety evaluation states that the flow regulating valves in each EFW line are normally open, and are sized to pass the required flow under accident conditions. The only action necessary to establish EFW flow is to start the pumps.

TS 3.7.1.2 states that in Modes 1, 2, and 3 at least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be operable.

The licensee [illegible]visions to station procedures, ON 1034.03, "Condensate System Operation," [illegible] April 1991 and OS 1035.02, "Startup Feed Pump Operation," in January 199[illegible], established procedural guidance authorizing non-emergency feedwater flow paths to the Sgs through the EFW system, with SG level being controlled by throttling the EFW flow control and isolation valves. The procedures were revised without performance of safety evaluations consistent with the requirements of 10 CFR 50.59 to determine if the changes involved an unreviewed safety question as defined in 10 CFR 50.59. It is recognized that the plant has actually been operated in the off-normal configuration authorized by ON 1034.03 one time from April 2-5, 1991 and that the off-normal alignment authorized by OS 1035.02 has never been established. Notwithstanding, the failure to perform safety evaluations to determine if revision of procedures ON 1034.03 and OS 1035.02 involved an unresolved safety question as defined by 10 CFR 50.59 is a violation. (VIO 50-443/96-10-02)

Several related aspects of this issue are of concern to the inspectors. Initially, the NSARC and NSARC OS expressed obvious concern for the lack of a 10 CFR 50.59 evaluation with respect to the OS 1035.02 revision at meetings for approximately two years beginning in March 1994, without satisfactory resolution. Ultimately, the Senior Site Officer and NSARC chairman, on May 1, 1996, directed that an immediate resolution to the concern be achieved. Secondly, the licensee originally justified the revision on the basis of a technical clarification, TS-148. When a subsequent 10 CFR 50.59 safety evaluation was performed it was qualitative in nature, lacking technical rigor supported by design and licensing basis information. Finally, the corrective action processes, including resolution to NSARC issues through initiation and resolution of associated ACRs, similarly lacked technical rigor and were not prompt or timely. These issues were discussed in detail with licensee management and will be addressed in the licensee response to the attached Notice of Violation.

II. Maintenance

M1 Conduct of Maintenance

M1.1 EDG Barring Device Micro-switch

a. Inspection Scope (62707)

On October 7, 1996, the inspector observed portions of the "A" EDG load test being performed per station procedure EX 1804.001.

b. Observations and Findings

During a walkdown of the EDGs, the inspector noted that engine lube oil was dripping directly onto the barring device lower micro-switches that are designed to prevent a start of the EDG with the barring device engaged. There is an upper (DGA/B-ZS-BD1) and lower micro-switch (DGA/B-ZS-BD2) for the barring device on each engine; only the lower micro-switch would be vulnerable to the oil leakage from the barring device. This observation was brought to the attention control room personnel who directed appropriate system engineering personnel to evaluate this observation.

System engineers verified the inspector observation and initiated ACR 96-1067. The licensee evaluation concluded that oil could enter the micro-switch casing and potentially prevent the switch from functioning. The oil is non-conductive and does not present an electrical short circuit potential. The primary micro-switch reliability concern would be that the switch may remain in its current state position and not respond to a change in barring device status. This concern presents minimal safety concern because the upper micro-switch would be capable of actuating the protective parallel circuit and prior to use of the barring device, the associated EDG would be removed from service and other administrative and physical controls would prevent automatic engine startup. Also should micro-switch remain in the engaged position after the barring device is removed from the EDG, a common EDG local annunciator and control room computer digital alarm point (Drawings 31087, SH-E93/8a,b,f 1-NHY-503490, 506393, and 9763-M-510000) would provide indication of the switch failure prior to returning the EDG to service.

System engineers contacted the micro-switch vendor, who indicated an optional rubber boot was available to minimize the potential of foreign materials entering the switch. An engineering work request was initiated for design engineering personnel to evaluate installing the rubber boot seals. Additionally, WRs 96W002024 through 96W002027 were generated to correct the source of the oil leakage and to inspect the micro-switches.

c. Conclusions

It appeared that the minor oil leakage from the barring device had been present for some time prior to being brought to the attention of the licensee by the inspector.

The inspector reviewed micro-switch logic diagrams and concurred in the licensee conclusion that failure of a micro-switch would be of minimal personnel or equipment safety significance. Subsequent engineering evaluation of the condition was thorough and identified a potential component enhancement involving the rubber boot seal for the micro-switches.

M1.2 Leak Sealant repair

a. Inspection Scope (62707,37551)

Previously during the second refueling outage (ORO2) main steam instrument root valve, MS-V-56 was repacked with undersized packing. MS-V-56 is the root valve for steam pressure transmitter FW-PT-544, which inputs to the safety injection and steam line isolation two-out-of-three low steam line pressure logic. The transmitter also provides post accident monitoring capability. The licensee controlled the root valve on its backseat to limit packing leakage, however over the course of the current operating cycle the packing performance had degraded. WR 96W000845 was generated to perform a temporary leak sealant injection repair of the packing gland. The inspector reviewed the work request, applicable sections of Section XI of the ASME Code governing repair of ASME class components, station procedure, MS 0526.09, "On Stream Leak Repairs," discussed specific work-related controls with cognizant engineers, and directly observed portions of the field work.

b. Observations and Findings

On October 29, 1996, the licensee held a pre-evolution briefing, placed transmitter, FW-PT-544, in bypass, and initiated the work activity. The actual physical work was performed by contractor technicians with direct licensee supervision and quality oversight. The injection preparation was a two step process involving an initial drill of 0.312 inches to a depth of approximately 0.360 inches for the installation of an injection valve followed by a center drill into the packing gland. The depth of the initial drill was controlled by setting a drill stop. At a pre-determined procedure hold point, license QC personnel alertly identified that the contractor technician had improperly set the drill stop. The stop was properly set and the initial drill was performed. However, due to physical interferences, the drill was not perpendicular to the packing gland and the minimum specified injection valve thread engagement of 3 turns could not be obtained.

The licensee supervisor suspended the work effort, engineering personnel evaluated further repair options, and ACR 96-1114 was initiated. An initial operability determination (OD) that was largely qualitative was rejected by operations management. A subsequent OD, ODF No. 96-015, dated October 30, 1996, that was supported by design-basis assumptions and calculations was approved. The OD concluded that with the drilled hole the valve maintains its original design requirements and remains operable.

On November 13, 1996, the licensee implemented WR 96W002299, that performed an ASME Section XI code repair of the drilled hole. Following completion of the code repair, the WR directed closure of the root valve and sequential removal of the packing gland follower nuts and cleaning of the follower stud threads and attempting to make up all the free travel on the follower. This effort proved successful and leakage from the packing gland was stopped.

c. Conclusions

The initial packing gland leak repair plan was well developed, however it was of a high degree of difficulty due to physical interferences. The repair plan was discussed in detail with NRC Region I ASME Code and temporary leak sealant repair specialists during a conference call on November 1, 1996. Licensee QC personnel provided excellent independent oversight as evidenced by the identification of the improperly set drill stop.

Operations section management demonstrated good standards by rejecting the initial OD that lacked design- and operational-bases technical evaluation. The subsequent OD was properly supported and approved.

The repair plan for the unsuccessful leak sealant attempt was well developed. Notwithstanding, it appears that the work planning process initially overlooked the less intrusive and technically challenging repair option of gland follower stud thread cleaning which ultimately proved successful, in favor of the leak sealant repair option. The inspector had no additional questions regarding this activity.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) LER 96-03 and LER 96-03-01, "Emergency Feed Water Pump Mechanical Seal Replacement"

On May 21, 1996, with the plant operating at 100% of rated thermal power the TDEFW pump was rendered inoperable when sparks were observed emanating from the outboard mechanical seal area during quarterly flow surveillance testing. The seal was disassembled and determined to have been improperly installed during the last refueling outage in November-December 1995. The licensee performed a formal root cause analysis which concluded a mechanical seal design deficiency and inadequate corrective actions for a previously identified event as the primary causes for the event. A contributing cause was inadequate predictive maintenance techniques. NRC Inspection Report 50-443/96-04 documented the event including a Notice of Violation for the improper seal assembly.

The LER properly documented the event and contained the appropriate reporting requirements of 10 CFR 50.72. Any further NRC inspection of this event will be documented in follow-up of the violation and violation corrective actions response documents. The inspector had no further questions regarding closeout of the LER.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 MOV Diagnostic Testing

a. Inspection Scope (62707,37551)

On October 22, 1996, the licensee removed motor operated valve, CBS-V-43, from service to perform diagnostic testing. The MOV actuator is a Limitorque SMB-000 model. The design-bases functions of the valve are to automatically open on a "P" signal to allow sodium hydroxide to gravity feed to the refueling water storage tank (RWST) and to close to terminate transfer of the spray additive to the RWST. The inspector reviewed TS 3.6.2.2.b, RTS 96RM19204001, WR 96W002181, and discussed various aspects of the work activity with involved personnel.

b. Observations and Findings

Initially, the valve was removed from service and the appropriate actions per TS 3.6.2.2.b, that limit continued operation with CBS-V-43 inoperable to 72 hours, were initiated. Performing technicians identified minor discrepancies for limit switch settings for valve position indicating lights. Additionally, technicians identified that the Bellville washer stack for the spring pack was not properly configured. Each of these discrepancies were corrected prior to completion of diagnostic testing. However, subsequent diagnostic testing identified that peak unseating force exceeded predicted values. In order to restore design-bases margins to the actuator, the motor pinion and worm shaft gear sets were replaced. Post-maintenance diagnostic testing indicated that with the new gear sets installed that approximately 114% of predicted design margins were restored to the actuator.

c. Conclusions

The licensee appropriately and conservatively addressed each discrepancy and test deficiency encountered. Replacement of the valve actuator gear set reflected sound understanding and respect for reliability and design margin goals of the MOV program established in response to NRC Generic Letter 89-10. The testing results and corrective actions were discussed with the NRC Region I MOV specialist during an October 24, 1996 conference call.

Notwithstanding the above noteworthy technical performance, the inspectors concluded that the licensee could have better planned for a contingency to replace the actuator gear set based on other recent plant experience. Specifically, diagnostic testing of the redundant flow path spray additive valve, CBS-V-38, had recently been conducted with similar as-found reductions in design margins. In the case of the CBS-V-38 valve adequate margins were restored by torque switch and limit switch adjustments. Based on the similarity of valve design, operational application, and test experience, the licensee should have anticipated similar as-found test data for the CBS-V-43 valve. However, the on-line maintenance planning was largely focused on diagnostic test performance with an absence of meaningful initial engineering assessment for the potential for design margin reductions that would necessitate detailed reactive engineering support. The net effect of the lack of contingency planning was a probable increase in the time that CBS-V-43 remained out of service. It should be noted that CBS-V-43 was returned to service within the TS 3.6.2.2.b allowed outage time. The inspector had no additional questions regarding this maintenance activity.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments

a. Inspection Scope

During the inspection period the inspector toured the radiologically controlled area (RCA) on several occasions to observe radiological controls practices.

b. Observations and Findings

The Seabrook Station radiological controls technicians at the RCA checkpoint were attentive and provided assistance to radiation workers to assure proper work practices were used when radiation workers signed in and out of the RCA. The inspector determined that radiation area postings were proper and well marked and survey results were current and posted properly. All personnel observed were properly wearing dosimetry while in the RCA. A sampling of high radiation area doors identified no discrepancies with locking or posting requirements.

c. Conclusions

The inspector determined that Seabrook Station was properly implementing the station radiological controls program requirements in the areas inspected. Radiological controls personnel were knowledgeable of station procedures and provided good oversight of radiation workers. Department managers were observed in the field observing and supervising department personnel.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comment (71707, 71750)

The inspectors observed security force performance during inspection activities. Protected area access controls were found to be properly implemented during random observations. Proper escort control of visitors was observed. Security officers were alert and attentive to their duties. Of particular note, the inspectors consistently found that security force members performing compensatory post duties during security computer power supply upgrades to be fully aware of watch requirements.

P7 Quality Assurance in Security and Safeguards Activities

P7.1 Closure of Previously Identified Violation

(Closed) VIO 50-443/96-02-03: The licensee failed to effectively implement a developed procedure which addressed contractors with unescorted access into the protected area that are away from Seabrook Station for more than 30 days and have not been under a continual behavioral program.

With respect to the above violation, the inspectors determined that the corrective actions described in the licensee's May 24, 1996 letter, in response to the NRC's Notice of Violation, were reasonable, complete, and appeared to be effective.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management, following the conclusion of the inspection period, on December 23, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X3 Other NRC Activities

Conference calls between NRC managers and technical staff specialists and licensee managers and technical staff leads were performed on the following dates.

- October 24, 1996-To discuss technical aspects of the MOV diagnostic test results and subsequent repairs to CBS-V-43. (Section E2.1)

- November 1, 1996-To discuss technical aspects of the temporary leak sealant repair of main steam instrument root valve, MS-V-56. (Section M1.2)

- November 15, 1996-To perform a final exit for an MOV inspection. The results of the inspection will be documented in a combined NRC Inspection Report.

- November 27, 1996-To discuss technical aspects of the indication of primary to secondary leakage. (Section O4.1)

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. Diprofio, Unit Director
G. Kline, Technical Support Manager
R. White, Design Engineering Manager
J. Peterson, Maintenance Manager
J. Grillo, Operations Manager
B. Seymour, Security Manager
W. Leland, Chemistry and Health Physics Manager

NRC

Albert W. DeAgazio, Project Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73051:	Inservice Inspection - Review of Program
IP 73753:	Inservice Inspection
IP 83729:	Occupational Exposure During Extended Outages
IP 83750:	Occupational Exposure
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902:	Followup - Engineering
IP 92903:	Followup - Maintenance
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Inspector Follow-up Item 50-443/96-10-01, "Review of Transmitter Configuration Task Force Report"

Violation 50-443/96-08-01, "Failure To Perform 10 CFR 50.59 Safety Evaluation"

Closed

LER 96-04, "Emergency Feedwater System Valve Closure," dated July 24, 1996
LER 96-03, "Emergency Feedwater Pump Mechanical Seal Replacement," as supplemented September 12, 1996
Violation 50-443/96-02-03, "Failure to establish access controls for contractors with unescorted access who would be absent from the site for more than 30 days

Discussed

None

LIST OF ACRONYMS USED

ACR	Adverse Condition Report
ASME	American Society of Mechanical Engineers
CAS	Central Alarm Station
CBS	Containment Building Spray
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
FME	Foreign Material Exclusion
gpd	gallons per day
gpm	gallons per minute
LCO	Limiting Condition for Operation
MOV	motor operated valve
MPCS	Main Plant Computer System
NSARC	Nuclear Safety and Audit Review Committee
NSARC OS	NSARC Operations Subcommittee
psig	pounds per square inch gauge
QC	Quality Control
RHR	Residual Heat Removal
SG	Steam Generator
SIR	Station Information Report
SORC	Station Operations Review Committee
SUFP	Startup Feedwater Pump
SW	Service Water
TDEFW	Turbine Driven Emergency Feedwater Pump
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WR	Work Request