ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Attachment:	Partial List of Persons Contacted List of Inspection Procedures Used List of Items Opened and Closed List of Documents Reviewed

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/96-11

This team inspection evaluated the current effectiveness of the WNP-2 system and design engineering organizations to respond to routine and reactive site activities which included the identification and resolution of technical issues and problems. This inspection also evaluated the performance of safety and operability evaluations. The inspection covered a 4-week period with two of these weeks conducted onsite.

Engineering

- The temporary modification program and the problem evaluation request program were effectively implemented (Sections E1.1.2 and E1.1.3).

- The permanent plant modification program was not effectively implemented due to the use of technical evaluation requests to perform certain plant modifications. Specifically, the plant modification process was not followed in that technical evaluation requests were used to perform some permanent plant modifications that were not equivalent changes. This was considered to be a violation for a failure to adhere to the requirements of plant procedures (Section E1.1.4).

- The safety-related status of the reactor core isolation cooling system could not be determined. This system was downgraded to nonsafety-related status by the licensee, but no documentation was provided to show that NRC approval for this downgrade had been granted (Section E1.1.4).

- Plant configuration control was not maintained when a technical evaluation request was used to: 1) modify a valve's lever arm clearance; 2) install piping composed of both carbon steel and stainless steel; and, 3) replace a welded pipe hanger with a bolted pipe hanger. In addition, controlled drawings associated with the plant modification for the installation of the carbon steel and stainless steel piping had not been revised. This was considered to be a violation for a failure to maintain plant configuration control (Sections E1.1.4 and E1.1.5).

- The problem with the standby gas treatment system involving the inability for the system to provide the necessary negative pressure in the secondary containment under certain environmental conditions, required a Justification for Continued Operation. This Justification for Continued Operation was not resolved even though the issue was identified by the licensee on September 29, 1989 (Section E1.1.6).

- Discrepancies between the Final Safety Analysis Report and actual plant conditions were minor with the exception of conflicting information regarding the safety-related status of the reactor core isolation cooling system (Section E2.1.1).

- The decision to terminate the design basis document program had the potential to affect the ability to control plant design changes as evidenced by the difficulty in retrieving design basis information for the standby gas treatment and control room ventilation systems (Section E2.1.2).

- While the 10 CFR 50.59 safety evaluations had some weaknesses, no safety evaluations that were reviewed were considered to be inadequate (Section E2.2).

- Overall plant housekeeping and equipment material condition were good with the exception of scaffolding material stored near safety-related equipment without the performance of the engineering evaluation required by plant procedures. This was considered to be another example of the violation for a failure to adhere to plant procedures (Section E2.3).

- Although the size of the engineering backlog was not unreasonable, a large percentage of the open items did not have completion dates. Initiatives have been implemented to reduce and control this backlog (E2.4).

- Several corrective actions to resolve long-standing issues were either inadequate or untimely. Specifically: 1) the corrective actions to resolve repeated failures of floor drain radioactive system primary containment isolation valves FDR-V3 and V4 were ineffective and resulted in repeated failures of these valves; 2) the corrective actions to resolve the failure of the Corporate Nuclear Safety Review Board to review all 10 CFR 50.59 safety evaluations were inadequate which resulted in an additional failure of the Corporate Nuclear Safety Review Board to review a 10 CFR 50.59 safety evaluation; and, 3) the corrective actions to correct a fire protection issue identified in an NRC violation were not timely in that all of the corrective actions had not been completed over a year after the violation occurred. The failure to provide adequate and timely corrective actions was considered to be a violation (Sections E2.5, E3, and E8.2).

- Two inconsistencies between the procedure directing the performance of 10 CFR 50.59 safety evaluations and the requirements of 10 CFR 50.59 were identified. One allowed changes requiring a Technical Specification revision or an unreviewed safety question to be implemented prior to NRC review and approval. The other did not require the performance of safety evaluations for accidents not previously analyzed if their likelihood for occurrence was less than accidents previously analyzed. These differences did not affect any reviewed safety evaluations (Section E3).

- Engineer training and qualification was effective toward supporting engineering activities. The implementation of a management certification course was a proactive approach toward improving engineer knowledge of integrated plant operations (Section E5.1).

- Training and qualification for 10 CFR 50.59 preparers and reviewers was considered adequate. While recently implemented refresher training was an important adjunct to this training, no frequency for this training was established and there was no program to periodically audit training effectiveness (Section E5.2).

- The overall performance of the system engineering organization was improving due to management initiatives to focus system engineer involvement toward resolving issues related to their systems (Section E6).

- Design engineering was effectively interfacing with maintenance and operations to ensure resolution of plant problems and that plant modifications were properly installed (Section E6.2).

- Probabilistic risk assessment was effectively integrated into plant planning activities (Section E6.2).

- Engineering was proactive in their response to an industry event at the Salem Station by immediately inspecting the Magna-Blast circuit breakers installed in the plant (Section E6.2).

- Engineering had implementing procedures established to maintain equipment and systems not covered by Technical Specifications but that were important for safe operation of the plant (Section E6.2).

- The Nuclear Assurance Division was aggressive in seeking out areas needing improvement and was also responsive to events and inputs from outside sources (Section E6.3).

- A written procedure was not established that described the activities of the Nuclear Safety Assurance Division as required by the Technical Specifications. The failure to establish this procedure was considered to be a violation (Section E1.4).

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Report Details

E1 Conduct of Engineering

E1.1 System Reviews (37550)

The team reviewed four safety-related systems to verify the licensee's ability to maintain these systems in an operable status. The four systems reviewed were: reactor core isolation cooling, standby gas treatment, 125 Vdc, and control room ventilation. The team reviewed the adequacy of the licensee's plant modification processes (permanent and temporary), engineering calculations, problem evaluation requests, and technical evaluation requests.

E1.1.1 Permanent Plant Modification Review

a. Inspection Scope

The team reviewed seven safety-related plant modification records to verify conformance with applicable installation and testing requirements as prescribed by procedures. Specific attributes reviewed and/or verified by the team included: 10 CFR 50.59 safety evaluations, post-modification testing requirements, safety-related drawing updates, Final Safety Analysis Report updates, training requirements, and field installation.

b. Observations and Findings

The team found that all seven permanent plant modifications had a proper 10 CFR 50.59 screening or safety evaluation performed and that none represented an unreviewed safety question. The team also found that the described post-modification testing requirements were adequate to assure component operability. The team verified that affected drawings and procedures were updated for the plant modification records. In addition, the team verified that the physical installations of Plant Modification Records 90-0057-0, 92-0095-0, and 94-0319-0 were consistent with the description in the modification packages.

E1.1.2 Temporary Plant Modification Review

a. Inspection Scope

The team reviewed three safety-related temporary plant modifications (95-030, 95-035, and 95-043) to verify conformance with applicable installation and testing requirements as prescribed by licensee procedures. Specific attributes reviewed by the team included: 10 CFR 50.59 evaluations, post-modification testing requirements, safety-related drawing updates, training requirements, field installation, and the process for periodically reviewing the status of the modifications. In addition, the team selected three additional temporary modifications (94-018, 95-001, and 96-012) to determine whether affected control room drawings were properly annotated signifying that a temporary modification was outstanding.

b. Observations and Findings

The team found that Temporary Modification Requests 95-030, 95-035, and 95-043 had been performed with proper safety evaluations and that post-modification testing requirements were properly specified. The team verified that the field installations were as described in the requests. The team also verified that the seven affected control room drawings associated with all six temporary modifications were properly annotated identifying the presence of an outstanding temporary modification. The licensee was tracking the open temporary modifications to assure closure. The team found that all the open temporary modifications either had a closure document assigned or the closure method determined.

E1.1.3 Problem Evaluation Request Review

a. Inspection Scope

The licensee issued problem evaluation requests as a means to identify problems with components and systems and to place these problems in their corrective action system for resolution. The team reviewed 53 problem evaluation requests associated with the four subject systems to determine the adequacy of the resolution, whether the systems' operability was properly determined, and that the proposed corrective actions were adequate to preclude recurrence.

b. Observations and Findings

The team found that the problem evaluation requests had resolutions with proper engineering justification and that proposed corrective actions were adequate to preclude recurrence.

E1.1.4 Technical Evaluation Request Review

a. Inspection Scope

Technical evaluation requests were used to request technical evaluations, document the evaluation, recommend action, and obtain management concurrence. These evaluations could be used to perform plant modifications which were considered equivalent changes and would not constitute a plant modification record or a minor design change. The team reviewed 12 technical evaluation requests associated with the four subject systems to determine whether proper engineering resolution was performed and that issues requiring the use of the plant modification process were properly identified.

b. Observations and Findings

Failure to Follow Plant Modification Procedure

The team found three examples where plant modifications were performed using the technical evaluation process instead of the project modification record or minor modification processes as required by procedure. The three examples were:

- Technical Evaluation Request 94-0264 modified a standby liquid control pipe hanger from a welded connection to a bolted connection. The team considered this to be a complex modification since it significantly impacted stress analysis calculations.

- Technical Evaluation Request 94-0306 replaced the ASME carbon steel drain line downstream of valve RCIC-V-26 with stainless steel piping and modified the pipe supports. The team considered this to be a complex modification since the modification required configuration changes to pipe supports and involved revisions to complicated stress analysis calculations which significantly impacted the original calculations.

- Technical Evaluation Request 96-0004 changed vacuum breaker valves RCIC-V-111/112 from carbon steel lift check valves to stainless steel swing check valves. The team considered this to be a complex modification since the changes significantly impacted design calculations due to the change in valve type and the 56 pounds of additional weight for each of the new valves.

Plant Procedures Manual 1.4.1, "Plant Modifications," Revision 22, was the governing procedure for the implementation of permanent plant modifications. The team noted that the use of a technical evaluation request to perform certain permanent plant modifications, which were considered to be equivalent changes, was allowed by Procedure 1.4.1. However, the team found that the use of the technical evaluation request process did not provide the same level of review and oversight that the plant modification record or minor modification processes afforded. The licensee defined, in Section 4.2 of Manual 1.4.1, an equivalent change as a hardware change that results in installation or modification of an item or items, not identical to the original, which meets the design basis requirements of both the item or items and applicable interfaces. Section 10.1.a stated that if the system engineer determines the modification to be an equivalent change, that the equivalent change evaluation be performed in accordance with Technical Services Instruction TI 1.2, "Equivalent Change Evaluations". Instruction TI 1.2 stated that the equivalent change process was not to be used for complex plant modifications or when formal calculations are significantly impacted.

10 CFR Part 50, Appendix B, Criterion V, requires activities to be conducted in accordance with procedures. The three technical evaluation requests reviewed were permanent plant modifications that either involved a complex plant modification or significantly impacted calculations. The failure to follow Procedure 1.4.1 is considered to be the first example of a violation involving a failure to comply with the requirements of 10 CFR Part 50, Appendix B, Criterion V (50-397/9611-01).

Reactor Core Isolation Cooling System Seismic Qualification

The team also found that two plant modifications performed under a technical evaluation request for the reactor core isolation cooling system did not provide for a seismic qualification analysis. Specifically, Technical Evaluation Requests 96-0046 and 96-0125 allowed modifications to the reactor core isolation cooling turbine without considering the effect of the modification on the seismic qualification of the turbine components. The licensee stated that the reactor core isolation cooling system was downgraded from a safety-related to a nonsafety-related status in 1985, which changed the seismic qualification of the system from seismic Category I to non-seismic. This downgrading was the result of a modification to the automatic depressurization system which allowed the safety function of the reactor core isolation cooling system to be enveloped by the automatic depressurization system. The team noted, however, that Chapters 3, 5, and 7 of the Final Safety Analysis Report specified that the reactor core isolation cooling system components were seismic Category I. After discussions with the licensee, the team noted that there had been no NRC correspondence provided to the licensee that approved this downgrading of the reactor core isolation cooling system classification.

The team will refer this issue to the Office of Nuclear Reactor Regulation to determine whether approval had been given to the licensee to downgrade the reactor core isolation cooling system to nonsafety-related status via a task interface agreement. This item is considered to be unresolved pending completion of this review by the Office of Nuclear Reactor Regulation (50-395/9611-02).

Inadequate Plant Configuration Control

Other reviews of technical evaluation requests by the team identified an instance where plant configuration control was not being maintained. Technical Evaluation Request 96-0125 documented a modification to increase the clearances at the lever arm pivot for valve RCIC-V-2 by removing one of the two washers which separated the remote servo arm from either side of the lever arm clevis. In addition, shims were placed on the governor's remote servo unit to ensure proper alignment of the remote servo arm and the governor lever arm. The team found that the licensee had not performed calculations to determine whether the new configuration met the vendor's requirements for clearances. In addition, the team found that the licensee had not determined whether environmental and seismic requirements were met and whether the shim material was compatible with the remote servo unit's materials.

10 CFR Part 50, Appendix B, Criterion III, requires that plant design bases be properly supported by design specifications. The failure to have design specifications for the installed plant configuration was the first example of a failure to maintain plant configuration control and was considered to be the first example of a violation of Criterion III of 10 CFR Part 50, Appendix B (50-397/9611-03).

E1.1.5 Review of Engineering Calculations

a. Inspection Scope

The team reviewed the adequacy of seven design engineering calculations associated with the four subject systems to determine whether the calculation assumptions were technically reasonable and properly supported.

b. Observations and Findings

The team found that five of the seven calculations were adequate. However, the team also found that two of the calculations did not support the installed in-plant configuration.

Calculation CMR-96-0128 associated with Technical Evaluation Request 94-0264-0, evaluated the structural analysis of changing a welded connection to a bolted connection for a standby liquid control system pipe hanger. The team found that the design stress calculation covered the welded connection, but not the installed bolted connection. As a result of this finding, the licensee performed a calculation that indicated that the current configuration was within the design basis.

Calculation CMR-95-0292 associated with Technical Evaluation Request 94-0306-0, evaluated the structural analysis of the reactor core isolation cooling system drain piping. This piping was modified from all carbon steel piping to a carbon steel/stainless steel combination. The team found that while the licensee's calculation covered an all carbon steel configuration or an all stainless steel configuration, the calculation did not cover the carbon steel/stainless steel configuration that was presently installed in the plant. As a result of this finding, the licensee performed an operability evaluation with a supporting design calculation and determined that the installed piping structural analysis was acceptable. The team reviewed the licensee's calculation and concurred with the results.

Furthermore, during the check to verify that drawings affected by the design changes had been revised, the team found that the piping isometric drawings for the technical evaluation request had not been revised.

10 CFR Part 50, Appendix B, Criterion III, requires that plant design configurations be supported by appropriate design analyses. The failure to have design analyses for the installed plant configurations was considered to be two additional examples of violation (50-397/9611-03) identified in Section E1.1.4 of this report.

E1.1.6 Review of Justification for Continued Operation of the Standby Gas Treatment System

a. Inspection Scope

During the review of the standby gas treatment system, the team was informed by the licensee that an outstanding Justification for Continued Operation was in effect. The team reviewed this document to determine why it was still in effect, if there was an operability concern, and the status of corrective actions taken.

b. Observations and Findings

In mid-1987, the Niagara Mohawk Corporation issued a Licensee Event Report for their Nine Mile Point Unit 2 plant regarding non-conservative standby gas treatment system operation. Specifically, the Licensee Event Report reported that the assumptions used to evaluate secondary containment pressure draw-down time following a postulated loss of offsite power combined with a loss of coolant accident were not conservative. As the result of this report, the licensee reviewed their draw-down time calculations for the same accident conditions. From this review, the licensee determined that their calculations were not conservative in that an assumed failure of certain emergency power buses could cause a delay or an inability to achieve the required secondary containment negative pressure. In addition, they determined that their calculations did not consider certain wind conditions that affected the amount of secondary containment leakage.

The team reviewed the licensee's Final Safety Analysis Report requirement for standby gas treatment system performance and determined that the system was designed to reestablish the secondary containment to a negative pressure of -0.25 inches water gage within 120 seconds of initiation after the loss of coolant accident event. The purpose for this requirement was to stop an unfiltered secondary containment release. Since this requirement could not be accomplished under the loss of coolant accident conditions when combined with the loss of offsite power event, the licensee issued Revision 0 of a Justification for Continued Operation on September 29, 1989. A copy of this Justification for Continued Operation was sent to the NRC to provide the NRC with early notification of this issue since the licensee determined that it involved an unreviewed safety question. The Justification for Continued Operation documented that secondary containment draw-down and leakage would be maintained when actual data was used in lieu of postulated data, upon which the original design criteria was based, to determine the draw-down times and secondary containment leakage rates. The purpose of this Justification for Continued Operation was to allow continued plant operation while this condition was in effect.

On January 3, 1990, the NRC responded to the licensee's September 29 letter. This response stated that the licensee had provided sufficient justification to allow continued operation. On February 16, 1990, the licensee sent a letter to the NRC that discussed a program plan for resolution of this issue. The plan included preparation of a secondary containment model to determine the wind and temperature conditions for which a defined secondary containment draw-down could be obtained and the required licensing document changes. On December 22, 1992, the licensee issued another letter to the NRC, which discussed changes for the resolution of the secondary containment issue that was presented in the February 16, 1990, letter. On February 6, 1995, a meeting was held between the licensee and the NRC. This meeting was documented in a March 6, 1995, letter to the NRC. This letter discussed the proposed changes to the standby gas treatment system design basis which included increasing the Technical Specification required minimum flow rate from 4460 cfm to 5000 cfm and increasing the draw-down time to achieve the secondary containment negative pressure from 2 minutes to 20 minutes. This letter also documented that a formal submittal requesting NRC review of the revised design basis would be sent in the spring of 1995.

The team's review of these letters indicated that the Justification for Continued Operation was still in effect, though it was revised several times as new computer programs were developed to perform refined calculations. The current revision was Revision 5. This review also indicated that the formal licensee submittal was not made. The team also noted that the licensee's actions were inconsistent with the guidance provided in NRC Generic Letter 91-18. The generic letter guidance for a Justification for Continued Operation specified that the Justification for Continued Operation be resolved in a timely manner. The team did not consider the approximate seven year period to resolve this issue to be timely.

The team discussed this Justification for Continued Operation with both licensee and NRC personnel who had been involved with this issue. The licensee stated that no equipment changes were required to accomplish the proposed standby gas treatment system operation and that only Technical Specification and Final Safety Analysis Report changes were necessary. These changes were necessary to reflect the new standby gas treatment draw-down time and flow rates that were determined by the calculations based upon actual system operation. In addition, the licensee was testing the standby gas treatment system to assure that system operation under both the present Technical Specifications and the proposed revised Technical Specifications was consistent with the present licensing conditions and their revised system analysis.

E1.1.7 Conclusions on System Reviews

The permanent plant modification process was not being effectively implemented because technical evaluation requests were being used to perform plant modifications that were not equivalent. The use of this process resulted in a reduction in review and oversight afforded to permanent plant modifications performed as a plant modification record or a minor design change. The use of the temporary evaluation request process resulted in design calculations not being performed for the current plant configuration and drawings affected by the modification not being revised. The temporary plant modification process was effectively implemented. Licensee actions to resolve a seven-year-old Justification for Continued Operation for the standby gas treatment system were inconsistent with the guidance in Generic Letter 91-18 since the issue was not resolved in a timely manner, however, no operability concern was identified.

E2 Engineering Support of Facilities and Equipment

E2.1 License and Design Basis Documents (37550)

E2.1.1 Review of Facility and Equipment Conformance to the Final Safety Analysis Report Description

a. Inspection Scope

A recent discovery of a licensee operating its facility in a manner contrary to the Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the Safety Analysis Report description. While performing the inspections discussed in this inspection report, the inspectors reviewed the applicable sections of the Final Safety Analysis Report that related to the selected inspection areas.

b. Observations and Findings

The team found that the Final Safety Analysis Report was generally consistent with the actual plant configuration, however, the team found one major discrepancy involving the inconsistency in describing whether the reactor core isolation cooling system was safety-related and seismic Category I. This issue is described in Section E1.1.4.b of this report.

E2.1.2 Review of Design Basis Documents

a. Inspection Scope

The team reviewed the design basis documents for the reactor core isolation cooling system, the standby gas treatment system, the 125 Vdc system, and the control room ventilation system to verify the validity of the design basis and determine the ease of retrieving the information.

b. Observations and Findings

The team noted that, of the four systems reviewed, only the reactor core isolation cooling and 125 Vdc systems had design basis documents (which the licensee called design requirements documents). The team also noted that the licensee had difficulty retrieving the design basis information for the two systems which did not have design requirements documents. This was evidenced during the inspection by the fact that it took a number of days (including the intermediate week that the team was not onsite) for the licensee's staff to retrieve information requested by the team. The licensee attributed this information retrieval delay to the fact that design requirement documents for the standby gas treatment system and the control room ventilation system were not available.

The licensee's basis for developing design requirements documents was the result of their commitment in response to NRC Inspection Report 50-397/87-19. In their response to this report, the licensee committed to undertake a program to develop and organize design basis documents for 17 safety systems, 40 non-safety systems and 20 safety-significant topical subjects. The licensee informed the team that the original design basis documents were all available, but were not readily retrievable. Therefore, the licensee developed a document for specific safety systems (i.e., the design requirements document) to make the design basis documents more easily available in one location. The licensee developed design requirements documents for 21 out of 54 systems.

The licensee's basis for discontinuing the design requirements documents was explained in interoffice memorandum entitled "White Paper: DRD/SSFI Program Scope Reduction," dated November 22, 1994. The memorandum stated that the 21 design requirements documents developed encompassed 19 safety systems, 29 non-safety systems and 11 safety-significant topical subjects. The memorandum stated that the results of developing these design requirements documents had provided reasonable assurance that the original design basis for the plant would be maintained, and that no programmatic problems were experienced with their design bases. The memorandum concluded that the objective of the action item, which was to verify a sound design basis, had been met. Therefore, the licensee decided to discontinue the design requirements documents program and re-direct their resources to better support plant operational needs.

E2.1.3 Conclusions

Discrepancies between the Final Safety Analysis Report and actual plant conditions were minor, with the exception of whether the reactor core isolation cooling system was safety-related and seismic Category 1. The team concluded that the difficulty the licensee had in retrieving design basis documents for the standby gas treatment and control room ventilation systems indicated that the licensee's decision to terminate the design requirements program adversely affected the ability to retrieve plant design information.

E2.2 10 CFR 50.59 Implementation (37001)

a. Inspection Scope

The team reviewed 12 safety evaluations performed in accordance with 10 CFR 50.59 and three 10 CFR 50.59 screenings which concluded that 10 CFR 50.59 was not applicable to the design change.

The team also attended a plant operations committee meeting while onsite. The plant operations committee was the organization tasked with review and approval of all 10 CFR 50.59 safety evaluations in accordance with Procedure 1.1.5, "Plant Operations Committee", and the Technical Specifications.

b. Observations and Findings

The team determined by review of documents that two of the safety evaluations were of very good quality. The documentation for Safety Evaluations 95-001 and 96-2B was sufficiently detailed and the conclusions logically supported such that safety evaluations properly considered the safety impacts of the changes on structures, systems, and components considered important to safety. However, the team identified the following problems with four of the safety evaluations:

- Safety Evaluation 95-80, written for a temporary modification to remove the reactor core isolation cooling turbine electrical overspeed trip, stated that the turbine was important to safety because it was credited for mitigation of the anticipated transient without scram event. However, the same safety evaluation also stated that the anticipated transient without scram event was analyzed assuming that reactor core isolation cooling turbine was not available.

- Safety Evaluation 95-086 was written for Safety Analysis Report Change Notice 94-077 and the associated plant procedure changes. The screening document for the safety evaluation described the deletion of the requirement in Final Safety Analysis Report Table F.3-2, that the five-person fire brigade contain three operations personnel and a change to the requirement that the fire brigade only contain the fire brigade leader and two members that were knowledgeable of plant fire safe shutdown systems. The change evaluated in the safety evaluation and accompanying fire protection engineering evaluation was different than specified in the screening document, in that the change only replaced one operations person with a health physics technician.

- Safety Evaluation 95-025, associated with Safety Analysis Report Change Notice 95-013, was written for relocation of the postulated break for standby liquid control system piping. While the safety evaluation acknowledged that the relocation of the break may cause some equipment located in the vicinity of the break to be damaged, the effect of this damage was not further evaluated other than concluding that the plant would still be able to safely shutdown.

- In Safety Evaluation 95-040, which was related to a design modification to reflect downgrading of the reactor water cleanup system from ASME Code Group C to Group D, the evaluation referred to several quality group standards and determined that they were not applicable without providing information regarding how these standards affected the safety evaluation.

The team also observed that during the review of these safety evaluations, many of the forms were outdated. This was not a significant safety concern.

During attendance at the plant operations committee meeting, the team determined by interviews with the plant operations committee secretary and the chairman that the discussion materials had been previously sent to the plant operations committee members for review. Additional copies of agenda item packages were also available. The team observed that the discussions were limited to those items on the agenda and that any additional discussions were rescheduled for a future meeting. The team noted that the committee's discussion of a previously unidentified safety-related aspect of a proposed change resulted in the change not being approved until further discussion at a future meeting. The team did not identify any failures to follow the procedural requirements of Procedure 1.1.5, "Plant Operations Committee," during the meeting.

c. Conclusions

The team concluded that the licensee's 10 CFR 50.59 safety evaluations had some weaknesses; however, no safety evaluations reviewed by the team were considered to be inadequate.

E2.3 System Walkdowns (37550)

a. Inspection Scope

The team performed a walkdown of the four subject systems and other selected plant areas to determine the overall material condition of equipment and maintenance of housekeeping.

b. Observations and Findings

During the plant walkdowns, the team found that effective housekeeping was being maintained. However, the team also observed some conditions that had the potential to affect the operability of safety-related equipment during a seismic event. Specifically, the following was observed:

- The grated area above reactor core isolation cooling Pump P-1 had a significant amount of unsecured scaffolding material stored approximately 10 feet from an opening directly above the pump; and,

- A significant amount of unsecured scaffolding material was found approximately 20 feet from residual heat removal Pump 2C.

The licensee confirmed these observations and discovered that similarly unsecured scaffolding material was being stored in the rooms housing residual heat removal Pumps 2A and 2B.

The team reviewed Maintenance Programs and Procedure 10.2.53, "Seismic Requirements for Scaffolding, Ladders, Man-Lifts, Tool Gang Boxes, Hoists, and Metal Storage Cabinets," Revision 14, to determine the requirements for storing scaffolding material near safety-related equipment. The team found that Section 4.4 of Procedure 10.2.53 required that all scaffolding be left in an acceptable seismic configuration and, if it did not meet procedural requirements, that an engineering evaluation be performed. In addition, Section 7.1.6 required that, for a partially erected or removed scaffold, engineering complete an evaluation which included a 10 CFR 50.59 safety evaluation. The team found that the required engineering evaluations had not been performed for the stored scaffolding material in the observed plant areas. The licensee subsequently removed the stored scaffolding.

10 CFR 50 Appendix B, Criterion V, requires adherence to plant procedures. The failure to follow the requirements of Procedure 10.2.53 was considered to be contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion V. This violation was considered to be the second example of violation (50-397/9611-01) identified in Section E1.1.4 of this report.

c. Conclusions

Overall, plant housekeeping and equipment material condition were effective with the exception of the storage of scaffolding material near safety-related equipment without an engineering evaluation being performed.

E2.4 Engineering Work Backlog (37550)

a. Inspection Scope

The team discussed the status of the engineering backlog with the system engineering manager and the assistant to the engineering general manager. The discussions included actions taken by the engineering organization to reduce the backlog. The team reviewed a sampling of 289 open problem evaluation reports and requests for technical services to determine if the disposition of any had been inappropriately deferred.

b. Observations and Findings

As of the completion of this inspection there were approximately 2800 open items in the engineering backlog. Of these approximately 1300 were greater than 18 months old and over 1400 had no scheduled completion date. From discussions with licensee personnel, the team determined that staff reductions in the engineering organization over the past several years had a negative impact on the engineering workload backlog.

To more effectively control the backlog the licensee implemented several initiatives. A recent reorganization increased the size of the system engineering department to reduce the workload and affect a reduction in backlogged activities. The engineering organization also tasked itself with establishing credible due dates for all open items not currently scheduled. Establishment of due dates and priorities was scheduled to be completed by the middle of August. Additionally, as part of the engineering organization's performance enhancement strategy, goals were established and promulgated regarding reduction in the engineering backlog.

A review of 222 problem evaluation reports and 67 requests for technical services that had yet to be dispositioned did not identify any actions that had been improperly deferred.

c. Conclusions

Although the licensee has not been effective at controlling the engineering backlog, a review of selected open items did not identify any safety issues that were improperly deferred. The licensee implemented several initiatives to reduce the engineering backlog.

E2.5 Failure of Floor Drain Radioactive System Primary Containment Isolation Valves, FDR V-3 and V-4 (37550)

a. Inspection Scope

During the inspection the team was informed by the resident inspectors of ongoing failures involving containment isolation valves FDR V-3 and V-4. Since the resolution of these failures provided a prospective of engineering involvement in plant activities, the team included this issue in the inspection scope.

On July 6, 1996, both floor drain radioactive system primary containment isolation valves, FDR V-3 and V-4, exceeded the stroke time action limits established in accordance with ASME testing requirements. Valve FDR V-3 subsequently failed a local leak rate test on July 12. The inspectors reviewed the licensee's actions to correct these deficiencies and reviewed the history of failures for these valves. The inspectors interviewed the system engineer and the system engineering manager in that regard.

b. Observations and Findings

Valves FDR V-3 and V-4 provide for primary containment isolation of the floor drain system piping from the drywell sump. These valves are air operated 3" ball valves and are normally open. They are in series with each other and a flow transmitter (FDR FT-38) used for measuring unidentified reactor water leakage into the drywell and are isolated from the drywell by means of a loop seal. The valves automatically close on a containment isolation signal.

Technical Specification 4.0.5 requires the testing of valves FDR V-3 and V-4 in accordance with ASME code requirements. The ASME code requires valves FDR V-3 and V-4 to be stroke tested on a quarterly basis. The code requires the stoke time to be measured during each test and compared to an initial reference stroke time. The code also requires that the measured stroke time have no more than a $\pm$50 percent change from the reference value. The reference closing stroke times for valves FDR V-3 and V-4 are 2 and 3 seconds respectively. The maximum closing time for these valves to meet accident considerations, as defined by Technical Specification 3.6.3, is 15 seconds.

Problem Evaluation Request 296-0045 was initiated on January 20, 1996, when valve FDR V-4 failed to close during surveillance testing. The surveillance test was being performed as part of the licensee's In-Service Testing Program in accordance with Plant Procedures Manual 7.4.0.5.6B, "FDR Valve Operability", Revision 0. The action statement for Technical Specification 3.6.3 was entered and valve FDR V-3 was deactivated in the closed position.

The licensee evaluated the cause of this event and concluded that the failure was a result of foreign material in the valve body. In the disposition of Plant Evaluation Request 296-0045, the licensee documented that the water in the floor drain piping loop seal contained a significant amount of particulate material. The report also documented that the accumulation of this material in the valve body and on the valve seat caused the failure of valve FDR V-4 to close.

The corrective actions developed in response to this failure were to perform a high pressure flush of the valve and to perform weekly stroke time testing of the valve for at least four weeks to assess any adverse performance trends. The valve was declared operable following the high pressure flush when subsequent stroke times consistently met the acceptance criterion. The system engineer evaluated the results of the additional stroke time testing and concluded that there were no adverse performance trends. However, the system engineer also recommended two additional corrective actions: (1) rework the air operator to valve FDR V-4, and, (2) clean the accumulated sludge from the FDR drywell sump.

The reworking of the valve actuator was based upon the system engineer's judgement that, while there was no clear correlation between the measured stroke times and length of time between stroke time surveillances, the engineer noted that the stroke times consistently decreased when the valve was stroked multiple times during the surveillances. Based upon these test results, the engineer concluded that the variability in the measured stroke times was as much a function of the status of valve FDR V-4 air operator as it was with any accumulation of foreign material in the valve body or on the valve seat. Based upon the system engineer's recommendation, the air operator for valve FDR V-4 was reworked during the April-May 1996 refueling outage.

On April 26, 1996, during refueling outage local leak rate testing, Problem Evaluation Request 296-0301 was initiated to address the failure of valve FDR V-3 to meet the ASME seat leakage rate limit of 442 sccm. The as-found leakage rate was 742 sccm. The licensee's determination of the probable cause of the failed leak rate test was foreign material on the valve seating surfaces. The valve was disassembled, inspected and than reassembled with new seat assemblies on May 18, 1996. The post-maintenance as-left testing performed on May 27, 1996, was satisfactory.

The cleaning of the FDR drywell sump was planned to be conducted during the refueling outage in conjunction with the cleaning already planned by the radiation protection department. The cleaning planned by the radiation protection department was intended to reduce area dose rates in the sump area for ALARA purposes. However, during the refueling outage, radiation protection personnel surveyed the FDR drywell sump and determined that the radiological conditions in the sump were acceptable for ALARA purposes and that cleaning of the sump was unnecessary. During this survey, to assist in determining the amount of foreign material in the sump, a small amount of particulate material was obtained from the floor of the sump. Based upon the survey and material sample results, radiation protection and engineering personnel agreed that cleaning of the sump was not necessary.

The use of a high pressure flush to clean the loop seal was also considered by engineering; however, this flush was not considered to be practicable due to system design limitations. Since a high pressure flush was not practicable, the normal draindown evolution of the sump that occurred at the end of the refueling outage was also considered. Engineering concluded that the flow velocities from such a draindown would be sufficient to sweep foreign material from the loop seal. The sump draindown was performed on May 29, 1996, two days after the as-left testing of valves FDR V-3 and V-4. No additional testing of these valves was performed prior to the testing on July 6.

On July 6, 1996, during quarterly ASME stroke testing, valves FDR V-3 and V-4 again failed their tests with stroke times of 8 and 15 seconds respectively. The valves were declared inoperable and isolated to comply with the action statement requirements of technical specification 3.6.3. Both valves were flushed and cycled a number of times before obtaining acceptable stroke times. Again, the licensee concluded that the cause of the long stroke times was foreign material in the valve bodies and on the valve seating surfaces. In a followup effort to ensure the operability of valves FDR V-3 and V-4, the licensee performed seat leakage rate tests on July 12. Valve FDR V-4 passed its seat leakage rate test. Valve FDR V-3 failed its seat leakage test with a leakage rate in excess of 20,000 sccm. A disassembly of valve FDR V-3 on July 18, 1996, again found foreign material on the seating surfaces of the valve. The valve internals were cleaned, the valve retested, and declared operable.

c. Conclusions

The licensee identified in January and April 1996, that foreign material was causing valves FDR-V3 and V4 to fail either stroke times or seat leakage tests. In addition, the licensee went through a refueling outage without flushing foreign material from the sump, the only corrective action that appeared to resolve the valve failure problem. As a result, the valves again failed tests on July 6 and 12. The team concluded that the licensee's corrective actions to resolve the foreign material problem were inadequate. 10 CFR Part 50, Criterion XVI, requires corrective actions for conditions adverse to quality to be adequate to correct the condition and to preclude repetition of the condition. The failure to develop and implement adequate corrective actions to correct the problem and prevent recurrence was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion XVI, and is considered to be a violation (50-397/9611-04).

E3 Engineering Procedures and Documentation

a. Inspection Scope (37001)

The team reviewed the licensee's procedures for implementing the requirements of 10 CFR 50.59. The purpose of this review was to verify that the licensee's procedures provided formal procedural guidance for implementing the requirements of 10 CFR 50.59 for proposed changes, tests and experiments.

b. Observations and Findings

The team verified that Administrative Procedure 1.3.43, "Licensing Basis Impact Determinations," Revision 9, provided adequate guidance for:

- Screening to determine if 10 CFR 50.59 applies and if a safety evaluation is required;

- The preparation of a safety evaluation;

- Whether the change would impact other regulatory programs or commitments and require a separate evaluation (emergency planning, fire protection, security plan, quality assurance, operator qualification or environmental protection programs); and,

- The assignment of specific responsibilities for conduct of activities related to 10 CFR 50.59 reviews to qualified preparer/reviewers.

The team also verified that the licensee's guidance for preparation of a safety evaluation addressed the three criteria for determining if an unreviewed safety question exists. The team did not identify any instances where screenings, licensing basis impact determinations, or 10 CFR 50.59 safety evaluations were performed or reviewed by those not on the list of qualified preparers/reviewers.

During a review of licensee Surveillance Report 294-008, dated February 18, 1994, the team noted a failure to conduct activities in accordance with Procedure 1.3.43. Report 294-008 identified that the Corporate Nuclear Safety Review Board was not receiving all 10 CFR 50.59 safety evaluations for review and approval as required by Technical Specification 6.5.2.7. Seven discrepancies were identified and the licensee wrote Problem Evaluation Request 294-0087 to document and track the corrective action. One of the corrective actions recommended in Problem Evaluation Request 294-0087 was that the 10 CFR 50.59 logbook have a column added so that the Corporate Nuclear Safety Review Board review can be recorded and that Procedure 1.3.43 be revised to require that the administrative assistant record the Corporate Nuclear Safety Review Board and Plant Operations Committee meeting number at which the reviews took place in the appropriate columns. The team verified that the 10 CFR 50.59 logbook and Procedure 1.3.43 had been revised to include these changes.

The team reviewed the 50.59 logbook entries for 1995-1996 and observed that the Corporate Nuclear Safety Review Board logbook column was blank for 1995-1996, and only approximately 50 percent of the Plant Operations Committee logbook entries were made for 1995 and that no logbook entries were made for 1996. The team interviewed the supervisor of regulatory services and verified that the required logbook entries were not made.

To determine if the safety evaluation reviews were being performed, the team reviewed Corporate Nuclear Safety Review Board meeting minutes 96-01 through 96-05 and Plant Operations Committee meeting minutes 96-01 through 96-09 which documented the review of ten 10 CFR 50.59 safety evaluations. The team found that Safety Evaluation 95-095, related to SAR Change Notice 062, involving a change to plant equipment and procedures, had not been reviewed by the Corporate Nuclear Safety Review Board.

The team considered the licensee's corrective actions to prevent the Corporate Nuclear Safety Review Board from missing safety evaluation reviews to be inadequate. 10 CFR Part 50, Criterion XVI, requires corrective actions for conditions adverse to quality to be adequate to correct the condition and to preclude repetition of the condition. The failure to develop and implement adequate corrective actions to correct the problem and prevent recurrence was contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion XVI, and was considered to be the second example of violation (50-397/9611-04) identified in Section E2.5 of this report.

The team also noted two examples during the review of Procedure 1.3.43 where the procedural requirements were inconsistent with the requirements of 10 CFR 50.59. In the first example, Procedure 1.3.43 allowed the Plant Operations Committee to recommend implementation of activities that involve a Technical Specification change or an unreviewed safety question prior to NRC approval so long as the equipment is not declared operable or relied upon for nuclear safety until NRC approval is obtained. The team noted that 10 CFR 50.59 makes no provision for this exception, however, the team did not identify any examples where application of this exception resulted in a safety concern.

In the second example, the guidance for conducting 10 CFR 50.59 safety evaluations allowed that an accident of a new or different type than any accident previously evaluated need not be considered if the likelihood of that accident is less than that of accidents previously evaluated. Again, the team noted that 10 CFR 50.59 did not provide for this exception and stated that an unreviewed safety question exists if the possibility of a new or different type of accident is created, however, the team did not identify any examples where application of this exception resulted in a safety concern.

c. Conclusions

The team determined that, except for the violation involving the inadequate corrective actions to prevent a recurrence of the Corporate Nuclear Safety Review Board missing a safety evaluation review, the licensee had adequate procedures and controls to implement the requirements of 10 CFR 50.59. Some inconsistencies, involving Plant Operations Committee review and approval of safety evaluations and determination of accident considerations, existed between the licensee's procedures and 10 CFR 50.59.

E5 Engineering Staff Training and Qualification

E5.1 System Engineering Staff Training and Qualification (37550)

a. Inspection Scope

A review was conducted of the licensee's training program for the engineering support staff. The team discussed the training program and requirements with nine system engineers during interviews. Training records for the entire system engineering organization were also reviewed.

b. Observations and Findings

Individual qualification guides were established for each engineer and approximately 60 percent of the staff had completed initial qualification training. The relatively high number of personnel that had not completed initial training requirements was attributed to recent staffing changes in the organization and revised requirements that have not yet been fully achieved. One of the relatively new requirements was a 13-week management certification course covering integrated plant operations, a substantial resource commitment. A majority of the engineers had already attended that training and all engineers were scheduled to complete the training by July 1997.

The team interviewed the system engineers responsible for the source range/intermediate range monitoring system, local power range/average power range monitoring system, high pressure core spray system, standby liquid control system, containment atmosphere control system, standby service water system, condenser offgas system, and the reactor pressure vessel system. In general, system engineer knowledge of their assigned systems was good. Most had a clear understanding of the design functions and parameters of the assigned system and the interfaces with other systems.

The team's review of continuing training showed appropriate scope and variation of topics from session to session. Topics covered included Technical Specifications, application of the guidance in NRC Generic Letter 91-18, probabilistic safety assessment, problem evaluation reports and plant and industry events. Three sessions of training were conducted each calendar year. Attendance records showed that a majority of engineers are attending scheduled training and that make-up sessions are conducted for those unable to attend. Read-and-sign training was also an option for those who were unable to attend classroom sessions.

c. Conclusions

The training program for the engineering staff supports the role of the system engineers. The inclusion of the 13-week management certification course as required training for all engineers demonstrated the licensee's commitment to improve engineer knowledge in integrated plant operations.

E5.2 Safety Evaluation Training (37550)

a. Inspection Scope

The team reviewed licensee training records for individuals whose names were contained on the lists of qualified preparers and qualified reviewers performing 10 CFR 50.59 safety evaluations and reviews. The team also reviewed training documents describing the course content of the training courses for these individuals and interviewed two training supervisors.

b. Observations and Findings

The team verified that the individuals whose names were listed on the lists of qualified preparers and qualified reviewers had completed the required training. The team reviewed the content of the training courses for the conduct of 10 CFR 50.59 related activities and determined that the courses provided adequate instruction for the development of safety evaluations including guidance for assessing relevant design information.

The team noted that the initial qualification training had been conducted for most preparer/reviewers in 1990-1991. During interviews, the training supervisors stated that the entire series of training for preparer/reviewers is designed to be conducted in one week and the goal is to have all preparers also qualified as reviewers. The supervisors also stated that refresher training had not been conducted until recently. The team reviewed the training records and confirmed that refresher training was not conducted prior to June of 1995. The training supervisors also had not established a periodic frequency for future refresher training.

The team also noted that the lists of qualified preparers and qualified reviewers contained approximately 300 preparers and 140 reviewers. A review of the 10 CFR 50.59 log for 1995-1996 indicated approximately 180 10 CFR 50.59 screenings and evaluations had been performed during this period. This resulted in an average of less than one per preparer and less than two per reviewer. The team noted during the review of the 10 CFR 50.59 log that, with few exceptions, the preparers had performed two or less 10 CFR 50.59 safety evaluations during the period and that many had not performed any safety evaluations. The team also reviewed 10 CFR 50.59 safety evaluations and noted in several instances that outdated forms were used in the conduct of these reviews.

c. Conclusions

The team concluded that the licensee currently has a training program to train those individuals performing 10 CFR 50.59 related activities. The content of the training program was good and was consistent with the licensee's procedural requirements for performing these activities. However, refresher training was only recently established and a refresher training periodic frequency had not been established.

E6 Engineering Organization and Administration

E6.1 System Engineering (37550)

a. Inspection Scope

The inspector interviewed the system engineering manager, 2 group supervisors, and 13 system engineers. Discussion topics in the interviews included management expectations for the system engineering organization, the processes for identifying and resolving plant equipment deficiencies, and the interface between engineering and other organizations. Additionally, the system engineers were questioned on technical information and outstanding deficiencies for their assigned systems, including actions they were taking to resolve those deficiencies. System walkdowns with the assigned engineer were performed on the service water system, nuclear instrumentation system, high pressure core spray system, reactor core isolation cooling system, standby gas treatment system, 125 Vdc system, and the control room ventilation system. A review was also conducted of the four technical instructions that outlined the conduct of the system engineering organization.

b. Observations and Findings

Expectations of the system engineers and the system engineering program were delineated in several plant technical instructions. These technical instructions outlined the attributes and responsibilities of engineers, and also included expectations for performing periodic system walkdowns and monitoring. The expectations had proper priority and were articulated with little ambiguity. Both supervisors and system engineers were knowledgeable of those expectations.

As part of those expectations, management directed the system engineers to be the focal point for resolution of all technical issues associated with their system. System engineers that were interviewed believed that this concept improved their effectiveness in interfacing with other organizations. Additionally, the team's discussions with operations staff during the system walkdowns indicated that system engineering improved its responsiveness to their need and has taken an active role in communicating with operations personnel regarding current issues and concerns.

Walkdowns and discussions with the system engineers showed that they were knowledgeable of their system's construction and operation and generally familiar with the existing deficiencies that impacted their system. In general, system walkdowns were being performed by the system engineers on a weekly or biweekly basis. The specific frequency and scope of the walkdown was discussed with and agreed upon by their cognitive supervisor. The walkdowns included an evaluation of component material condition and recording of operational parameters for trending purposes.

c. Conclusions

The team noted an improvement in system engineering from that observed during the last engineering inspection documented in NRC Inspection Report 50-397/95-03, dated June 16, 1995. Management initiatives to focus the system engineer's involvement in resolving issues related to their systems showed a commitment toward continued improvement.

E6.2 Design Engineering (37550)

a. Inspection Scope

The team conducted interviews with personnel from the maintenance, operations, and quality assurance departments to evaluate the extent and effectiveness of engineering communications and responsiveness to recent plant problems and modifications. The team also reviewed 13 problem evaluation requests of safety significant issues that required engineering involvement to determine how technical issues were resolved.

In addition, the team reviewed documentation to verify that the licensee was maintaining equipment for systems that were not covered by Technical Specifications to ensure system reliability and operability. This included equipment for systems such as anticipated transient without a scram, station blackout, safety parameter display system, and Regulatory Guide 1.97 instrumentation.

The licensee's integration of probabilistic risk assessment into plant planning activities was also evaluated. In addition, the team reviewed the engineering organization's response to the recent May 3, 1996, event that occurred at the Salem nuclear plant involving Magna-Blast circuit breakers to determine if design engineering was proactive to industry events.

b. Observations and Findings

The team found that the problem evaluation requests had technical resolutions with proper engineering justification and that the proposed corrective actions were adequate to preclude recurrence.

The team interviews with three project and six design engineering personnel revealed that contract project engineers were normally hired for a specific task and, therefore, were not directly involved in the modification process. Contract engineers indicated that cooperation between various organizations was good and that management encouraged identification of plant problems.

To verify the effectiveness of this interface, the team selected Plant Modification Record 94-0364-0, "Drywell Elevation 501 Additional Shield Clips," for review to determine the amount of licensee engineering oversight of contractor activities. The team noted that this modification was designed by Bechtel Corporation and that Bechtel Corporation also supplied field engineers to coordinate the field modifications that were performed by another contractor, Raytheon. The team interviewed the cognizant licensee project engineer regarding engineering involvement with the contractors responsible for their respective work assignments. The team found that the project engineer was actively involved in all phases of the plant modification record which included the preparation and approval of the basic design change package, daily status meetings with the contractor field engineers, and the final walkdown of the modification.

The team found substantial evidence that probabilistic risk assessment, which included the individual plant examination and individual plant examination for external events, was integrated into plant planning activities. For example, Procedure 1.4.1, "Plant Modifications," contained six action requirements addressing probabilistic risk assessment. Probabilistic risk assessment was also considered in scheduling and coordination of plant work, project team meeting notices, and work order instructions. Procedure EDP 2.43, "Configuration Control of IPE Database and Plant Models," outlined the process utilized by the licensee for maintaining configuration control of individual plant examination models used in probabilistic risk assessment applications. The team reviewed Basic Design Change 90-0361-0A, "Spare Service Water Motor Operated Valves," which affected the risk factored into the probabilistic risk assessment, and found that the probabilistic risk assessment was updated to reflect the changed risk profile.

The team found that engineers were appropriately utilizing available design basis documents to determine if a proposed change was within the original design basis. The team noted that the system engineer qualification guide required the system engineer to demonstrate an understanding of the design basis documents. Engineering enhancement training topics included probabilistic risk assessment application.

The team interviewed nine engineering, one operations, two quality assurance, and one maintenance department personnel and found that these departments were effectively communicating to assure that modifications were properly installed, and that plant problems were adequately resolved. Engineers acknowledged that while the recent layoffs had resulted in increased work loads, these workloads continued to be manageable.

The team reviewed documentation to verify that engineering had received information on the multiple failures of the General Electric 4.16 kV Magna-Blast breakers that occurred at the Salem plant on May 3, 1996. The team found that engineering had issued a work order to perform a 100 per cent inspection of the Magna-Blast breakers. The high voltage system engineer had contacted General Electric on May 29, 1996, regarding how to perform a visual inspection to identify if the breakers were prone to the failure mode of the Salem event. The team found that this work was completed on May 31, 1996 with no breaker problems identified.

The team found that surveillance procedures were implemented to ensure the operability of equipment for systems such as anticipated transient without a scram, station blackout, safety parameter display system, and Regulatory Guide 1.97 instrumentation.

c. Conclusions

The team concluded that the licensee was effectively implementing their program to respond to requests for engineering resolution of plant problems and that the licensee had adequate implementing procedures to maintain such equipment and systems that were not covered by Technical Specifications to ensure system reliability and operability. In addition, the team noted that probabilistic risk assessment was affectively integrated into plant planning activities, and that modifications affecting the risk profile were factored into the probabilistic risk assessment. Finally, the team found engineering to have been proactive in their response to the Salem event by immediately inspecting the Magna-Blast circuit breakers installed in the plant.

E6.3 Independent Safety Engineering Group (37550)

a. Inspection Scope

The team evaluated the overall effectiveness of the independent safety engineering group, which the licensee had identified as the Nuclear Safety Assurance Division. The team accomplished this by reviewing the qualifications and authority of the safety engineers performing the quality technical reviews, reviewing the last 12 monthly reports and implementation of corrective action recommendations, and by interviewing members of the Nuclear Safety Assurance Division regarding their day-to-day functions of their organization.

b. Observations and Findings

The team compared Technical Specification 6.2.3, which described the function, composition and responsibilities of the Nuclear Safety Assurance Division to the ongoing activities of this group to determine how the licensee was meeting the Technical Specification requirements. The team found that the licensee did not have a procedure describing how the Technical Specification requirements were being met. The licensee informed the team that the Nuclear Safety Assurance Division was in the process of being merged into the quality department and that the quality services department had issued an interoffice memorandum on March 7, 1996, delineating how the Nuclear Safety Assurance Department functioned and how the Nuclear Safety Assurance Division personnel were incorporated into the quality department functional teams.

Technical Specification 6.2.3 was established to address the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements." NUREG-0737 Section I.B.1.2, "Independent Safety Engineering Group," requires the licensee to establish an onsite independent safety engineering group to perform independent reviews of plant operations. The licensee established the Nuclear Safety Assurance Division to meet this NUREG-0737 requirement. Technical Specification 6.8.1.b requires that written procedures be established, implemented, and maintained, covering activities required to implement the requirements of NUREG-0737. Technical Specification 6.8.1.b applies to the Nuclear Safety Assurance Division. Failure to have a written procedure to implement the requirements of NUREG-0737 is a violation of Technical Specification 6.8.1.b (50-397/96011-05).

The team found that the qualifications and experience of the five dedicated, full-time engineers exceeded the Technical Specification requirements. The team reviewed five monthly reports and two surveillance reports documenting Nuclear Safety Assurance Division activities during the past twelve months. The team found that these reports addressed not only known plant weak areas but also probed into potential problem areas needing improvement.

c. Conclusions

The team concluded that the Nuclear Assurance Division was aggressive in seeking out areas needing improvement. The team identified a violation for not having a written procedure describing the Independent Safety Engineering Group as required by Technical Specifications to implement the requirements of NUREG-0737.

E7 Quality Assurance in Engineering Activities

a. Inspection Scope (37550)

The team reviewed two recent quality assurance audit reports related to engineering activities. Audit Reports 296-016, "Annual WNP-2 Fire Protection Program Audit, dated June 3, 1996, and 296-017, "WNP-2 Design Control Audit," dated March 22, 1996, were reviewed to evaluate the effectiveness of the licensee's controls in identification and resolution of plant problems.

b. Observations and Findings

The team found that the audits were broad in scope and provided meaningful findings and recommendations for potential program enhancements. For instance, the fire protection audit resulted in three problem evaluation requests on material condition of the fire protection equipment for untimely resolution of degraded foam carts, inadequate resolution of repetitive fire control panel ground faults, and one degraded fire pump. The design control audit identified four strengths in engineering performance and provided seven recommendations to further strengthen the engineering design control process. The responses to the quality assurance audits were timely and acceptable.

c. Conclusions

The team concluded that the quality assurance audits reviewed were effective.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Licensee Event Report 50-397/94020: Potential of spurious actuation of pneumatic supply valve to impair the operation of main steam safety relief valves. The licensee postulated that a control room fire could result in closure of motor-operated containment isolation valve CIA-V-20. Closure of this valve could result in depletion of pneumatic supplies for the main steam safety relief valves during the interval between control room evacuation and resuming control at the remote shutdown panels. The containment isolation valve provided nitrogen to the 18 accumulators for the 18 main steam safety relief valves. Six of the 18 valves were equipped with two accumulators, a large one for use only in the automatic depressurization mode and a smaller accumulator for use in the ordinary relief mode. The two remote control panels each contained controls for three of the main steam safety relief valves. One of the remote panels contained controls for three of the main steam safety relief valves designed to be part of the automatic depressurization system. The other remote control panel contained controls for three of the main steam safety relief valves which were not designed as part of the automatic depressurization system. The licensee stated that their commitment required six automatic depressurization main steam safety relief valves to be opened to assure reactor depressurization in time to prevent peak fuel cladding temperatures from exceeding limits. In the case of a control room fire, only three of the six automatic depressurization system main steam safety relief valves could be opened from the remote panels.

The licensee determined that the root cause of the problem was a design deficiency during the original design phase which did not have all six of the automatic depressurization system main steam safety relief valves on the remote control panel. The licensee stated that the event was not safety significant since General

Electric had performed an analysis which showed that the reactor could be depressurized with two main steam safety relief valves while maintaining peak fuel clad temperatures slightly below the threshold for fuel damage.

The team reviewed Plant Modification Record 94-0386, "Change MSRVs Controllable from Alternate Remote Shutdown Panel." The team noted that this modification replaced control of the three non-automatic depressurization system main steam safety relief valves from the alternate shutdown panel with three automatic depressurization system main steam safety relief valves. The modification was completed September 6, 1995.

E8.2 (Closed) Violation 50-397/9518-01: This violation involved the licensee using their fire protection water system for a non-fire protection activity while only a single source of fire water was available. This practice was contrary to the requirements of Procedure 1.3.10, "Fire Protection Program Implementation," and the requirements of Technical Specification 6.8.1 which requires the implementation of plant procedures. Procedure 1.3.10 prohibited fire protection system water to be used for non-fire protection system purposes unless both fire protection system water supplies are available.

The licensee's corrective actions included stopping this practice at the time it was identified, counseling individuals about the importance of procedure compliance, discussions of the event with operating crews by the operations manager, and making the violation response required reading for all licensed and non-licensed operations personnel. During the review of the licensee's corrective action activities, the team noted that the counseling of individuals and discussions with the operations manager had been completed. However, a review of the required reading activities indicated that the required reading was not completed by 50 of the 111 personnel that were required to do the reading.

10 CFR Part 50, Criterion XVI requires corrective actions for conditions adverse to quality to be promptly corrected. The failure to complete the required reading was considered to be the third example of violation (50-397/9611-04) identified in Section E2.5 of this report involving inadequate corrective actions.

E8.3 (Closed) Inspection Followup Item 50-397/9524-02: This inspection followup item involved the removal of nine motor-operated valves from the valve testing program based on the fact that these valves were placed in their non-accident positions only during short-time periodic surveillance testing activities. The inspectors considered the licensee's justification of removal of these valves from the testing program to be inadequate and inconsistent with the operability guidance of Generic Letter 89-10.

This issue is presently being reviewed by the Boiling Water Reactor Owner's Group and the NRC Office of Nuclear Reactor Regulation. Since the results of this review will be documented by the Office of Nuclear Reactor Regulation, no further inspection activity is required at this time.

V. Management Meetings

The team presented the inspection results to members of licensee management at the conclusion of the inspection on July 26, 1996. During the exit meeting the team requested that the licensee docket their commitment date to resolve the standby gas treatment Justification for Continued Operation issue in the near future. The licensee acknowledged the findings presented and agreed to docket the resolution of the standby gas treatment Justification for Continued Operation issue in the near future.

The licensee did not identify that any propriety information was reviewed by the team.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Atkinson, Manager, Reactor and Fuel Engineering
P. Bemis, Vice President, Nuclear Operations
R. Brownlee, Licensing Engineer
D. Coleman, Supervisor, Regulatory Services
L. Fernandez, Licensing Manager
G. Gelhaus, Assistant to the General Manager of Engineering
R. Lulebring, Vice President, Operations Support
D. Mand, Manager, Design and Projects Engineering
T. Meade, Manager, Engineering Programs
J. McDonald, Manager, System Engineering
M. Rice, Electrical Engineer
J. Swailes, General Manager, Engineering

NRC

R. Barr, Senior Resident Inspector
G. Replogle, Resident Inspector
C. VanDenburgh, Chief, Engineering Branch

LIST OF INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 37001	10 CFR 50.59 Safety Evaluation Program
IP 92903	Followup - Engineering

LIST OF ITEMS OPENED AND CLOSED

Opened

50-397/9611-01	VIO	Failure to follow modification and scaffolding procedures
50-397/9611-02	URI	NRR to determine safety-related status of the reactor core isolation cooling system
50-397/9611-03	VIO	Failure to maintain plant design basis
50-397/9611-04	VIO	Failure to implement adequate and timely corrective actions
50-397/9611-05	VIO	Failure to implement a Nuclear Safety Assurance Division procedure

Closed

50-397/94020	LER	Potential of spurious actuation of pneumatic supply valve to impair the operation of main steam safety relief valves
50-397/9518-01	VIO	Failure to have a redundant fire protection water supply while using the fire water supply for non-fire protection activities
50-397/9524-02	IFI	Removal of nine motor operated valves from the valve testing program

LIST OF DOCUMENTS REVIEWED

Plant Procedures

Procedure No.	Revision	Title
1.1.5	19	"Plant Operations Committee"
1.3.9	18	"Temporary Modifications"
1.3.12	22	"Problem Evaluation Request (PER)"
1.3.12A	4	"Processing of Problem Evaluation Requests"
1.3.43	9	"Licensing Basis Impact Determinations"
1.4.1	22	"Plant Modifications"
1.4.5	13	"Technical Specification, FSAR, and ODCM Change Control Process"
1.6.2	20	"Document Control"
1.6.12	2	"Configuration Document Change Request"
1.16.6C	4	"Conduct of Infrequently Performed Tests or Evolutions"
4.10.2.5	1	"Abnormal Operating Procedure - Control Room High Temperature"
EI 2.8	11	"Generating Facility Design Change Process"
EDP 2.36	1	"Partial Implementation"
EDP 2.43	0	"Configuration Control of IPE Database and Plant Models"

TI 1.2	1	"Equivalent Change Evaluations"
TI 2.1	5	"System Engineer Responsibilities"
TI 2.2	5	"System Engineer Walkdowns/Monitoring"
TI 2.3	0	"System Engineer Communication"
TI 2.4	0	"System Engineer Attributes"

Plant Modifications

Number	Title
90-0057-0	"125 VDC Power System, New HPCS Battery Racks"
92-0095-0	"125 VDC Power System, Replace 2GN-13 Cell Units with 2GN-15 Cell Units"
93-0251-0	"SGT System, Removed Brakes from Motor Operated Valves"
94-0043-0	"RCIC System, Eliminate Pressure Locking Susceptibility"
94-0072-0	"Replace Pinion Gear and Shaft Gear on Motor Operated Valve RCIC-V-76"
94-0152-0	"RCIC System, Replace 3400 RPM Motor with 1700 RPM Motor on Motor Operated Valve RCIC-MO-63"
94-0319-0	"CR/HVAC System, Replace Control Room Toilet Fan Motor"
94-0364-0	"Drywell Elevation 501 Additional Shield Clips"
BDC 90-0361-0A	"Spare Service Water Motor Operated Valves"

Temporary Modifications

TM Number	Title
94-018	"Demineralizer Added to the Existing Filter to Reduce Corrosion in Reactor Closed Cooling System"
95-001	"Circulating Water Blind Flange Installed to Support On-Line Tube Plugging"
95-030	"Deactivate Check Valve RCIC-V-66 Position Indication"

95-035	"Remove the Electronic Overspeed Trip Feature from the RCIC Turbine Control Logic by Lifting Leads on the AIRPAX Overspeed Unit"
95-043	"Modify Suction and Discharge Lines from Radiation Monitors for Control Room Ventilation"
96-012	"Replace Potable Water Makeup With Demineralized Water"

Problem Evaluation Requests

PER Number	Title
294-0370	"Relief Valve Failed Set Pressure Test"
294-0429	"Loose Metal Pieces Found in Turbine Steam Trap"
294-0549	"Cracked Plug in Check Valve"
294-0575	"Contacts on Relay Failed to Change Status During Testing"
294-0577	"Work Order Declared Valve Operable Without Post-Modification Testing"
294-0601	"Damaged Wiring on Control Room Ventilation Moisture Element Discovered During Surveillance Testing"
294-0614	"Degraded Motor Leads for Motor Operators"
294-0627	"During Maintenance the Packing Gland and Bracket Blew off of Containment Isolation Valve Under Pressure"
294-0642	"Standby Gas Treatment Fan Failed Operability Test"
294-0694	"During Testing, Train A of Standby Gas Treatment Failed to Achieve -.25 Inch of Water Gage in the Reactor Building in Less than 120 Seconds"
294-0728	"Miswiring During Maintenance on Control Room Differential Pressure Alarm"
294-0742	"Problems Found While Running RCIC Operability Test"
294-0807	"Incorrect Rebuild Kit Installed in Safety-Related Drain Trap"
294-0920	"Relief Valve Lifted During Pump Test"
294-0922	"During Pump Testing RCIC Mini Flow Valve Opened Unexpectedly"

294-0953	"A Number of Errors Found After Surveillance Test"
294-0995	"Standby Gas System Operability Test Failed"
294-1035	"RCIC Turbine Drain Pot Level High Alarm Came in"
294-1092	"Control Room Chiller Failed Differential Pressure Alert Value"
295-0003	"Plastic Equipment Tags were Found Melted and Burned"
295-0189	"DC Meters Found Out of Tolerance"
295-0199	"The Ramp of the RCIC Turbine was Found to be Erratic"
295-0367	"RCIC Turbine Relief Valve Lifted"
295-0519	"As-found Local Leak Rate Tests for Four Penetrations Failed"
295-0558	"AC Input Breaker Tripped During Surveillance Testing of 125 VDC Battery Charger"
295-0752	"Procedure did not Reflect Correct Configuration of Valve RCIC-V-66"
295-0707	"After Work Was Performed on a Primary Containment Boundary, Post-Maintenance Tests were not Performed"
295-0735	"RCIC Mini Flow Valve Cycled When the Pump was Running"
295-0864	"The 10CFR50.59 Review for a Temporary Modification was not Done Correctly"
295-1036	"Check Valve was Sticking Shut"
295-1070	"RCIC Valves were Downgraded"
295-1117	"Licensee Failed to Test Vacuum Pump Check Valve"
295-1161	"Standby Gas Treatment Fan Tripped"
295-1261	"Pipe Support Damaged from Transient Load"
296-0023	"A Pressure Indication Isolation Valve was Found Closed with 80 Pounds of Trapped Pressure"
296-0079	"RCIC High Suction Pressure Alarm Intermittently Alarming"
296-0180	"Work Performed Outside of Approved Troubleshooting Procedure on CCH-CR-1B"

296-0284	"Wind speed indication at 245' (MET-WSR-4, point 2) indicating downscale intermittently"
296-0310	"A Number of Failures of Valve Packing"
296-0324	"While conducting PPM 7.4.0.5.11, MSRV Discharge Vacuum Breaker Operability, valves MS-V-37K, MS-V-37V and MS-V-38G failed"
296-0340	"During MOV diagnostic testing of HPCS-MO-4, the yoke-to-bonnet connection moved by -0.025""
296-0370	"RC-V-40/RCC-V-21 failed ASME leak rate test PPM 7.4.6.1.2.4"
296-0376	"While on rounds, an equipment operator discovered the cabinet L(FP-CP-2B) for fire pump FP-P-2B to be unusually hot"
296-0384	"There have been seventeen work requests, fourteen work orders and twelve MWRs initiated against the fire protection"
296-0397	"Nine 10 pound portable fire extinguishers were observed in the power block with expired annual inspection dates"
296-0400	"RCIC-V-63 valve stem galled as a result of maintenance"
296-0404	"During the DG-3 semi-annual run DMA-FN-31 did not auto start"
296-0429	"During performance of PPM 7.4.3.7.5.1A 'Emergency Ventilation Damper Channel Calibration Test' WOA-V-51D failed to open"
296-0452	"An as found local leak rate test (LLRT) was not performed on RRC-V-19 prior to valve disassembly"
296-0465	"Instrument rack drain manifold valves (TUFLINE, 3-way) have been incorrectly positioned/verified as per direction of PPM 3.1.6"
296-0487	"Primary Containment was Declared Operable and Mode 2 was Entered Without Performing Operability Test"
296-0499	"RCIC-DT-1 Tripped During Test"
296-0508	"Lube Oil Cooler Pressure Control Valve Caused Annunciator to Alarm"
296-0549	"Received a RPV level high trip from channel B concurrently with a S1-2 ground"
296-0575	"This PER documents out of specification level 2 acceptance criteria as required by PPM 8.3.339 for section 8.5 delay time"

Technical Evaluation Requests

TER Number	Title
94-0264-0	"SLC Pipe Hanger Modification with Bolted Connections and Location Change"
94-0306-0	"Replace the RCIC Drip Leg Drain Downstream of Valve RCIC-V-26 with Stainless steel"
95-0013-0	"Replace Relative Humidity Sensor in Control Room Ventilation Duct"
95-0041-0	"Modification to Allow CCH-CR-1A and -1B to Operate in the Automatic Mode"
95-0073-0	"DWC-H-1C, Remove Thermal Overload on Diesel Generator Immersion Heaters"
95-0123-0	"RHR-M-P/3, Change a Thermal Overload for a New RHR Keep Fill Pump"
96-0004-0	"RCIC-V-111/112 Vacuum Breaker Changed from Lift Check to Swing Check"
96-0012-0	"Replace Damper Motor for DMA-AD-21/2"
96-0046-0	"Sample Point for RCIC Lube Oil Cooler was Changed"
96-0047-0	"Replace 1-Inch Pipe with 1.5-Inch Pipe on RCIC Outboard Bearing Housing"
96-0103-0	"Trip Setting for Breaker DMA-42-4A5A, Diesel Fan DMA-FN-31"
96-0125-0	"Increase Clearances at the Lever Arm Pivot for RCIC-V-2"

Calculations

Calculation Number	Title
CMR 94-0517	"Supports PMR 94-0152-0, Replace Motor on RCIC-MO-63"
CMR-95-0292	"Structural Analysis of RCIC Drain Pipe"
CMR 95-0630	"Evaluation of Losing Penetration Between the RCIC Pump Room and the MCC Room"

CMR-96-0128	"Qualify Modification to Support No. SLC-4452-11 Which was Changed Per TER 94-0264-0"
CE-02-91-38	"HPCS Battery Component Calculations"
ME-02-94-32	"Supports PMR 94-0043-0, Pressure Binding Fix"
NE-02-84-24	"Evaluations of RCIC-V-8 and RCIC-V-63 Closing Time on Isolation of RCIC Pipe Break Events"

Training Documents

Program Course Code	Title
82-TSR-0400-LP Rev 4	"Licensing Basis Impact Determination Process"
82-TSR-0900-LP Rev 0	"FSAR Overview"
82-TSR-4000-LP Rev 0	"Introduction to Analysis in the FSAR"
82-TSR-1700-LP Rev 0	"Nuclear Codes, Standards and Regulations"

10 CFR 50.59 Safety Evaluations

Change Document	System Name/Title	Safety Evaluation No.
BDC 55-2927-OA	Downgrading the non-safety-related RWCU system	95-040
BDC 89-0356-OA	Replace MSIV solenoid pilot valves	95-001
BDC 91-0293-OA	MSLRM scram and MSIV trips removal	94-202 and 95-035
BDC 93-0089-OA	RPV level reference leg valve installation	93-142
TMR 95-0035	RCIC electrical overspeed trip removal	95-80
TER 96-0100	Relocation of ODCM alarm/main plant vent release monitor	SE 96-41
TER 96-0093-0	Installation of jet pump restrainer wedges	96-39

TER 96-0009-0	Identified/unidentified leakage instrument Bailey card relocation	SE 96-2B
ISCR 1280	Main condenser vacuum monitor instrument setpoint change	SE 96-14
SCN 95-053	Revision to Diesel Generator load tables 8.3-1, 8.3-2, and 8.3-3	95-026
SCN 95-013	Relocation of SLC pipe break location	95-025
SCN 94-077	Fire brigade training and composition	95-086

Miscellaneous Documents

Technical Specifications

Final Safety Analysis Report

Technical Memorandum TM-1158, "WNP-2 Control Room HVAC Design Basis Control Room Habitability During Postulated Events," Revision 4

Design Specification 15A.4, "General Heating, Ventilating and Air Conditioning Equipment," Revision 1

Design Specification 15B.2, "Essential Heating, Ventilating and Air Conditioning Systems," Revision 0

Mechanical Engineering Criteria 1E, Revision 0

Engineering Directorate Manual EDP 2.50, "Generating facility Minor Design Change Process," Revision 1

License Training System Descriptions, "Control Room, Cable Room and Critical Switchgear Rooms - HVAC (CR-HVAC)," August 1995

AC/DC Electrical Distribution System Verification Checklist

Design Specification for AC/DC Electrical Distribution System

Design Specification for Reactor Core Isolation Cooling System

White Paper describing the Equivalent Change Process

White Paper describing the DRD/SSFI Program Scope Reduction