

APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-382/94-26

License: NPF-38

Licensee: Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana

Facility Name: Waterford Steam Electric Station, Unit 3

Inspection At: Taft, Louisiana

Inspection Conducted: November 14 through December 2, 1994

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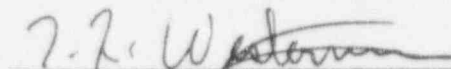
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1-6-95
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EXECUTIVE SUMMARY

This team inspection was conducted to assess the overall performance of the engineering organizations at the Waterford Steam Electric Station, Unit 3 facility. This inspection was performed under the guidance of NRC Inspection Procedures 37550, "Engineering," and 37001, "10 CFR 50.59 Safety Evaluation Program." The inspection was performance based, with the team evaluating the quality of the engineering work products.

The team found the overall performance of the Waterford 3 engineering organizations to have improved. The licensee's implementation of the programs related to engineering activities was very good with the exception of updating design documents, as identified by the team's review of condition reports and the identification of the violation discussed below. This conclusion was based on the quality and detail found in the condition reports, temporary alterations, problem evaluation information requests, substitute part evaluation reports, and calculations reviewed. Engineering's responsiveness to problems and questions that were presented through the documents that were reviewed indicated very good support of operations and maintenance in the daily operations of the plant.

The team noted improved management attention in the assignment of work to manage work loads of the engineering staff. This was evidenced by a relatively small backlog of work of which little, if any, was safety-related. The team found that the engineering staff had an understanding of management expectations and placed appropriate emphasis on work activities. Additionally, the engineering staff exhibited rigor in implementing the design change process and in complying with engineering procedures, including the testing of heat exchangers in accordance with Generic Letter 89-13. One exception involved the installation of vent valves on containment spray piping, the engineering modification was incomplete in that operating procedures were not revised to control the operation of the vents. These were exceptions to an otherwise high level of engineering performance.

The team was informed of several programs that were recently implemented, or would be implemented in the near future. These included: a program to trend condition reports for repetitive issues that were not considered to be safety-significant; the use of probabilistic risk assessment (risk monitor) for daily plant activities; and improvement in the control over vendor supplied work products, such as the core reload analysis. While none of these programs were evaluated during this inspection, the team considered them to be indicative of management's desire to provide focus and direction to the engineering organization for the purpose of continuing improvement.

The licensee's performance of calculations was very good. One exception to this good performance involved a calculation that required a year for approval. In addition to the untimeliness of the approval of the calculation, the resultant setpoint change was implemented without updating the station information management system. This was a concern, not only because of the failure to update the information management system, but that it was not identified during the resolution of Condition Report CR-94-761.

The licensee's safety evaluations, design changes, substitute parts equivalency evaluations, and plant engineering information requests were found to have been accomplished in accordance with approved plant procedures and regulatory requirements. The team found that the licensee had made improvements to the performance of safety evaluations; however, there were some examples of safety evaluations that were not stand-alone documents. This made the evaluation of the documents difficult.

Based on the 102 condition reports reviewed, the team concluded that engineering's response was very good, in general. There were observations of areas for improvement. The failure to trend repetitive reports was considered to be the most significant area for improvement.

The licensee had established adequate calculation control and test procedures to meet their commitment for the once per fuel cycle testing of heat exchangers in response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." In addition to the licensee's commitments to the generic letter, periodic trending of operational data was occurring, but the program had not been formalized.

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DETAILS

This inspection was conducted pursuant to NRC Inspection Procedures 37550, "Engineering" 37001, "10 CFR 50.59 Safety Evaluation Program"; and 92903, "Followup - Engineering."

1 10 CFR 50.59 SAFETY EVALUATION PROGRAM (37001)

The team reviewed the 36 safety evaluations listed in Attachment 3 in accordance with the guidance of NRC Inspection Procedure 37001. All safety evaluations were found to be consistent with the requirements of 10 CFR 50.59, "Changes, tests and experiments." The team had no comments on the scope and results of the evaluations.

The observation was made by the team that the degree of detail in the evaluations had increased, compared to the previous inspection (NRC Inspection Report 50-382/93-14); however, certain evaluations could have been more detailed so that the evaluations would stand alone without reference to initiating documents, such as a design change package or nonconformance report. For example, in order to review the safety evaluation for Design Change DC 3033, the team had to refer to the design change package to fully understand the safety classifications. Also, for the safety evaluation prepared for Nonconforming Condition Identification 287462, the team had to refer to the condition identification report to understand the reason for the change being implemented.

2 ENGINEERING (37550)

2.1 Design Modifications

The team reviewed the five design changes listed in Attachment 4 and found that each package reviewed was well engineered and implemented in accordance with approved plant procedures, including appropriate post-modification testing.

2.2 Condition Reports

The team reviewed the 102 condition reports listed in Attachment 5. The reviewed condition reports were written against design, system, and maintenance engineering products. The team found that, in general, engineering's response to the condition reports was very good.

In general, the condition reports were of low safety significance, but several were repetitive. This repetition has occurred as the result of the condition reports being treated as isolated instances by the licensee: fixing the symptoms, not the causes. The licensee stated that this had been an observation of recent audits of the Waterford 3 site and that a program was being developed to address this concern. The program would review all condition reports, categorize the identified concern, and evaluate for repetitive problems.

The team observed that approximately 15 percent (15 of 102) of the sampled condition reports were granted at least one extension before corrective actions were completed and 14 percent (14 of 102) addressed problems between the plant configuration and a governing design document. Areas identified in the condition reports included abnormal setpoints (including surveillance procedure updates), design basis document errors, plant drawing differences, and incorrect information in the station information management system. The team found that 5 of the condition reports addressed plant configuration and design document differences related to the station information management system.

2.2.1 Condition Reports CR-93-044 and CR-93-102

The team reviewed two condition reports which identified problems with vendor supplied information. In Report CR-93-044, the licensee identified that a nonconservative analytical limit was applied to the applicable safety analysis for determining the high log power trip setpoint. This condition was present since the second operating cycle. Similarly, in Report CR-93-102, the licensee identified that nonconservative values were supplied in the plant data book curves for rod worth when determining shutdown margin. In each case, the licensee performed an analysis to verify safe operation within the safety analysis described in the plant's Final Safety Analysis Report. The team was concerned with the licensee's control and verification process of vendor supplied information which could effect the safe operation of the plant. The licensee had issued a revision to Design Engineering Procedure NOECP-702, "Processing and Approval of Groundrules [sic] and Reload Analysis," Revision 0-1, in response to the condition reports, which delineated the processing and approval of ground rules and the reload analysis report. The licensee required a thorough review with the respective vendor, including evaluations of the data used within the analyses, in accordance with Procedure NOECP-702.

2.2.2 Condition Report CR-94-364

The licensee initiated Condition Report CR-94-364 to investigate the failure of a pressurizer pressure transmitter. The licensee had returned the failed transmitter to the vendor for failure analysis. The vendor, Rosemount Nuclear Instruments, Incorporated, identified a glass and metal contamination in the delta cell halves of the Rosemount Model 1154 pressure transmitter, which was returned by the licensee.

The team concluded that the licensee properly addressed the failed transmitter, but was concerned with the identification of the glass to metal contamination. The vendor had reportedly corrected this problem prior to the development of the Model 1154 transmitters. The team contacted the vendor inspection section in the Office of Nuclear Reactor Regulation regarding this concern and forwarded the information with respect to this condition report.

2.2.3 Condition Report CR-94-439

The licensee initiated Condition Report CR-94-439 because a part number from a drawing had been entered incorrectly in the vendor engineering technical interface program. This resulted in a work delay since the appropriate spare parts had not been ordered in time.

The team noted that the root cause for the condition report was a restatement of the problem without determining an actual root cause. The team considered this to have been a weak root cause analysis.

2.2.4 Condition Report CR-94-072

The licensee initiated Condition Report CR-94-072 when both trains of the auxiliary component cooling water system were found to have been operated at a pressure greater than the system design and hydrostatic test pressures. The licensee identified this problem when the B train pump was started and the shell side heat exchanger thermal relief valve lifted.

The licensee determined that the pressure in both trains was approximately 96 psig (662 kPa). This pressure exceeded the system design pressure of 75 psig (517 kPa) and the hydrostatic test pressure of 94 psig (648 kPa). The team found that the licensee had performed a comprehensive root cause analysis. The licensee concluded that an inadequate review during the design process had been performed for the auxiliary component cooling water pressure. In addition, the licensee determined that adequate venting had not been provided for air entrapped in the system.

The team noted that the corrective actions included a rerating of the system piping, increasing the set pressure of the thermal relief valve, and flushing the system after maintenance, or draining, when air could be introduced. The licensee had gagged the thermal relief valve closed until the set pressure was increased. To compensate for the gagged relief valve, the licensee hung a caution tag on the heat exchanger shell side inlet and outlet valves to inform the operators that isolation of the heat exchanger could result in the overpressurization of the heat exchanger shell side.

The licensee performed an engineering evaluation to justify exceeding the design pressures of the line between the pump and the heat exchanger. The licensee determined that the allowable stress was not exceeded. Additionally, the components in the line were reviewed against the appropriate ANSI class rating. The licensee determined that the maximum allowable pressures of the components was greater than the overpressure condition. The licensee concluded that the overpressure condition was not a concern and that the integrity of the auxiliary component cooling water system remained assured.

The team concluded that the licensee had performed a very thorough and comprehensive root cause analysis and engineering evaluation.

2.2.5 Condition Report CR-94-206

The licensee initiated Condition Report CR-94-206 when the protective cover for the bellows on Penetration 32 was found attached to the bellows assembly on the non-bolted side. The protective cover was attached to the protective ring with tack-welded shipping tabs. The licensee found that this condition could have prevented bellows expansion when subjected to a thermal transient resulting from a design basis accident.

The licensee determined that the shipping tabs should have been removed after installation. The licensee identified the root cause to have been inadequate attention to detail during plant construction. The team noted that the licensee's corrective actions included an inspection of all accessible penetrations. The licensee identified those penetrations which had the tack-welded shipping tabs and removed the shipping tabs.

The licensee performed Calculation EC-P94-005, "Qualification of Containment Penetrations 32, 48, 56, 60, and 25," Revision 0, which the team reviewed. The team found that the licensee had evaluated the structural integrity of the affected penetrations. In the calculation, the licensee determined that the weakest link for failure was the shipping tabs. The licensee calculated the failure load by determining the total thermal growth. From this force, the force per shipping tab was calculated and compared with the failure force on the tack-welds. The licensee determined that the force to fail the welds was less than the force to fail any of the components associated with the penetration assembly.

The team concluded that the licensee's calculation was well performed and that the corrective actions were complete.

2.2.6 Condition Identifications CI 287461 and CI 287462

The team reviewed Condition Identification CI 287461, "Pressure Surge and Fluid Transient in the Containment Spray Train A System," Revision 1, which documented a pressure surge which occurred in the Containment Spray Header A in September 1993 during post-maintenance testing. The licensee attributed the high pressure to line overpressurization caused, in part, by air entrained within the piping and trapped between the closed isolation valve and an upstream check valve. The licensee measured pressures of 469 psig (3.23 MPa) in a portion of the system with a 300 psig (1.17 MPa) design pressure. The licensee noted that similar conditions may have occurred in the past, and that pressures as high as 570 psig (2.22 MPa) may have existed during those events. The licensee performed an engineering evaluation and concluded that there was no direct damage to the piping from this event.

The team found that Condition Identification CI 287462, "2CS10-7A Vent Line Addition," Revision 1, was initiated to install two vent lines and valves, in areas of the system where local high points existed, in accordance with Nonconforming Condition Identification Package 287462. The licensee initiated this action because it was believed that the air pockets were the root cause of the hydraulic transient evaluated in Condition Identification CI 287461.

Although the licensee provided instructions for the installation of the vent piping and valves, neither instructions nor procedure changes relating to when the areas should be vented were developed. For example, the periodic operation of the containment spray riser level pump to maintain spray riser level could result in the addition of aerated water to the system, and, as a result, air could find its way to the high point in the system.

Although Procedure OP-009001, "Containment Spray System," was updated to indicate the addition of the two new vent valves to the description, there were no instructions which described when the valves should be used to ensure that a similar event would not occur in the future. The licensee stated that consideration would be given to updating other procedures to provide for venting of the system. In addition, the resident inspectors had previously identified that there are no administrative controls for venting of flow transmitters (NRC Inspection Report 50-382/94-20, Section 6.5). The licensee's actions with regard to venting instructions is considered an inspection followup item (382/9426-02).

2.3 Temporary Alterations

The team reviewed all nine of the installed temporary alterations in accordance with the guidance of NRC Inspection Procedure 37550, "Engineering." The team found that the licensee had implemented the temporary alterations in accordance with Administrative Procedure UNT-005-004, "Temporary Alteration Control," Revision 10. The team also noted that the temporary alterations had been maintained and audited in accordance with Procedure UNT-005-004.

During the review of the temporary alterations, the team selected Temporary Alteration TA-94-012 (the only safety-related document) for an in-depth review. The team attempted to verify that the affected safety-related drawings had been posted and marked and found that there were no control room affected drawings related to this temporary alteration. The team noted that only selected safety-related drawings were maintained in the control room.

Since temporary alterations were only posted to control room drawings, the team observed that limited plant personnel would have access to correct and up-to-date drawings during an emergency situation when the technical support center, the emergency operations facility, and the operational support center were manned. The licensee stated that such a scenario had not been considered and that an evaluation would be performed to determine the necessity to expand the scope of the annotating drawings associated with temporary alterations.

The team did not identify any violation of procedures for the temporary alteration program.

2.4 Problem Evaluation Information Requests

The team reviewed the plant engineering information requests listed in Attachment 6 in accordance with the guidance of NRC Inspection Procedure 37550. The team found that the requests were for information exchanges only, as the licensee intended. The team concluded that the evaluations were well performed and in accordance with approved plant procedures.

2.5 Substitute Part Equivalency Evaluation Report

The team reviewed the substitute part equivalency evaluation reports listed in Attachment 7 in accordance with the guidance of NRC Inspection Procedure 37550. The team found that the evaluations were well performed and in accordance with approved plant procedures.

2.6 Calculations

The team reviewed the calculations listed in Attachment 8 in accordance with the guidance of NRC Inspection Procedure 37550.

2.6.1 Calculation EC-M94-005

The team reviewed Calculation EC-M94-005, "Determination of Performance of CCW Heat Exchanger 'A,' NRC Generic Letter 89-13," Revision 0. The team found that the licensee performed this calculation to determine the heat transfer capability of Component Cooling Water Heat Exchanger A at design conditions. The licensee obtained test data on March 6, 1994, during the sixth refueling outage, using Special Test Procedure 01120153. The licensee had developed the special test procedure to meet their commitment for a once per fuel cycle test in response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

The component cooling water heat exchangers were part of the component cooling water system which transferred heat from several primary system components, shutdown cooling components, and the diesel generators. The component cooling water system transferred the heat to either the dry cooling towers or to the auxiliary component cooling water system through the component cooling water heat exchangers. The dry cooling towers had the capability to remove all of the heat during normal operation in the winter months. During the summer months, and for design heat loads when additional cooling capability would be needed, the component cooling water heat exchangers would be utilized to transfer a portion of the heat load to the alternate component cooling water system, which then would transfer that heat to the wet cooling towers and ultimately to the atmosphere. The two component cooling water heat exchangers were single pass, once through, shell and tube heat exchangers with stainless steel tubes and a carbon steel shell. Alternate component cooling water, which had the potential for fouling and corrosion, passed through the shell side of the component cooling water heat exchangers.

The team noted that the licensee utilized a computer program used throughout the industry to model a component cooling water heat exchanger using the temperature and flow data taken during the test. Special temperature instrumentation was constructed for the test because the existing resistance temperature detector and flow element instrumentation were estimated to have an accuracy only to within several degrees. The data and model were used to solve the fouling which existed at the time of the test. The licensee then used the results, along with design basis heat loads and maximum expected ultimate heat sink conditions (83°F (28°C) wet bulb temperature), to develop the component cooling water discharge temperature that could have occurred during a design basis event. The licensee included an assessment of instrument error and uncertainty.

The licensee determined that the resulting temperature was 117.18°F (47.3°C), which was higher than the 115°F (46.1°C) temperature used in the final safety analysis report. The resultant temperature indicated that the heat exchanger was degraded as a result of fouling. The licensee initiated Condition Report CR-94-174 and Licensee Event Report 94-004 to document this condition. Licensee inspections and chemical analyses, subsequently performed, indicated the presence of microbiological activity, algae, and slime. The licensee then cleaned the heat exchangers and removed several hundred pounds of material. Post-test boroscopic inspections that were performed indicated that substantial clearing had occurred and noted the residues that remained. On the basis of the removed material and the visual inspection, the licensee concluded that the thermal performance of the heat exchangers had been restored to design conditions. The heat exchangers were not retested to verify heat transfer capability. The licensee stated that they have not retested with the more accurate test equipment because of insufficient heat in the component cooling system to get meaningful results.

The licensee has taken trend data using the in-place instrumentation. The team found that the results of the trend data indicated improved performance over that found during the test; however, there was no procedure governing this test to provide such things as test methodology or quantitative acceptance criteria. The team also noted that the recent trend testing in August 1994 indicated temperature differences of 6°F to 8°F (3.3°C to 4.4°C) across the heat exchangers. Such temperature differences could provide meaningful results for the accurate instrument package according to industry guidelines.

At the time of the inspection, the plant had no plans for further testing other than trending with the limited accuracy in-place instrumentation, until the next scheduled special performance test during the seventh refueling outage, in the fall of 1995. The inspectors were concerned that, since the alternate component cooling water system was generally stagnant during the winter months, there was a potential for fouling during that time which could affect performance during the heavier load season of the summer. The Director, Design Engineering, indicated that the utility would consider testing prior to the next refueling outage if an outage of sufficient deviation occurred.

2.6.2 Calculation EC-M92-049

The team reviewed Calculation EC-M92-049, "Determination of Performance of CCW Heat Exchangers, NRC Generic Letter 89-13," Revision 0. The team found that the licensee performed this calculation to demonstrate the heat transfer capabilities of the component cooling water heat exchangers during the fifth refueling outage to determine whether they met design and accident requirements. An earlier version of the computer program, discussed above, was used. The licensee obtained the test data with the in-place instrumentation (see above), with no consideration made for instrument error. In addition, the team could not determine if conservative assumptions were used in all cases. However, the licensee did recognize the shortcomings of the methodology, and revised the test methodology for the testing performed during the sixth refueling outage, and utilized the new, more accurate instrumentation, as discussed above for Calculation EC-M94-005.

2.6.3 Calculation EC-M92-050

The team reviewed Calculation EC-M92-050, "Charging Pump NPSH for Gravity Feed from the BAMTs," Revision 0, and found that the licensee had performed this calculation to assess the adequacy of available net positive suction head for the charging pumps when taking suction from the boric acid makeup tanks in the gravity feed mode. The team noted that a loss of net positive suction head could occur upon loss of the flow path from the boric acid transfer pumps and the simultaneous running of all three charging pumps to mitigate a design basis accident. The team recognized that such a configuration was not very probable; however, the consequences of a loss of net positive suction head could be great. The boric acid transfer pumps normally transfer 3.5 to 4 percent boric acid solution from the boric acid makeup tanks to the suction of the three charging pumps.

The licensee concluded that, with three charging pumps in the gravity feed mode, there was 6.5 ft (1.98 m) of net positive suction head available with the configuration and operating parameters then being maintained, such as a boric acid makeup tank temperature of 170°F (76.7°C). This compared with a required net positive suction head of 12.0 ft (3.66 m) with three charging pumps operating at rated flow of 44 gpm (165.4 Lpm). The licensee also found that the temperature being maintained in the boric acid makeup tank would have to be lowered to 149°F (65°C) to maintain adequate net positive suction head for this case with three charging pumps running. The licensee originated and reviewed the calculation March 1993, but did not approve the calculation until March 1994, after testing and evaluation was completed in January 1994. The period of one year to approve the calculation was not viewed as timely.

The licensee initiated Setpoint Change 93-011 to conservatively address the results of Calculation EC-M92-050, but did not implement the change until the calculation was approved. During a review of Setpoint Change 93-011, the team observed that the boric acid makeup tank temperature controllers and alarm setpoints had been lowered on July 23, 1994, in accordance with Work Authorization 01125613 and Condition Identification CI 291307. Setpoint Change 93-011 called for boric acid makeup tank heater operation and alarm setpoints to be changed as shown below:

	INITIAL VALUE	FINAL VALUE
Hi Alarm	180°F (82.2°C)	120°F (48.9°C)
Heater Off	167°F (75.0°C)	110°F (43.3°C)
Heater On	163°F (72.8°C)	100°F (37.8°C)
Low Alarm	155°F (68.3°C)	90°F (32.2°C)

In an attempt to verify the updated setpoint as documented in the station information management system, the team observed that none of the temperature controller setpoints had been updated with the final values as of December 1, 1994, and only two (BAMITIC0207 AND BAMITIC0209) of the four temperature controllers had been identified or posted as having an outstanding work authorization in effect. Procedure UNT-007-014, "Administrative Procedure Setpoint Change Control," Revision 6, stated, in part, that "[t]he responsible engineer shall . . . initiate . . . SIMS data base updates." Contrary to Procedure UNT-007-014, the information management system data base was not updated. The team verified that the operator logs were still using an acceptance range of greater than 55°F to 175°F (12.8°C to 79.4°C). A portion of this range was greater than the value determined in the engineering calculation for adequate net positive suction head, as previously stated.

Procedure MD-001-032, "Administrative Procedure Work Authorization Closing," Revision 0, stated, in part, that "[i]f the WA implemented a setpoint change . . . then a copy of the WA shall be routed to System Engineering." When questioned by the team as to the closure of Work Authorization 01125613, the licensee stated that the documentation could not be found and must, therefore, be "lost".

The failure to update the station information management system in accordance with Procedure UNT-007-014 was identified as a violation (382/9426-01). This was of concern because of the failure of the licensee to identify this issue in response to Condition Report CR-94-761, discussed in Section 2.2, above. The station information management system was the licensee's setpoint document and, therefore, a design document. This example, along with the examples identified by the licensee, indicated weakness in the procedures for ensuring that all design documents would be updated in a timely manner after completion of field work.

In addressing the potential loss of net positive suction head to the charging pumps under the conditions postulated above, the licensee issued Setpoint Change Notice 93-011. The team concluded that, while a potential loss of net positive suction head existed, the charging pumps would have been able to provide sufficient flow to mitigate a design basis accident based on the vendor's pump curve with reduced suction head.

2.6.4 Calculation EC-P92-053

The team reviewed Calculation EC-P92-53, "Evaluation of Pipe Wall Thinning Due to Erosion/Corrosion, Evaluation of Component No. 111-007," Revisions 0 and 1, and found that the licensee performed this calculation to determine the acceptability of safety-related Piping Component 111-007, which had been identified to be affected by erosion/corrosion through Problem

Evaluation/Information Request DE-31. The licensee had reported the results of an ultrasonic test performed during the fifth refueling outage, which measured the wall thickness to be 1.234 in (3.135 cm). This was less than the minimum acceptable thickness, which had been established for the pipe of 1.285 in (3.264 cm), following the rules of the licensee's flow accelerated corrosion program. The licensee assessed the remaining wall using ASME Code Case N-480, regarding local thinning acceptance criteria for continued service, and determined that it was acceptable as-is on the basis of the evaluation of the code case until the sixth refueling outage. The licensee reached this conclusion even though the code case was not listed in Regulatory Guide 1.147, and authorization for the use of the code case had not been requested as required by 10 CFR 50.55a(a)(3) for use of code cases not listed in the regulatory guides.

During the sixth refueling outage, the licensee documented the discovery of the error in Condition Report CR-94-162. The corrective action for the condition report was to route a copy of it to all design engineering personnel in the mechanical and civil areas, and to all remaining department heads in the engineering organization. Additionally, the licensee issued Revision 1 of the calculation to void Revision 0 and to void the need to rely on the code case. The team was concerned that, although this would remind all existing supervisors and current mechanical and civil personnel of the requirements for notification of the use of code cases, it would not notify or train future personnel. The team found that the licensee's training program did have courses for new and existing personnel in such areas as codes and standards overview and licensing requirements; however, neither of those courses discussed the use and rules for requesting approval for code cases. The team concluded that the licensee actions were short term. Licensee training personnel contacted during the inspection stated that they would evaluate including this kind of information in the training program in the future.

2.6.5 Operational Experience Engineering Evaluation

The team reviewed an operational experience engineering evaluation which considered the potential overpressurization of the main steam system. The evaluation was a draft copy addressing NRC Information Notice 94-60, "Potential Overpressurization of Main Steam System." The team concluded that the licensee's evaluation was conservative and would lead to reactor trip setpoints being reduced to a value less than specified in the Technical Specifications should a number of appropriate main steam safety valves be rendered inoperable. Combustion Engineering, the nuclear steam system supplier for the Waterford 3 plant, had not provided any detailed information to justify the present setpoint reduction values. However, a presentation to representatives of various Combustion Engineering plants was planned in the near future to discuss assumptions, applicability, and modifications of the "new" equation identified in NRC Information Notice 94-60, to justify use in a Combustion Engineering facility. The team noted that there were no plans to submit a Technical Specification change until further information was available, and analysis was performed.

2.7 Engineering Initiatives

During the inspection, the licensee made a presentation to members of the team related to the engineering organization at Waterford 3. Attachment 8 contains the handout provided by the licensee for the presentation. At the presentation, the licensee discussed the self-assessments that had been performed for the engineering organization, areas which they considered strengths in engineering, engineering issues, and new initiatives.

The team was informed that the licensee used probabilistic risk assessment for determining which safety system functional inspection would be performed. The team noted that the latest audit of engineering that was presented to the team was from December 1993; therefore, the licensee did not have any recent reports to provide the team for review. The team found that the licensee planned to perform self-assessments of the flow accelerated corrosion program, the vendor engineering technical interface program, common failures, and the software control program.

The team noted that the licensee considered its contractor reduction program, peer groups, and design engineering's safety role in daily plant activities as strengths. The licensee was active in 25 peer groups among the Entergy Operations, Inc., sites.

The licensee also presented information related to advanced training of the engineering staff. Engineers were being provided senior reactor operator training, systems training, root cause analysis training, and training on such topics as water hammer and pumps. The engineers were required to obtain checkout from senior reactor operators, as if the engineers were obtaining an operating license. The team considered this training to be capable of producing better informed engineers who had better understanding of integrated plant operations.

The licensee also considered its steam generator integrity program to be a strength. Other activities the licensee considered as strengths were the thermal performance improvement team, the secondary leak reduction program, the charging pump seal reliability program, the reliability improvement team, and the setpoint program.

The licensee considered several areas to be of interest for engineering. These included the design processes to prevent common mode failure, component and system failure root cause analysis, the relief valve program, the seismic qualification upgrade program, electrolytic capacitor replacement, and reload safety analysis ground rules improvement.

New initiatives include the probabilistic safety assessment and development of the risk meter, replacement of the plant computer, development of an improved air-operated valve actuator program, and implementation of the engineering study performed for the four Entergy sites.

The team noted that one of the areas discussed that needed improvement was the tracking and trending of condition reports for generic concerns. At present, the team found that the licensee, in effect, did not look for repetitive failures. The team found this to be the case because the licensee would research previous failures of a specific component, not previous failures of the same make or model in other applications in the plant. The licensee stated that an audit group was being established to review all condition reports for generic implications, as well as to ensure that failures were investigated for similar components in other applications. Evaluation of the effectiveness of the licensee actions is considered an inspection followup item (382/9426-03) and will be performed during future inspections.

2.8 Probabilistic Risk Assessment

The licensee has been aggressive in the use of probabilistic risk assessment information. Probabilistic risk assessment information was used with success for outage management during the sixth refuelling outage. This involved the use of the outage risk assessment management program. The team found that, by the end of the year, the licensee planned to have a risk meter, or monitor, implemented. The risk monitor will be used, initially, for online maintenance planning and scheduling. Later, operations will use the risk monitor for scheduling of surveillance and other activities. The team noted that the licensee had developed lists of risk significant systems, components, and activities. This information had been used for informal reviews of design modifications by engineering. These lists also have been used for activities such as auditing.

2.9 Pressurizer Relief Valves

The team reviewed the licensee's pressure relief valve program. The team noted that the main steam safety valves had not been able to meet the seat leakage criteria using saturated steam after being pressure tested for setpoint. The licensee had documented this problem in Condition Report CR-94-197 and Nonconforming Condition Identification NCI-289966. The licensee concluded that the root cause of the problem was an inadequate design specification which did not require the manufacturer of the main steam safety valves to meet a zero leakage criteria on saturated steam at 90 percent of set pressure. The licensee accepted the main steam safety valves as they were. The team found that the licensee had determined that the amount of leakage from the valves would not affect the set pressure. In addition, the licensee determined that the off-site dose rate was not significantly affected by the seat leakage in the event of a steam generator tube rupture.

The team found that the licensee had revised the design specification to state that each main steam safety valve could have no more than 65 lbs/hr (29.5 kg/hr) leakage at 900 psig (6.2 MPa). Additionally, the licensee initiated a revision of the final safety analysis report for the revised allowable leakage and the corresponding off-site dose. The licensee also installed acoustic monitors on the valves and had taken base-line data so that future evaluation of seat leakage could be monitored.

The inspectors concluded that the pressure relief valve program at the Waterford 3 site was functioning well.

2.10 Conclusions

The team found the overall performance of the Waterford 3 engineering organizations to have been very improved. The licensee's implementation of the programs related to engineering activities were very good with the exception of updating design documents.

The licensee's safety evaluations, design changes, substitute parts equivalency evaluations, and plant engineering information requests were found to have been accomplished in accordance with approved plant procedures and regulatory requirements. The team did identify an area related to safety evaluations for which the licensee had made improvements; however, there were examples of a lack of completeness of the safety evaluations to be considered stand-alone documents.

The team's review of calculations led to the identification of the violation for failure to update the information management system. Also during review of calculations, the team identified a concern with the timeliness of approving calculations.

The team found that although vent valves had been added to the containment spray trains as a result of the pressure surge which occurred in September 1993, plant procedures or instructions did not establish when venting should be performed.

The team considered the licensee's evaluation of NRC Information Notice 94-60, "Potential Overpressurization of Main Steam System," to have been well performed and conservative. The licensee, however, was awaiting the analysis of the nuclear steam system supplier prior to taking any actions.

3 FOLLOWUP - ENGINEERING (92903)

(Closed) Licensee Event Report 93-009: Pressurizer Code Safety Valve Setpoints Out of Tolerance

In November 1993, the licensee found that both pressurizer code safety valves had as-found setpoints outside the allowable tolerance of ± 1 percent. These valves had been installed during the fifth refuelling outage for use during the sixth operating cycle. The test results indicated that Valve BS-08031 was 3.5 percent above the required set pressure and Valve BS-08030 was 3.25 percent above the required set pressure.

The licensee concluded that one of the causes for the setpoints to be outside of allowable tolerance was the use of the "jack and lap" procedure following set pressure testing. The licensee's corrective actions included revising the safety valve test procedure to require steam set pressure verification if the "jack and lap" procedure was used. In addition, the licensee was in the process of preparing a change to Technical Specification 3.4.2.2 which would change the setpoint tolerance to ± 3 percent.

The team found that the licensee had performed an evaluation to review the effect of the setpoints being found out-of-tolerance on the safety analysis. The licensee concluded that, for the most limiting event, the reactor coolant pressure would have been below the safety limit of 2750 psia (18.96 MPa) for the peak pressure.

The team concluded that the licensee had adequately addressed the out-of-tolerance setpoints for the pressurizer code safety valves.

ATTACHMENT 1

PERSONS CONTACTED AND EXIT MEETING

1 PERSONNEL CONTACTED

1.1 Entergy Personnel

R. Azzarello, Director, Design Engineering
A. Cilluffa, Supervisor, Maintenance Engineering
R. Finch, Senior Staff Engineer, Design Engineering - Specialties
J. Hologa, Manager, Mechanical and Civil Engineering
J. Houghtaling, Manager, Technical Services
J. Howard, Manager, Procurement and Programs Engineering
P. Melancon, Manager, Safety and Engineering Analysis
W. Pendergrass, Shift Supervisor
P. Prasankumar, Manager, Design Electrical and Instrumentation and Controls
D. Vinci, Manager, Licensing

1.2 NRC Personnel

T. Pruett, Resident Inspector, Waterford 3
T. Westerman, Chief, Engineering Branch

The personnel listed above attended the exit meeting. In addition to the personnel listed above, the team contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on December 2, 1994. During this meeting, the team reviewed the scope and findings of the report. The licensee did not express a position on the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the team.

ATTACHMENT 2

Inspection Finding Index

Violation 382/9426-01 was opened (Section 2.6.3).

Inspection Followup Item 382/9426-02 was opened (Section 2.2.6).

Inspection Followup Item 382/9426-03 was opened (Section 2.7).

Licensee Event Report 93-009 was closed (Section 3).

ATTACHMENT 3

SAFETY EVALUATIONS REVIEWED

Design Change Package-3033, "Install Mass Flow Probes on Eleven Process and Iodine Gas Monitors." November 8, 1991

Design Change Package-3195, "Safety Injection Sump Outlet Valves SI-602A and B Valve Operators (Revision 1)." May 14, 1991

Design Change Package-3265, "Emergency Diesel Generator Engine Control Cabinets Ventilation (Revision 0)." May 7, 1992

Design Change Package-3271, "Reactor Cavity Cooling Fans Inlet Damper Deletion." August 30, 1991

Design Change Package-3296, "Phase I, Revision 0: Main Steam Isolation Valve Replacement/Enhancement Stem Improvement." October 9, 1990

Design Change Package-3315, "Heated Junction Thermocouple Cable Supports (Revision 0)." May 8, 1992

Design Change Package-3329, "A Hoist Addition to Reactor Containment Building and Q-Deck Area (Revision 8)." January 30, 1991

Design Change Package-3348, "Emergency Feedwater Condensate Storage Pool Level and Steam Generator Wide Range Level Recorders. (Revision 0)." February 22, 1992

Design Change Package-3363, "Hydraulic Power Unit for Fuel Handling System (Revision 0)." July 16, 1992

Design Change Package-3372, "Alternate 120 VAC Power for Process Analog Control Cabinets, 29, 30, 31 and 62 (Revision 0)." April 19, 1992

Design Change Package-3377, "SI-407A/B Valve Motor Changeout (Revision 0)." August 13, 1992

Design Change Package-3408, "Replacement of Agastat 7000 Series Time Delay Relays for Emergency Diesel Generator 'A' and 'B' Sequencer Circuits. (Revision 0)." February 22, 1994

Design Change Package-3409, "Containment Spray Operation (Revision 2)." March 8, 1994

Temporary Alteration Request 92-018, "Replacing Core Protection Calculator Channel D RCP-B Speed Sensor with Core Operating Limit Supervisor System RCP-2B Speed Sensor." July 7, 1992

Temporary Alteration Request 92-043, "Qualified Safety Parameter Display System Channel 2 Heated Junction Thermocouple 4 Disconnection." November 6, 1992

Temporary Alteration Request 92-045. "Isolating the Load Transfer Control Element Drive Mechanism Coil of CEA 38." November 23, 1992

Temporary Alteration Request 92-046. "Cut and Cap SI-211 Drain Line to Reduce Safety Injection Tank Leakage." December 8, 1992

Temporary Alteration Request 93-002. "Feedwater Regulation Valve FW-173A Solenoid Valve Repair." February 12, 1993

Temporary Alteration Request 93-004. "Replace RTD (RCITE0122HC) for Core Protection Calculator "C" with RCITE0121X." April 1, 1993

Nonconforming Condition Identification 270579. "Repositioning of Valve FS-307." March 26, 1993

Nonconforming Condition Identification 281397. "Core Operating Limit Supervisor System Rework for Cycle 6." October 22, 1992

Nonconforming Condition Identification 287461. "Pressure Surge and Fluid Transient in the Containment Spray Train A System. (Revision 1)." September 29, 1993

Nonconforming Condition Identification 287462. "2CS10-7A Vent Line Addition." September 26, 1993

Nonconforming Condition Identification 287492. "Addition of Safety Valve to CS-125A Actuator (Revision 0)." September 29, 1993

Nonconforming Condition Identification 289755. "Gear Change on Motor Operated Valves SI-120 A and B and SI-121 A and B and Licensing Document Change Request No. 94-205." March 9, 1994

Special Test Procedure No. 01102098. Test of SI-207A(B)(AB), Revision 0. May 25, 1993

Special Test Procedure No. 01113917. "Test of B Containment Spray Header." Revision 0, Change No. 1. October 5, 1993

Special Test Procedure No. 01117875. "Component Cooling Water Discharge Check Valve Test (Revision 0)." January 12, 1994

Special Test Procedure No. 01117875. "Change 2 of Component Cooling Water Pump Discharge Check Valve Test (Revision 0)." January 18, 1994

Licensing Document Change Request No. 93-0091. "Revise Sections, Tables and Figures in the Final Safety Analysis Report with the Revised Loading of the Emergency Diesel Generators and Sizing of the Fuel Oil Storage Tanks." November 23, 1993

Licensing Document Change Request No. 93-0094. "Revision to Final Safety Analysis Report Section 8.3.1.2.4.(c)." January 18, 1994

Licensing Document Change Request No. 93-0164. "Change the Definition of a Harsh Environment and Update Final Safety Analysis Report Section 3.11."
July 13, 1993

Procedure Change No. 3. "OP-002-005. Chemical and Volume Control (Revision 10)." May 26, 1993

Procedure Change No. B. "OP-002-005. Chemical and Volume Control (Revision 10)." July 26, 1993

Work Authorization 01117875. "Component Cooling Water Discharge Check Valve Test (Revision 1)." February 7, 1994

"Cycle 7 Core Reload." February 17, 1994

ATTACHMENT 4

DESIGN CHANGES REVIEWED

Design Change DC 3388	Revision 0	Corrosion Rate Monitoring for Emergency Diesel Generator Jacket Water, Essential Chilled Water, Supplemental Chilled Water, and Supplemental Chillers Condensing Water Systems
Design Change DC 3402	Revision 0	Sizing of the Class 1E 3AB-S Battery
Design Change DC 3405	Revision 1	Motor-Operated Valve Modification
Design Change DC 3408	Revision 0	Replacement of Agastat 7000 Series Time Delay Relays for Emergency Diesel Generator 'A' and 'B' Sequencer Circuits
Design Change DC 3409	Revision 2	Containment Spray Operation

ATTACHMENT 5

CONDITION REPORTS REVIEWED

CR-93-027	CR-93-294	CR-94-281	CR-94-618
CR-93-031	CR-93-307	CR-94-293	CR-94-621
CR-93-035	CR-93-311	CR-94-331	CR-94-634
CR-93-044	CR-93-315	CR-94-343	CR-94-649
CR-93-053	CR-94-004	CR-94-347	CR-94-661
CR-93-061	CR-94-005	CR-94-353	CR-94-678
CR-93-071	CR-94-007	CR-94-361	CR-94-751
CR-93-074	CR-94-025	CR-94-364	CR-94-772
CR-93-079	CR-94-029	CR-94-377	CR-94-783
CR-93-087	CR-94-031	CR-94-379	CR-94-788
CR-93-096	CR-94-036	CR-94-380	CR-94-820
CR-93-102	CR-94-044	CR-94-390	CR-94-844
CR-93-112	CR-94-060	CR-94-395	CR-94-852
CR-93-117	CR-94-076	CR-94-396	CR-94-857
CR-93-156	CR-94-092	CR-94-406	CR-94-861
CR-93-181	CR-94-107	CR-94-434	CR-94-864
CR-93-189	CR-94-119	CR-94-435	CR-94-908
CR-93-198	CR-94-147	CR-94-439	CR-94-909
CR-93-206	CR-94-191	CR-94-471	CR-94-920
CR-93-208	CR-94-196	CR-94-474	CR-94-927
CR-93-209	CR-94-232	CR-94-507	CR-94-933
CR-93-214	CR-94-249	CR-94-548	CR-94-960
CR-93-257	CR-94-254	CR-94-550	CR-94-968
CR-93-262	CR-94-272	CR-94-566	CR-94-973
CR-93-266	CR-94-273	CR-94-575	CR-94-991
CR-93-284	CR- -276		

ATTACHMENT 6

PROBLEM EVALUATION INFORMATION REQUESTS REVIEWED

DE-055
NS-001
OM-040
OM-024
OM-030
TS-005
TS-010
TS-013
TS-031
TS-032

ATTACHMENT 7

SUBSTITUTE PART EQUIVALENCY EVALUATION REPORTS REVIEWED

9301118
9301176
9301190
9401212
9401213
9401218
9401231
9401240
9401244

ATTACHMENT 8

CALCULATIONS REVIEWED

EC-M92-049
EC-M92-050
EC-M94-005
EC-P92-017
EC-P92-053
EC-P94-005
EC-P94-006

ATTACHMENT 9

WATERFORD 3 ENGINEERING PRESENTATION

WATERFORD 3 ENGINEERING PRESENTATION

AGENDA

SELF ASSESSMENTS

DBD VERIFICATION PROGRAM
SSFI'S
OTHER SELF ASSESSMENTS

R.G. AZZARELLO
R.G. AZZARELLO
R.G. AZZARELLO

STRENGTHS IN ENGINEERING

CONTRACTOR REDUCTION
PEER GROUPS
DE'S SAFETY ROLE IN DAILY PLANT ACTIVITIES
TRAINING AND DEVELOPMENT
MOV PROGRAM
FLOW ACCELERATED CORROSION
STEAM GENERATOR INTEGRITY COMMITTEE
PLANT PERFORMANCE AND RELIABILITY IMPROVEMENT
KEY SYSTEM CONCEPT
PERFORMANCE IMPROVEMENT TEAM
SECONDARY STEAM LEAK REDUCTION
CHARGING PUMP SEAL RELIABILITY
RELIABILITY IMPROVEMENT TEAM
SETPPOINT PROGRAM

R.G. AZZARELLO
R.G. AZZARELLO
R.G. AZZARELLO
J.D. HOLOGA
J.D. HOLOGA
J.E. HOWARD
B.N. PROCTOR
A.M. CILLUFFA

P.V. PRASANKUMAR

ENGINEERING ISSUES

DESIGN PROCESSES TO PREVENT COMMON MODE FAILURE
ROOT CAUSE ANALYSES
RELIEF VALVE PROGRAM
SEISMIC QUALIFICATION UPGRADE PROGRAM
VETIP PROGRAM
ELECTROLYTIC CAPACITOR REPLACEMENT
RELOAD SAFETY ANALYSIS

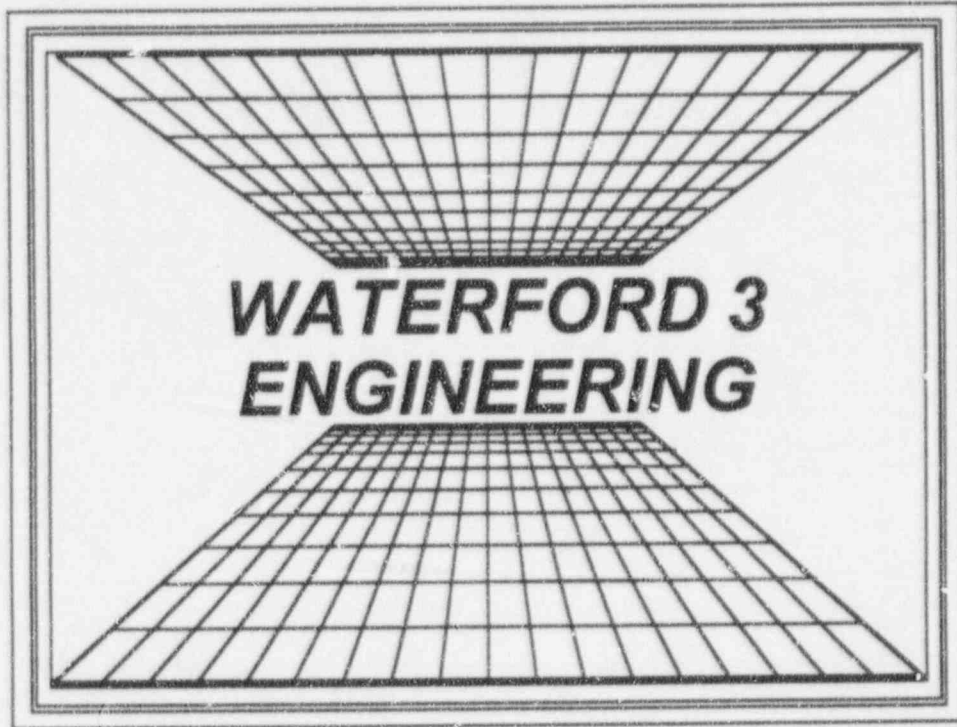
P.V. PRASANKUMAR
B.N. PROCTOR
J.D. HOLOGA
J.D. HOLOGA
J.E. HOWARD
A.M. CILLUFFA
P.M. MELANCON

NEW INITIATIVES

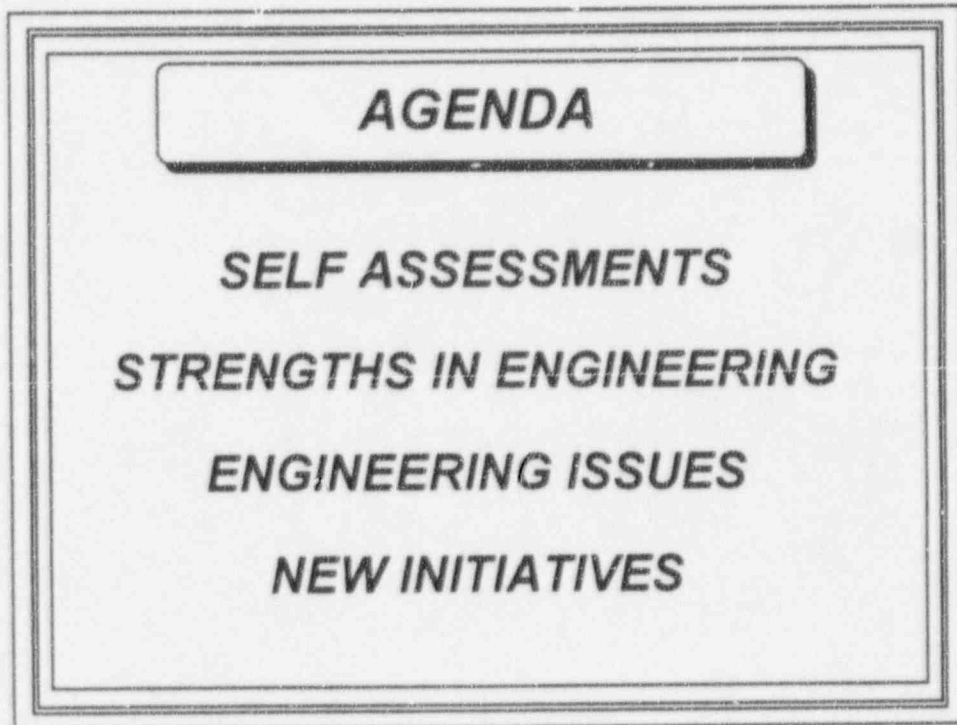
SAFETY ANALYSIS ENHANCEMENTS
SAFETY ANALYSIS DBD'S
PROBABALISTIC SAFETY ASSESSMENT
PMC REPLACEMENT
AOV ACTUATORS
ENGINEERING STUDY

P.M. MELANCON

P.M. MELANCON
A.M. CILLUFFA
R.G. AZZARELLO

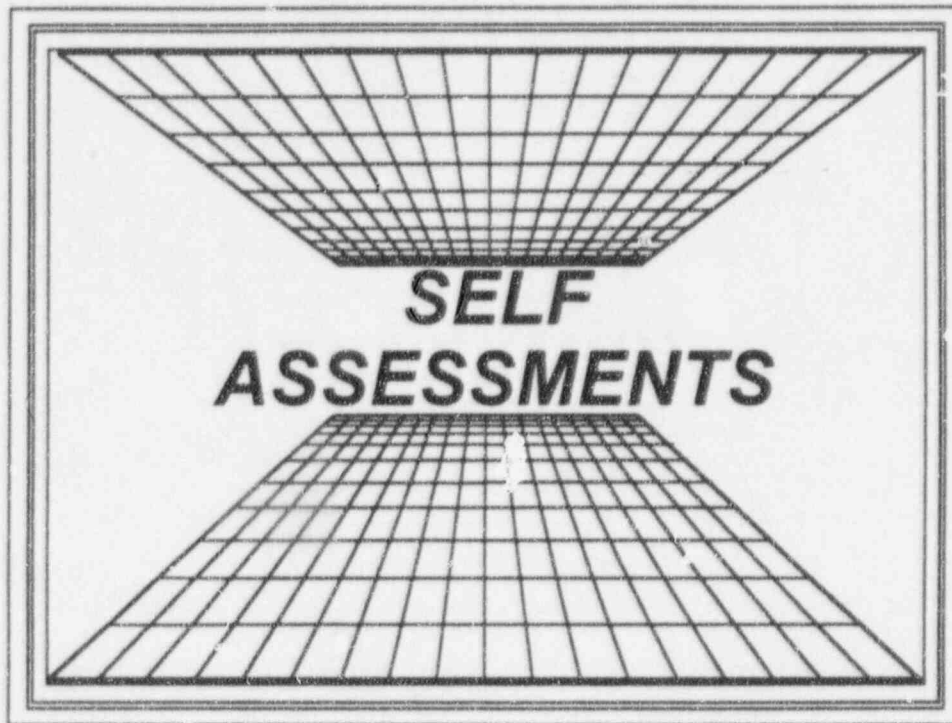


**WATERFORD 3
ENGINEERING**



AGENDA

- SELF ASSESSMENTS**
- STRENGTHS IN ENGINEERING**
- ENGINEERING ISSUES**
- NEW INITIATIVES**



**DBD VERIFICATION
PROGRAM**

SSFI'S BASED ON PRA

MOV PROGRAM

SETPOINT PROGRAM

ELECTRICAL CALCULATION PROGRAM

RELIEF VALVE PROGRAM

SSFI'S

COMPLETED

1984 - DESIGN VERIFICATION OF EFW

1986 - RAB HVAC

**1989 - COMPONENT COOLING WATER
AND AUXILIARY COMPONENT
COOLING WATER**

1990 - EMERGENCY DIESEL GENERATOR

SSFI'S

COMPLETED/PLANNED

1991 - NRC EDSFI

1992 - CHEMICAL & VOLUME CONTROL

1993 - SAFETY INJECTION

**1995 - EMERGENCY FEEDWATER
SYSTEM**

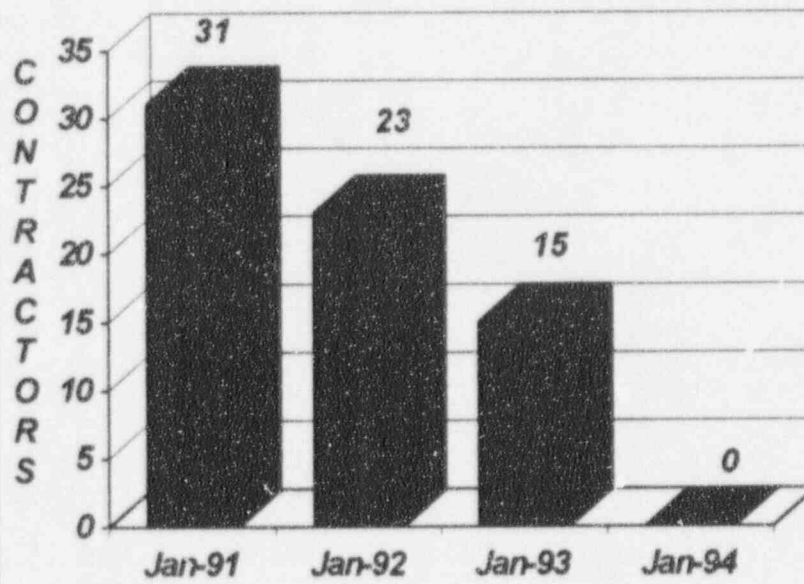
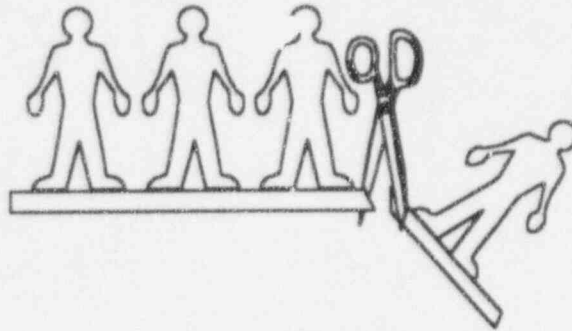
OTHER SELF ASSESSMENTS

- ★ **FLOW ACCELERATED CORROSION**
- ★ **VETIP**
- ★ **COMMON MODE FAILURE**
- ★ **SOFTWARE CONTROL PROGRAM**

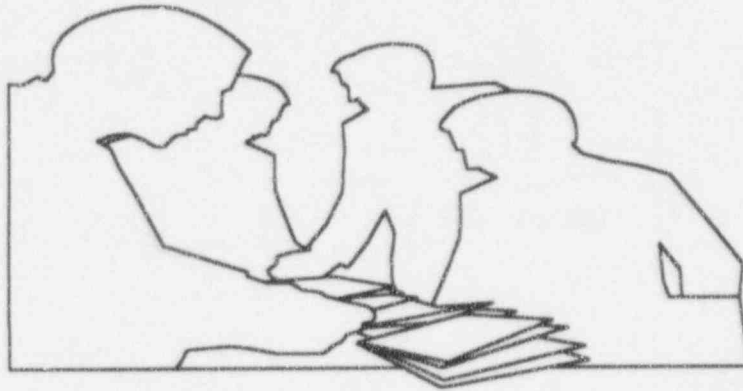


**STRENGTHS IN
ENGINEERING**

CONTRACTOR REDUCTION PROGRAM



PEER GROUPS



PEER GROUPS



**CONFIGURATION MANAGEMENT
ENVIRONMENTAL QUALIFICATION
SPECIAL PROCESS (WELDING, ETC.)**

**STEAM GENERATOR
TRAINING**

COMPUTER APPLICATIONS

BUSINESS PRACTICE

ELECTRICAL DESIGN

PEER GROUPS

(CONT'D)

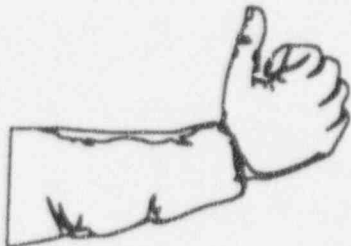
**SEISMIC QUALIFICATION/STRUCTURAL
FLOW ACCELERATED CORROSION
MOTOR AND AIR OPERATED VALVES
SECTION XI PROGRAMS
SECURITY
MECHANICAL DESIGN
PROCUREMENT ENGINEERING
PIPING STRESS AND SUPPORT**



PEER GROUPS

(CONT'D)

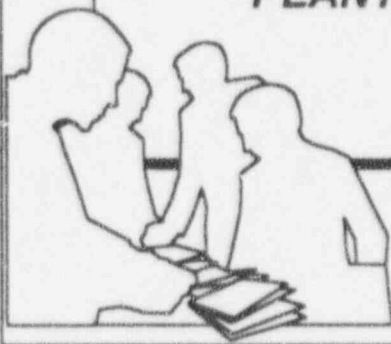
**CHECK VALVE
SYSTEM ENGINEERING
REACTOR ENGINEERING
DESIGN PROCESS
CAD
PRA/PE
FIRE PROTECTION
SAFETY ANALYSIS
I&C**



**DESIGN ENGINEERING
SAFETY ROLE IN DAILY
PLANT ACTIVITIES**



**RELIABILITY IMPROVEMENT TEAM
CONDITION REVIEW BOARD
SAFETY REVIEW COMMITTEE
PLANT OPERATING REVIEW
COMMITTEE
PLAN OF THE DAY**



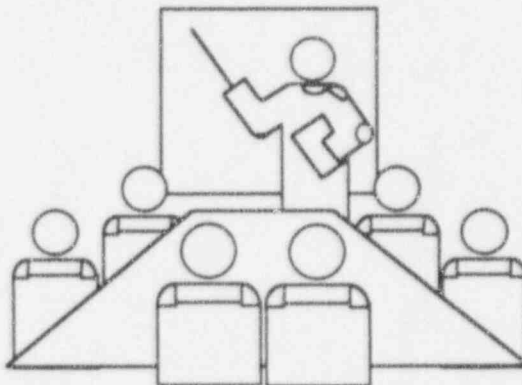
**DESIGN ENGINEERING SAFETY
ROLE IN DAILY PLANT ACTIVITIES
(CONT'D)**

**SYSTEM ENGINEERING
MEETINGS**

JOINT TEST GROUP

**COST BENEFIT LICENSING
ACTIVITY**

**TRAINING AND
DEVELOPMENT**



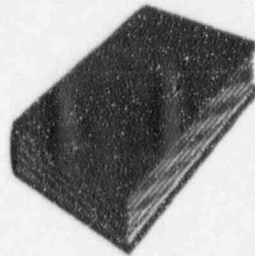
B A S I C S

REQUIRED READING

ORIENTATION TRAINING

QUALIFICATION CARDS

RECURRING TRAINING



ADVANCED TRAINING

✓ SRO/RO TRAINING

✓ SYSTEMS TRAINING

✓ WATERHAMMER

✓ PUMP

✓ ROOT CAUSE



OTHER TRAINING

- ✓ **REACTOR ENGINEERS
SIMULATOR TRAINING WITH OPS**
- ✓ **SYSTEM ENGINEERING
CHECKOUTS WITH SRO'S**
- ✓ **SYSTEM ENGINEER ONE-ON-ONE
WITH RESIDENT OPERATIONS
EXPERT**

DEVELOPMENT OF ENGINEERING PERSONNEL

- **REFUELING ACTIVITIES (e.g. HP,
CONTAIN. COORDINATOR, TURBINE
DECK, etc.)**
- **SUPERVISOR ROTATIONS**
- **LEADERSHIP SEMINARS**
- **FINANCIAL/SUPERVISOR SEMINARS**

MOV PROGRAM

- ✓ **EVOLVED FROM 85-03 AND 89-10**
- ✓ **SIGNIFICANT IMPROVEMENT IN PROGRAM**
- ✓ **ONE OF THE FIRST PROGRAMS CLOSED OUT**
- ✓ **PERIODIC TESTING TO CONTINUE**
- ✓ **CATEGORY 2 VALVE WORK**

FLOW ACCELERATED CORROSION

- SINGLE PERSON ACCOUNTABILITY AND OWNERSHIP**
- FORMAL SMALL BORE PROGRAM**
- MULTIDISCIPLINARY PERFORMANCE TEAM**
- MODELING AND INSPECTION OF LP EXTRACTION LINES INSIDE CONDENSER**
- AGGRESSIVE REPLACEMENT OF AFFECTED COMPONENTS**
- ACTIVE INVOLVEMENT IN INDUSTRY FAC GROUPS**

**STEAM
GENERATOR
INTEGRITY**

**STEAM GENERATOR
INTEGRITY COMMITTEE**

MISSION:

**TO MAINTAIN AND ENHANCE THE
INTEGRITY OF THE W3 STEAM
GENERATORS TO ENSURE SAFE,
EFFICIENT OPERATION AND TO
MAXIMIZE THE ECONOMIC LIFE OF
THE W3 STEAM GENERATORS**

COMPOSITION

CORPORATE PROJECT MANAGER

**DESIGN ENGINEERING; ISI AND
MECHANICAL SPECIALTIES**

CORPORATE AND SITE CHEMISTRY

LICENSING

**STEAM GENERATOR SYSTEM
ENGINEER**

STEAM GENERATOR INTEGRITY STRATEGIC PLAN

**LIST OF GOALS AND ACTIVITIES TO
HELP W3 ACHIEVE A 40 YEAR STEAM
GENERATOR LIFE**

**RESULTS/OUTCOME ORIENTED VICE
PROCESS ORIENTED**

**ADMINISTERS AND DOCUMENTS
INTEGRITY ACTIVITIES**

**PLANT PERFORMANCE
AND
RELIABILITY
IMPROVEMENT**

KEY SYSTEM CONCEPT

- ✓ RESOURCES FOCUSED ON
PLANT SAFETY &
RELIABILITY
- ✓ CONCEPT MESHES WITH
MAINTENANCE RULE SCOPE

PERFORMANCE IMPROVEMENT TEAM

IMPROVED THERMAL EFFICIENCY

MULTIDISCIPLINED TEAM



RECOVERED 10-15 MEGAWATTS

SECONDARY STEAM LEAKAGE REDUCTION

- **PUT INTO PLACE A BOP SMALL VALVE UPGRADE PROJECT TO REDUCE EXTERNAL BONNET LEAKAGE**
- **PACKING PROGRAM UPGRADE TO REDUCE EXTERNAL PACKING LEAKAGE**
- **MAINTENANCE LAPPING TOOL AND TRAINING UPGRADE**
- **INCREASED USE OF LEAK DETECTION EQUIPMENT IN IDENTIFYING SEAT LEAKING VALVES**

CHARGING PUMP SEAL RELIABILITY

- **WORKED WITH CEOG CHARGING PUMP WORKSHOPS**
- **W3 PACKING LIFE HAS GONE FROM AN INDUSTRY AVERAGE OF 800 HOURS TO APPROXIMATELY 1800 HOURS**
- **LONGEST LIFE PACKING HAS 3900 HOURS TO DATE**

RELIABILITY IMPROVEMENT TEAM (RIT)

- **ESTABLISHED TO EMPHASIZE AND FOCUS ON AFFECTING NUCLEAR AND PERSONNEL SAFETY, AVAILABILITY, RELIABILITY AND PERFORMANCE OF THE PLANT AND ITS SYSTEMS**
- **SUCCESSFULLY PRIORITIZED ISSUES, ESTABLISHED ISSUE OWNERSHIP AND FOCUSED RESOURCES TO ENSURE RESOLUTION OCCURS**

SETPOINT PROGRAM

- ✓ INITIATED IN 1990
- ✓ SCOPE OF 165 CALCS
- ✓ FOCUSED ON SAFETY RELATED AND RELIABILITY RELATED
- ✓ 9 SETPOINT CHANGES RESULTED
- ✓ PROGRAM COMPLETE BY 12/31/94



ENGINEERING ISSUES

DESIGN PROCESSES

TO PREVENT

COMMON MODE FAILURE

- **SELF ASSESSMENT**
- **AREAS BEING CONSIDERED FOR IMPROVEMENT**
 - SAFETY SIGNIFICANCE CONSIDERED FOR LEVEL OF EFFORT
 - ASSIGNMENTS BASED ON SAFETY SIGNIFICANCE
 - DESIGN INPUT DOCUMENTATION
 - DESIGN REVIEW DOCUMENTATION
 - CLARIFY RESPONSIBILITY AND GOALS FOR POST MOD TESTING

**COMPONENT/SYSTEM
FAILURE
ROOT CAUSE
ANALYSIS**

**RCA PROCESS
ENHANCEMENTS**

**COMMON APPROACH TO ROOT
CAUSE ANALYSIS AT EACH
FACILITY**

**DESK TOP GUIDE FOR RCA'S
TRAINING**

EXPANDED ROOT CAUSE CODES



RELIEF VALVE PROGRAM

SCOPE

- **RELIEF VALVE SETPOINT
REPORT**
- **RELIEF VALVE DESIGN
BASIS**
PRESSURE
CAPACITY

STATUS

- **SETPOINT REPORT COMPLETE**
- **ORIGINAL CALCULATIONS ACQUIRED**
- **CCW, ACCW, CS SYSTEMS COMPLETE**

FUTURE ACTIVITIES

- **COMPLETE ADDITIONAL CALCULATIONS**
- **SOLICIT NSSS INPUT**
- **ASSESS SKID MOUNTED VALVES**
- **ASSESS NON-SAFETY VALVES**

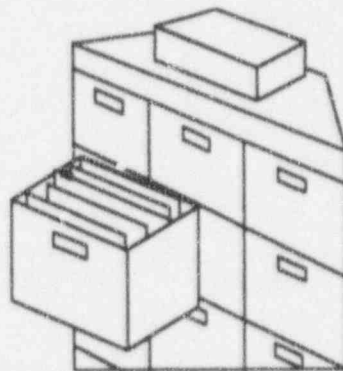
**SEISMIC
QUALIFICATION
UPGRADE
PROGRAM**

**CONSOLIDATE SQRT
FILE INFORMATION**

**ASSURE CORRECT
SQRT FILES
IDENTIFIED**

**REWORK SQRT FILE
REVISION PROCESS**

**POTENTIAL USE OF
SQUG**



VETIP

- **SELF ASSESSMENT PERFORMED**
 - OPEN ACTION ITEMS
 - ANNOTATION OF SERVICE BULLETINS OR INSTRUCTIONS
 - KEY COMPONENT SUPPLIER PROGRAM
 - DUPLICATE INFORMATION
 - TRAINING
- **RECOMMENDATIONS COMPLETE BY JULY 1996**

ELECTROLYTIC CAPACITOR REPLACEMENT

- **FAILURES RESULTED IN CHALLENGES TO SAFETY SYSTEMS**
- **ADDRESSES AGING OF ELECTROLYTIC CAPACITORS**
- **PERIODIC RIPPLE VOLTAGE MONITORING**
- **PHASE ONE 90% COMPLETE**

**RELOAD SAFETY ANALYSIS
GROUNDRULES IMPROVEMENT**

- SAFETY ANALYSIS INTERFACE
DOCUMENT BETWEEN W3 AND THE
FUEL VENDOR
- REFINE CONTENTS AND CLARITY OF
THIS DOCUMENT
- IDENTIFY ANALYSES THAT DEPEND ON
EACH GROUNDRULE



NEW INITIATIVES

**SAFETY
ANALYSIS
ENHANCEMENTS**

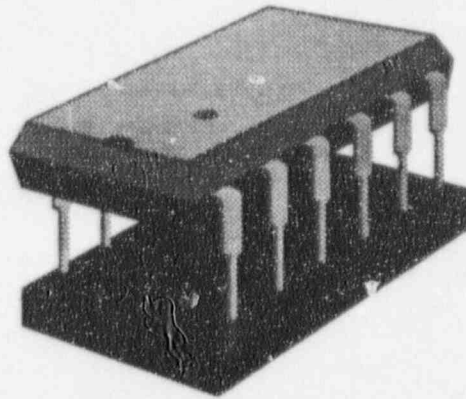
**SAFETY ANALYSIS DESIGN
BASIS DOCUMENTS
(DBD'S)**

- DOCUMENTS SAFETY ANALYSES DONE TO SUPPORT LICENSING OF WATERFORD 3
- INCLUDES EXISTING DESIGN INFORMATION AND CHANGES TO LICENSING BASIS ANALYSES
- COMPLETE PROJECT IN 1996

PROBABILISTIC SAFETY ASSESSMENT (PSA)

- ✓ SHUTDOWN RISK
- ✓ RISK MONITOR
- ✓ PSA TRAINING
- ✓ FUTURE ACTIVITIES

PMC REPLACEMENT



LARGEST MODIFICATION IN W3 HISTORY

COMPLETE SYSTEM REPLACEMENT

**WILL IMPROVE SYSTEM RELIABILITY AND
MAINTAINABILITY**

**REAL TIME AND ARCHIVED DATA
AVAILABILITY FOR ENGINEERS VIA THE
LAN**

FULLY REDUNDANT TO CARD LEVEL

**STATE OF THE ART DEC ALPHA
PROCESSORS**

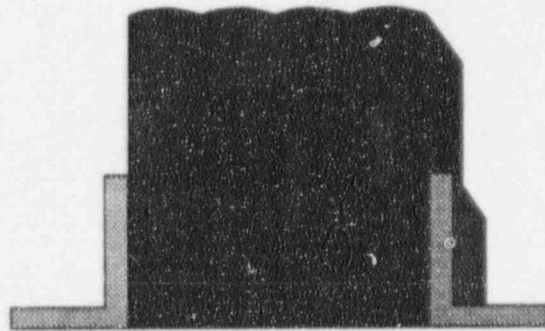
REPLACEMENT IN RF 7



AOV ACTUATORS

- **AGGRESSIVE TESTING FOR IDENTIFYING VALVE WORK, POST MAINTENANCE TESTING AND AOV PROGRAM TESTING**
- **DEVELOP PREVENTATIVE MAINTENANCE TASKS WHEN NEEDED**

ENGINEERING STUDY



RECOMMENDED STRATEGIES

ENGINEERING IS ONE FUNCTION

**PARTNERSHIP BETWEEN ENGINEERING AND
OPERATIONS & MAINTENANCE**

**FOCUS ON SAFETY & 'MATERIAL CONDITION'
OF THE PLANT**

**CONTINUOUS ENGINEERING PROCESS
IMPROVEMENTS**

UTILIZE INFORMATION TECHNOLOGY

**FOCUS RESOURCES ON SAFETY/RELIABILITY
ACTIVITIES**

CONCLUSION

★ENGINEERING ATTRIBUTES

SAFETY CONSCIOUS

PROACTIVE

SELF CRITICAL

★PARTNERING CONCEPT

★TECHNICAL COMPETENCE

★PLANT FOCUS