March 7, 2003

Mr. William Kanda
Vice President - Nuclear
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INSPECTION REPORT 50-440/03-02

Dear Mr. Kanda:

On February 14, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on February 14, 2003, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The NRC inspection team reviewed selected procedures and records, observed activities, and interviewed personnel. Specifically, the inspection focused on permanent plant modifications and changes, tests, and experiments evaluated in accordance with 10 CFR 50.59 requirements.

Based on this inspection, the team identified a Severity Level IV violation of NRC requirements associated with two examples of failure to perform a review as required by 10 CFR 50.59. Specifically, your staff failed to complete a documented safety evaluation for a change to the facility as described in your Updated Final Safety Analysis Report that involved: 1) the incorporation of new electrical standards affecting battery maintenance and acceptance criteria; and 2) changes to a plant drawing and procedure which reduced electrical separation criteria. We determined that this issue is a violation of 10 CFR 50.59 requirements. Because the violation was non-willful and non-repetitive and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

Additionally, two findings of very low safety significance (Green) were identified which involved violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you contest the Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Perry Nuclear Power Plant.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Mechanical Engineering Branch
Division of Reactor Safety

Docket No. 50-440
License No. NPF-58

Enclosure: NRC Inspection Report 50-440/03-02(DRS)

cc w/encl: B. Saunders, President - FENOC
K. Ostrowski, Director, Nuclear
Maintenance Department
V. Higaki, Manager, Regulatory Affairs
J. Messina, Director, Nuclear
Services Department
T. Lentz, Director, Nuclear
Engineering Department
T. Rausch, Plant Manager,
Nuclear Power Plant Department
Public Utilities Commission of Ohio
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	50-440/03-02
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	P.O. Box 97 A200 Perry, OH 44081
Dates:	February 10 through 14, 2003
Inspectors:	M. Holmberg, Reactor Inspector R. Daley, Reactor Inspector R. Winter, Reactor Inspector J. Ellegood, Resident Inspector
Approved by:	David E. Hills, Chief Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000440-03-02; First Energy Nuclear Operating Company; on 02/10-14/2003; Perry Nuclear Power Plant. Permanent plant modification and changes, tests, and experiments.

The baseline inspection was conducted by resident and region-based inspectors. The inspectors identified two Green findings associated with Non-Cited Violations and one Severity Level IV Non-Cited Violation (NCV) associated with a failure to perform 10 CFR 50.59 safety evaluations. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000, and can be found on the NRC website at: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- NCV. The team identified a Severity Level IV Non-Cited Violation associated with the licensee's failure to perform safety evaluations in accordance with 10 CFR 50.59 for changes made to the facility as described in the Updated Final Safety Analysis Report. Specifically, the licensee failed to complete a documented safety evaluation for a change to the facility as described in the Updated Final Safety Analysis Report that involved: (1) the incorporation of new electrical standards affecting battery maintenance and acceptance criteria; and (2) changes to a plant drawing and procedure which reduced electrical separation criteria.

  Because the Significance Determination Process (SDP) is not designed to assess the significance of violations that potentially impact or impede the regulatory process, this issue was dispositioned using the traditional enforcement process in accordance with Section IV of the NRC Enforcement Policy. However, the results of the violation, that is, the failure to evaluate changes in battery maintenance and acceptance criteria and electrical separation criteria, were assessed using the SDP.

  The team considered this issue of more than minor significance, because if left uncorrected, the finding could become a more significant safety concern. The team was concerned that two examples of failure to perform a documented safety evaluation in the relatively small sample of screenings reviewed indicated a potential for a more significant safety concern if the underlying cause was not identified and corrected. For the examples identified, the team determined that the issue was of very low safety significance because: (1) other licensee's had been successful in evaluating changes under 10 CFR 50.59 from the older IEEE 450 requirements to compliance with the newer IEEE 450-1995 requirements; and (2) the reduced electrical separation distance that the licensee had incorporated into their design drawing and configuration met the requirements in IEEE 384 which had been endorsed by the NRC in Regulatory Guide 1.75. Therefore, the results of the violation were determined to be of very low safety significance and the violation of 10 CFR 50.59 was classified as a Severity Level IV violation.

Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program, this issue is being treated as a Non-Cited Violation, consistent with the NRC Enforcement Policy (Section 1R02).

- Green. The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the for the licensee's inadequate design reviews associated with the installation of half-couplings on a B train 14 inch emergency service water elbow. The licensee installed half-couplings in response to a through-wall leak and an area of wall loss identified on a 14 inch emergency service water elbow. However, the licensee's design review was inadequate in that, it failed to include the requirements of Section XI of the American Society of Mechanical Engineers Code. Specifically, the licensee failed to identify the cause of the flaw, failed to adequately characterize the dimensions of the flaw, nor was the potential growth of these flaws considered. Further, the repair design did not include flaw removal or component replacement.

  This finding was more than minor based on the degradation in the plant's design basis for the B train emergency service water elbow. This degraded elbow affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of the emergency service water system, because the non-Code repairs (half-couplings) and associated piping flaws could have resulted in premature failure of the elbow. The finding was determined to be a licensee performance deficiency of very low safety significance (Green) by the significance determination process because the finding was a design or qualification deficiency that did not result in a loss of system function (Section 1R17).

- Green. The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's inadequate design review associated with installation of a rupture disc in the exhaust piping of the division 3 high pressure core spray emergency diesel generator. This finding was self-revealed on October 25, 2000, after the diesel generator was placed in service following this modification, the rupture disc failed in less than 3 minutes due to pressure induced fatigue. The licensee's design review for the rupture disc was inadequate because it did not adequately consider pressure induced fatigue loading.

  This finding was more than minor based on the failed rupture disc impacting the mitigating systems objective of ensuring the availability, reliability, and capability of the division 3 high pressure core spray diesel generator in response to initiating events. Specifically, the failed rupture disc created an opening which could have allowed foreign material to enter the exhaust system and cause premature failure of the division 3 diesel generator. The finding was determined to be a licensee performance deficiency of very low safety significance (Green) by the significance determination process because the finding was a design or qualification deficiency that did not result in a loss of system function (Section 1R17).

B. <u>Licensee Identified Findings</u>

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

Review of Evaluations and Screenings for Changes, Tests, or Experiments

a. Inspection Scope

From February 10, 2003, through February 14, 2003, the team performed an on-site (room 205 of the site administrative building) review of 10 safety evaluations required by 10 CFR 50.59 and 14 safety evaluation screenings or regulatory applicability determinations (RADs). The team reviewed these documents to ensure compliance with the requirements of 10 CFR 50.59. The team also referred to Nuclear Energy Institute 96-07, Guidelines of 50.59 Evaluations, Revision 1, to determine acceptability of the completed evaluations, and screenings. The NRC endorsed the Nuclear Energy Institute 96-07 in Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, November 2000. The team also consulted Inspection Manual, Part 9900, 10 CFR Guidance: 50.59. The documents reviewed during this inspection are listed at the end of the report.

b. Findings

b.1 Failure to Perform 10 CFR 50.59 Safety Evaluations

Introduction

The team identified a Severity Level IV Non-Cited Violation associated with the licensee's failure to perform safety evaluations in accordance with 10 CFR 50.59 for changes made to the facility as described in the Updated Final Safety Analysis Report (UFSAR).

Description

On January 22, 2002, the licensee completed RAD 02-00055. In this RAD, the licensee evaluated the change to UFSAR Table 1.8-1 which identified the license basis commitments to Regulatory Guides. This UFSAR table identified compliance with Regulatory Guide (RG) 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants." Additionally, the UFSAR table identified compliance with IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." The licensee had changed the UFSAR Table 1.8-1 to indicate compliance with IEEE 450-1995 instead of IEEE 450-1980. This change was substantial in that, the 1995 standard had reduced the required battery testing frequency and had changed the acceptance criteria for testing the battery. However, the licensee incorrectly concluded in RAD 02-00055 that a documented review in accordance with 10 CFR 50.59 for this change to the facility (battery testing and acceptance) as described in the UFSAR was not required.

On November 15, 2002, the licensee completed 10 CFR 50.59 screen 02-01534. The licensee performed this screen to evaluate a revision to drawing D-214-004 and procedure PAP-0204 which added electrical separation requirements for the installation of temporary cables. In one of these drawing changes, the licensee applied a separation criteria of 6 inches between a temporary cable with a barrier (such as temporary cable in conduit) and a Class 1E cable tray. The team noted that this condition was analogous to establishing a separation requirement of 6 inches between non-Class 1E cable in conduit and a Class 1E cable tray. The licensee concluded that a documented safety evaluation in accordance with 10 CFR 50.59 was not required based on, “The separation criteria adapted for temporary cable installations as documented on drawing D-214-004 is consistent with that described in UFSAR, Section 8.3.1.4.1.7 and IEEE-384 for permanent plant cable installations.” The team agreed that this change was consistent with IEEE-384, however it was not consistent with more restrictive electrical separation requirements identified in UFSAR Section 8.3.1.4.1.7. This UFSAR Section stated, “Separation between conduit and tray system containing Class 1E circuits or non-Class 1E circuits is 3 foot horizontal and 5 foot vertical, except for the cable spreading area which is reduced to 1 foot horizontal and 3 foot vertical.” Consequently, the changes the licensee made to the plant’s design drawing and implementing procedure were not in accordance with the UFSAR Section 8.3.1.4.1.7. Therefore, a documented safety evaluation was required in accordance with 10 CFR 50.59 for this change to the facility as described in the UFSAR.

Analysis

Because violations of 10 CFR 50.59 are considered to be violations that could potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the SDP. However, the results of this violation are assessed using the SDP. In these cases, the licensee’s failure to perform safety evaluations in accordance with 10 CFR 50.59 resulted in inconsistencies between the UFSAR and the plant’s design basis and configuration for electrical separation requirements, and the maintenance and testing associated with the station batteries.

The team considered this issue of more than minor significance, because if left uncorrected, the issue could become a more significant safety concern. The team was concerned that two examples of failure to perform a documented safety evaluation in the relatively small sample of screenings reviewed indicated a potential for a more significant safety concern if the underlying cause was not identified and corrected. For the examples identified, the team determined that the issue was of very low safety significance. The team concluded this because; 1) other licensee’s had been successful in evaluating changes under 10 CFR 50.59 from the older IEEE 450 requirements to compliance with the newer IEEE 450-1995 requirements, and 2) the reduced electrical separation distance that the licensee had incorporated into their design drawing and configuration met the requirements in IEEE 384 which had been endorsed by the NRC in RG 1.75. The team considered that if the licensee had completed a safety evaluation in accordance with 10 CFR 50.59 for these examples, it would have likely been successful and therefore this issue was determined to be of very low safety significance.

Enforcement

10 CFR 50.59(d)(1) stated, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides for the determination that the change, test, or experiment does not require a license amendment. Contrary to these requirements, as of February 14, 2003, the licensee failed to perform two written safety evaluations for changes to the facility as described in the UFSAR documented in RAD 02-00055 completed January 22, 2002 and screen 02-01534 completed on November 15, 2002. Specifically, the RAD and screening did not provide a written evaluation which provides for the determination that the change, test, or experiment did not require a license amendment. The results of this violation were determined to be of very low safety significance; therefore, this violation of the requirements contained in 10 CFR 50.59 was classified as a Severity Level IV Violation. However, because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program (condition report (CR) 03-00724), it is considered a Non-Cited Violation (NCV 50-440/03-02-01) consistent with Section VI.A.1 of the NRC Enforcement Policy.

b.2 Inadequate 10 CFR 50.59 Safety Evaluation

Introduction

The team identified an unresolved item associated with an inadequate safety evaluation. Specifically, the licensee failed to assess the affects of noble metal chemical addition (NMCA) on the peak cladding temperature (PCT) under accident conditions due to potential catalytic affects on the Zirconium (Zr) metal-water reaction rate.

Description

In February of 2001, the licensee approved a safety evaluation 01-0007, "TXI-321 Noble Chemical Metal Addition," to document the safety evaluation for the addition of NMCA on the reactor primary coolant system components and fuel cladding. The NMCA process was implemented to reduce the susceptibility of components within the core and reactor coolant systems to stress corrosion cracking. The licensee concluded in evaluation 01-0007, that implementation of this activity would not require prior NRC approval based on meeting the criteria of 10 CFR 50.59. The licensee subsequently implemented the NMCA at the Perry site. However, the team identified that the licensee had failed to consider the potential affects of NMCA on PCT under accident conditions.

The noble metals used in the NMCA act as a catalyst to facilitate recombination of hydrogen and oxygen. Under normal operation, oxygen is produced in the core by radiological decomposition of water and is removed by combination with hydrogen which is added to the reactor coolant system. During a loss of coolant accident (LOCA) hydrogen can be produced as a byproduct of the Zr metal-water reaction and oxygen is present in the emergency injection water sources such as the condensate storage tank or the containment sump. Under LOCA conditions, the NMCA could cause the hydrogen to be removed at an accelerated rate from the product side of the Zr metal-water reaction ($Zr + 2H_2O \rightarrow ZrO_2 + \underline{2H_2} + \text{heat}$) which may tend to increase the reaction rate. This would occur because the NMCA has been demonstrated to increase the reaction rate for recombination of hydrogen and oxygen ($2H_2 + O_2 \rightarrow 2H_2O$ + heat). Additionally, the NMCA may directly increase the reaction rate for the Zr metal-water reaction through catalytic action, by lowering the initiation temperature for this reaction which is normally about 1700 degrees Fahrenheit (e.g. lower the required activation energy). Thus, the addition of NMCA could increase the heat input to the fuel cladding due to the enhanced exothermic Zr metal-water and hydrogen-oxygen recombination reactions. The heat produced from the Zr metal-water reaction is considered in computer models used to evaluate the emergency core cooling system performance with respect to PCT (10 CFR 50.46). If the Zr metal-water and hydrogen-oxygen reaction rates are substantially increased by NMCA, the heat input to the cladding following a design basis LOCA could be increased above that accounted for in the computer models used to calculate PCT. If the increase in heat input to the cladding was substantial, the team was concerned that it could increase the likelihood of cladding failure by thermally induced failure mechanisms related to creep or creep-rupture in a post LOCA environment. Cladding failure would result in increased fission product activity released into the coolant which could potentially impact the on and off-site dose consequences previously considered for this accident. The licensee entered this potentially generic concern into the corrective action system (CR 03-0071) and completed an operability determination.

The licensee considered that this issue did not affect operability principally because their fuel reload analysis showed a substantial margin to the PCT limit (e.g., maximum PCT of 1560 degrees Fahrenheit for the limiting GE 14 fuel configuration, vice the 2200 degree Fahrenheit limit). Additionally, the licensee’s operability determination indicated that the steam environment (with highest PCT) which could exist for 2-3 minutes after a LOCA would not allow substantive amounts of oxygen to be produced through radiological decomposition and without a source of oxygen recombination of hydrogen and oxygen (facilitated by NMCA) can not occur. The team concluded that specific information on post LOCA PCT profiles in concert with potential sources and levels of oxygen and hydrogen may be needed to resolve this issue. The team considered this issue an unresolved item pending review of the revised 50.59 safety evaluation on the NMCA process (URI 50-440/03-02-02).

1R17 Permanent Plant Modifications (71111.17B)

Review of Recent Permanent Plant Modifications

a. Inspection Scope

From February 10, 2003, through February 14, 2003, the team performed an on-site (room 205 of the site administrative building) review of 10 permanent plant modifications. Additionally, the team reviewed a sample of setpoint changes, equivalency evaluations and a commercial grade dedication that were performed by the licensee's engineering staff. The team reviewed these modifications and supporting calculations to verify that the completed design changes were in accordance with design and licensee requirements identified in the Perry UFSAR and Technical Specifications. The teams also reviewed applicable industry design standards, such as the Institute of Electrical and Electronics Engineers or American Society of Mechanical Engineers Code (ASME) Sections XI and Section III, to evaluate acceptability of the modifications.

Additionally, the team reviewed applicable post-modification testing to verify that the system, and associated support systems, functioned properly and that the modification accomplished its intended function.

b. <u>Findings</u>

b.1 <u>Inadequate Design Review for Emergency Service Water Elbow Modifications</u>

<u>Introduction</u>

Green. The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s inadequate design reviews associated with the installation of half-couplings on a 14 inch emergency service water (ESW) elbow.

<u>Description</u>

On November 6, 2000, the licensee identified (CR 00-3433) a through-wall leak on a 14 inch ESW elbow downstream of valve 1P45-F0541B at the outlet of the B ESW heat exchanger. The licensee did not identify a root cause for this degradation and used only an ultrasonic thickness gauge to determine the extent of the flaw. The licensee evaluated and approved the design of a 1-1/2 inch half-coupling with plug to cover up the through-wall hole in the 14 inch elbow in engineering change package (ECP) 00-8092. This modification was approved and installed on November 7, 2000.

On August 2, 2001, the licensee identified (CR 01-2974) a new area of wall loss on the same 14 inch ESW elbow with the through-wall leak as discussed above. The licensee did not identify a definitive cause for the degradation and again used only an ultrasonic thickness gauge to determine the extent of the flaw. On October 29, 2001, the licensee evaluated and approved ECP 01-8043 for the design of a 1-1/2 inch half-coupling with plug to cover up the new area of degradation. The degraded area had reduced the nominal 0.375 inch wall thickness to 0.148 inch. The licensee installed the half-coupling and plug over the degraded area on April 4, 2002.

The licensee had used an ultrasonic thickness gage on the degraded ESW elbow, which was not adequate to detect or characterize the extent of these flaws (hole/degraded areas). Specifically, the ultrasonic thickness gage utilized a 0 degree ultrasonic transducer, which was not capable of detecting planar flaws (e.g., cracks). To confirm that cracking (planar flaws) did not exist, the licensee should have performed an ultrasonic examination (UT) using an angle beam type transducer. The team was concerned that if undetected cracking (planar flaws) existed, the residual weld tensile stress induced in the pipe wall for these highly constrained repair weldments could have increased the rate of flaw growth and caused a premature failure of the elbow. Therefore, the team questioned the operability of the pipe elbow and consequently the B ESW train. On February 12, 2003, the team's concern prompted the licensee to perform additional UT (using 0 degree dual element, 60 degree and 70 degree angle beam transducers) of the ESW elbow to characterize the nature of the base metal surrounding the repaired area. During this inspection, the licensee identified three new flaws, two of which were just under the attachment weld of the half-coupling installed in 2000. The deepest of these flaws reduced the 0.375 nominal wall thickness to 0.14 inches. However, due to the half-couplings installed over the flaw areas, the original flaws could not be measured.

On February 14, 2003, the team reviewed the licensee’s operability determination 03-00699. As part of this operability determination, the licensee identified a number of potential causes for the observed degradation in this ESW elbow including erosion/corrosion and microbiological corrosion. In this evaluation, the licensee evaluated a bounding sized planar and non-planar flaws, which were assumed to exist under the half-couplings. The licensee then compared the bounding sized flaws to the maximum allowable sized flaws calculated in accordance with the methodology discussed in Code Case N-513, “Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping” (NRC approved methodology). However, the team identified that the licensee had not explicitly documented how the acceptance criteria of Section 4.0 of Code Case N-513 were met. The licensee subsequently revised their operability determination and demonstrated that the flaws left within the ESW elbow did not reduce the structural margins below the acceptance criteria for planar flaws in Section 4.0 of Code Case N-513. Therefore, the licensee considered the ESW elbow operable until the upcoming refueling outage. Because the licensee determined that the affected component was operable, the team did not have an immediate safety concern.

The team’s concern for the B loop elbow, appeared to prompt the licensee to perform UT of the same elbow in the A loop of the ESW system prior to the normally scheduled monthly UT checks. During this UT, the licensee identified a degraded area (0.109 inch thick) which was below the required minimum wall thickness (0.116 inch) (CR 03-0833). The licensee again applied Code Case N-513 to demonstrate operability of the degraded area until the upcoming refueling outage. Because the licensee determined that the affected component was operable, the team did not have an immediate safety concern.

The licensee’s design reviews for ECP 00-8092 and ECP 01-8043 were inadequate in that, the licensee failed to incorporate the applicable requirements of Section XI of the ASME Code. Specifically, the licensee had failed to identify a definitive cause for the flaws, failed to adequately characterize the dimensions of the flaws, nor was the potential growth of these flaws considered. Each of these actions was required by the ASME Code Section XI (paragraphs IWA -3300, IWA-4130, and IWD-3000). The licensee's repair design did not include removal of the flaw as required by paragraph IWA-4300 of Section XI. Additionally, the use of a half-coupling and plug to cover up the flaw was not a recognized repair method identified in paragraph IWA-4130 of Section XI. Further, the licensee’s repair method was not consistent with NRC approved alternative repair methods discussed in Code Cases N 513, and N 523 or Generic Letter 90-05 "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2 and 3 piping."

<u>Analysis</u>

The team reviewed this finding against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of IMC 0612, “Power Reactor Inspection Reports.” In particular, the team compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the team concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the team compared this performance deficiency to the minor questions contained in Section C, “Minor Questions,” to Appendix B of IMC 0612. The team concluded that the finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on April 29, 2002, because the finding: (1) involved the design control and equipment performance attributes of the mitigating systems cornerstone; and (2) affected the mitigating systems objective of ensuring the availability, reliability, and capability of the ESW system in response to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, these non-Code repairs represented a degradation in the plant’s design basis (ASME Code Sections III and XI) for the B train emergency service water elbow. This degraded elbow affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of the emergency service water system, because these non-Code repairs and associated piping flaws could have resulted in premature failure of the elbow.

The team evaluated the finding using Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding was a design or qualification deficiency that did not result in loss of component/system function. Therefore, the team screened this finding to have very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion III, “Design Control,” required, in part, that applicable regulatory requirements and design basis are correctly translated into specifications, drawings, procedures and instructions. 10 CFR 50.55a(g)(4) required, in part, that throughout the service life of boiling or pressurized water cooled nuclear power facility, components, which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the requirements of Section XI (applicable regulatory requirement). The 14 inch ESW elbow downstream of valve 1P45-F0541B was a Code Class 3 component. Contrary to these requirements, in engineering change package 00-8092 approved on November 7, 2000, and 01-8043 approved on October 25, 2001, the licensee failed to incorporate the applicable regulatory requirements of the 1989 Edition of ASME Code Section XI, paragraphs (IWA-3300, IWA-4130, IWA-4300, IWD-3000) associated with flaw evaluation, flaw removal and component repair into applicable specifications, drawings, procedures and instructions for the 14 inch ESW elbow downstream of valve 1P45-F0541B. Consequently, the licensee had to conduct substantial additional non-destructive examinations and flaw growth analysis to confirm that the system was operable. Failure to incorporate the applicable regulatory requirements into these engineering change packages is a violation of 10 CFR 50 Appendix B, Criterion III. This violation is associated with an NRC identified finding that is characterized by the significance determination process as having very low risk significance (Green) and is being treated as a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion III, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-440/03-02-03). The licensee entered this finding into the corrective action system (CR 03-0744).

b.2 Inadequate Design Review for Rupture Disc Modification

Introduction

Green. The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s inadequate design review associated with installation of a rupture disc in the exhaust piping of the division 3 high pressure core spray (HPCS) emergency diesel generator (EDG). This finding was self revealed on October 25, 2000, after the diesel generator was placed in service following this modification, the rupture disc failed in less than 3 minutes due to pressure induced fatigue.

Description

The team identified a self-revealing finding associated with ECP 00-6009, in which the licensee evaluated and approved (on August 22, 2000) the design of a rupture disc for installation into the exhaust piping for the division 3 HPCS EDG. The normal EDG exhaust vent piping was located on the roof of the EDG building and was susceptible to damage induced by tornado or high wind driven projectiles. The licensee installed a rupture disc in an exhaust branch line to serve as an alternate vent path in the event that the normal exhaust vent piping was restricted or damaged. The rupture disc was bolted into the terminal end of a vertical run of exhaust duct. If the rupture disc was actuated, it would allow EDG exhaust gases to vent into a partially enclosed concrete corridor with open doorways to the building roof.

October 25, 2000, after the EDG was placed in service following this modification, the rupture disc failed in less than 3 minutes. The licensee promptly identified the failed rupture disc during the initial EDG operation and implemented a modification to remove the rupture disc to allow the exhaust to vent directly through this branch. The licensee also added steel screens from the floor to the ceiling around the perimeter of the open vertical run of exhaust duct. This barrier served to protect the open duct from introduction of foreign materials. However, with the modified exhaust configuration, the high temperature EDG exhaust gases contacted the concrete ceiling of the building structure above the opening and caused concrete spalling (chips). If these concrete chips had fallen into the open vertical run of duct which used to contain the rupture disc, the EDG could have been rendered inoperable. Specifically, concrete debris could have entered the vertical exhaust duct and lodged in the turbocharger or other engine components causing failure of the EDG. Fortuitously, the licensee identified this condition before introduction of concrete chips and added a steel plate mounted to staging above the open exhaust vent to provide a barrier from falling concrete chips. Because the licensee determined that the affected component was operable, the team did not have an immediate safety concern.

The licensee had a vendor perform an investigation of the failed rupture disc and provide a report “Failure Analysis Report Diesel Generator Rupture Disc” dated November 22, 2000. In this report, the vendor concluded that the rupture disc failed by fatigue induced by cyclic exhaust pressure transients during normal diesel engine operation. The licensee’s had not adequately considered the impact of normal cyclic pressure exhaust loads on the design service life of the rupture disc in ECP 00-6009.

Analysis

The team reviewed this finding against the guidance contained in Appendix B, “Issue Dispositioning Screening,” of IMC 0612, “Power Reactor Inspection Reports.” In particular, the inspectors compared this finding to the findings identified in Appendix E, “Examples of Minor Issues,” of IMC 0612 to determine whether the finding was minor. Following that review, the team concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the team compared this performance deficiency to the minor questions contained in Section C, “Minor Questions,” to Appendix B of IMC 0612. The team concluded that the finding was greater than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on April 29, 2002, because the finding: (1) involved the design control and equipment performance attributes of the mitigating systems cornerstone; and (2) affected the mitigating systems objective of ensuring the availability, reliability, and capability of the division 3 HPCS EDG in response to initiating events to prevent undesirable consequences (i.e., core damage). When the licensee removed the failed rupture disc and declared the EDG operable, the EDG was in a less reliable configuration. Specifically, the removed rupture disc created an opening which could have allowed foreign material (e.g. concrete chips) to enter the exhaust system and cause premature failure of the division 3 HPCS EDG. The licensee subsequently installed foreign material exclusion barriers to ensure the continued reliability of the HPCS EDG.

The team evaluated the finding using Manual Chapter 0609, “Significance Determination Process,” Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding was a design or qualification deficiency that did not result in loss of component/system function. Therefore, the team screened this finding to have very low safety significance (Green).

Enforcement

10 CFR 50, Appendix B, Criterion III, “Design Control,” required, in part, that measures shall provide for verifying and checking the adequacy of design, such as by the performance of design reviews. Contrary to these requirements, in engineering change package 00-6009 approved on August 22, 2000, and installed on October 25, 2000, the licensee failed to consider the in-service cyclic pressure induced fatigue loads on the design life of the component. Consequently, the rupture disc failed within 3 minutes after commencing EDG operation. Failure to perform an adequate design review and identify the in-service cyclic pressure transients as a design parameter and incorporate this into the rupture disc design is a potential violation of the 10 CFR 50 Appendix B Criterion III, “Design Control.” This violation is associated with a self revealing finding that is characterized by the significance determination process as having very low risk significance (Green) and is being treated as a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-440/03-02-04). The licensee entered this finding into the corrective action system (CR 00-3320 and CR 01-2283).

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team performed an on-site (room 205 of the site administrative building) review of a sample of permanent plant modifications and 10 CFR 50.59 program problems that were identified by the licensee and entered into the corrective action program in condition reports. The team reviewed these condition reports to confirm that the licensee had appropriately described the scope of the problems as documented in the condition reports. Additionally, the team's review included confirmation that the licensee had an appropriate threshold for identifying issues and had implemented effective corrective actions related to design issues. The specific corrective action documents that were reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting

The team presented the inspection results to Mr. W. Kanda and other members of licensee management at the conclusion of the inspection on February 14, 2003. The licensee acknowledged the information presented during this meeting. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

W. Kanda, Vice President - Nuclear
T. Rousch, Plant Manager
T. Lentz, Engineering Director
D. Haviland, Supervisor Structural Mechanical Design
C. Angstadt, Engineering Assessment Board Chairman
D. Gartner, I&C Lead Engineer
K. Russel, Compliance Engineer
D. Miller, Staff Consultant
J. Zarea, Electrical Design Engineering

NRC

Ray Powell, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-440/03-02-01	NCV	Failure to perform a safety evaluation for changes to the plant as described in the UFSAR.
50-440/03-02-02	URI	Increased heat input on PCT from the Zr metal-water and hydrogen-oxygen reactions facilitated by Noble Metals.
50-440/03-02-03	NCV	Failure to perform adequate design reviews for installation of half-couplings on a B train emergency service water elbow.
50-440/03-02-04	NCV	Failure to perform adequate design reviews for installation of a rupture disc in the exhaust piping of the division 3 high pressure core spray diesel generator.

Closed

50-440/03-02-01	NCV	Failure to perform a safety evaluation for changes to the plant as described in the UFSAR.
50-440/03-02-03	NCV	Failure to perform adequate design reviews for installation of half-couplings on a B train emergency service water elbow.
50-440/03-02-04	NCV	Failure to perform adequate design reviews for installation of a rupture disc in the exhaust piping of the division 3 high pressure core spray diesel generator.

Discussed

None

LIST OF DOCUMENTS REVIEWED

1R02 Evaluations of Changes, Tests, or Experiments

10 CFR 50.59 Safety Evaluations

00-0014	SMRF 99-5010, R/0 and associated USAR Change Request (CRF)	April 3, 2000
00-0021	SMRF 99-5056 Rev. 0 and associated USAR Change Request	March 31, 2000
00-0052	DCP 00-6009 Revision 0 and Associated UFSAR Change Request	August 24, 2000
00-0084	Safety Evaluation of SMRF 00-5018	November 21, 2000
00-0089	Safety Evaluation for Temporary Modification 1-00-004	November 30, 2000
01-0001	TXI-321 Noble Chemical Metal Addition	January 30, 2001
01-0007	TXI-321 Noble Chemical Metal Addition	February 8, 2001
01-0018	DCN 5910 R/0 and UFSAR Change Request	March 18, 2001
01-0026	Temp Mod 1-01-002 Rev 0	April 16, 2001
01-0036	DCP 99-5020	October 23, 2001

10 CFR 50.59 Safety Evaluation Screenings or Regulatory Applicability Determinations

01-00384	Plant Data Book, Administrative Requirements Section, Section 7.5.5, Diesel Generator Maintenance Program procedure change	October 08, 2001
02-00055	Changed USAR Table 1.8-1 to change commitment for compliance with IEEE 450-1980 to compliance with IEEE 450-1995	January 22, 2002
02-00384	ECP 02-00028 and USAR Change Request 02-034	July 16, 2002
02-00436	RPV Steam Dome and RHR Cut-in Permissive High Pressure Channel B Calibration for 1B21-N678B	Revision 0
02-00453	RPV Steam Dome and RHR Cut-in Permissive	April 22, 2002

02-00530	Installation of Jumpers to Defeat Faulty Circuitry for the Extraction Steam to the Steam Jet Air Ejector and Off-Gas Preheater Drain Pot Level Switch N22-N273	Revision 0
02-00569	Replace Existing PSA Snubber for 1N25-H0027 with a Lisega Snubber	Revision 0
02-00940	Emergency Closed Cooling System “A loop	August 8, 2002
02-01210	Elimination of MSIV Stem Leakoff Piping	October 3, 2002
02-01301	Temporary Modification to Eliminate Locked in High Oil Level Annunciator for Reactor Recirculation Motor A	Revision 0
02-01534	Revision to Drawing D-214-004 (ECR-02-0303) and Procedure PAP-0204 to add separation requirements for the installation of temporary cables.	November 15, 2002
02-01617	Augmented Visual Inservice Inspection/Examination of Safety-Related Snubbers	December 9, 2002
02-01669	ESW Pump A(B) and Valve Operability Test	December 11, 2002
02-01799	Scram Discharge Volume Vent and Drain Valves Operability Test	December 23, 2002

<u>Condition Reports</u>

01-1332	RFO8 Need: Circ WTR SYS Steel to Concrete Groundwater In-Leakage	May 12, 2001
01-3269	Fire Protection Program	September 6, 2001
01-3683	Improperly prepared Regulatory Applicability Determinations	October 17, 2001
01-3747	No 10CFR50.59 Evaluation Performed for Changes to P45 SVI’s with Capacitance Sensing Instrumentation	October 25, 2001
01-4365	50.59 Self-Assessment Results	December 26, 2001
02-00250	Plant underdrain temperary modification 1-02-001 50.59 not conservative	January 23, 2002

02-00272	Temporary Modification with No 10CFR50.59 Evaluation	January 28, 2002
02-00745	Temporary Modification Installed with no 10 cfr 50.59 Evaluation Performed	May 13, 2002
02-01332	M&TE for monitoring RWCU pump mechanical seal cooling line for RWCU pumps	April 30, 2002
02-01422	Equivalent Changes Incorrectly Exempted from 10 cfr 50.59 Applicability	May 8, 2002
02-03487	Service water chlorination system	September 12, 2002
02-03788	DCP 96-044 safety evaluation 00-0007	October 10, 2002

<u>Condition Reports Initiated as the Result of NRC Inspection</u>

03-00478	50.59 database incomplete	January 30, 2003
03-00695	Potential failure to perform required 50.59 evaluation related to RAD 02-00055	February 11, 2003
03-00721	Nuclear fuel	February 12, 2003
03-00723	Raceways (conduits and cable trays)	February 12, 2003
03-00724	Electrical raceways (conduits and cable trays)	February 12, 2003
03-00743	NRC identified a number of issues with 50.59 documentation quality	February 13, 2003

<u>Other Documents</u>

NEI 96-07	Guidelines for 10 CFR 50.59 Implementation	Revision 1
License Amendment No. 74	Amendment No. 74 to Facility Operating License N0. NPF-58 - Perry Nuclear Power Plant, Unit No. 1 (TAC No. M92190)	November 16, 1995
Change Request 01-090	Replacement of Division 3 100Ah batteries with 250 Ah batteries	March 6, 2000

<u>Procedures</u>

NOP-LP-4003	Evaluations of Changes, Tests and Experiments	Revision 0

PAP-305	Safety Evaluations	Revision 8
PAP-520	Changes to the Updated Safety Analysis Report and Other Licensing Documents	Revision 6

1R17 Permanent Plant Modifications

Calculations

R48-05	Overpressure Protection Rupture Disc	Revision 3
R48-21	Rupture Disc Qualification	Revision 0
P42-005	Emergency Closed Cooling Surge Tank Hi/Lo Alarm 1P42N0131A,B	Revision 4
1P42-H0183	Qualification of Instrument Support 1P42-H0183 Emergency Closed Cooling Surge tank Elevation 665'-0"	Revision 0
X-302	Surge Tank Instrument Support	Revision 1
X-506	I/F 1P42-A001A	Revision 1
R45-T04 R/7	Calc Adjusts the 24 hour Inventory and the 7 Day Inventory	November 12, 2001

Condition Reports

99-2439	Stuck Relay in ESW system caused by current induction	October 9, 1999
99-3033	Fuel Oil Transfer Pump was inadvertently started	December 5, 1999
00-2216	While attempting to declare the Unit 2 Div 3 Battery Operable, a DCP found Open	July 21, 2000
00-2495	Installation of the Isolation Transformer did not Correct the Problem	July 17, 2000
00-3433	ESW B through wall pipe leak downstream of 1P45-F0541B	November 6, 2000
00-3320	Exhaust system rupture on division 3 diesel generator opened after modification	October 25, 2000
01-0230	Division 1,2, and 3 testable rupture disc	January 22, 2001
01-1277	1BN31 Accumulator Tank With Magnetrol Level Switch	March 10, 2001
01-1485	Breaker F1F05 tripped	March 19, 2001

01-1715	ECC-B Surge tank Valve 1P42-F0668 Out of Position	April 2, 2001
01-2181	FME, Metal Chunk that is not Part of the Valve was removed from 1n27F0170	May 12, 2001
01-2283	Diesel generator testable rupture disc supply fan dampers and core water pumps	May 21, 2001
01-2620	Premature Use of Calculation Results	July 2, 2001
01-2949	Auto Operation OD Damper Will Not Work	July 31, 2001
01-2994	Emergency service water loop B 14 inch piping elbow degraded areas	August 2, 2001
01-3776	P42 Latent Issues ECC Hi/Lo Temp ARI doesn't address temp Control Valve Failure	October 29, 2001
01-4073	Latent Issues Review- Procedural Enhancements	November 27, 2001
01-4149	Design control audit	December 3, 2001
02-00723	Latent Issues ESW Valves do not Have Hot Short Mod	March 12, 2002
02-01458	RFA to Evaluate Strain Relief Used During the RC&IS (J50/P50) Cable Modification	May 14, 2002
02-01465	Error in the EQ Evaluation for ECP 02-0028 Rev. 0	May 14, 2002
02-01650	Emergency Closed Cooling Valves' Differential Pressure	May 28, 2002
02-01869	Safety Class Code Designators on Drawing D-302-621	June 12, 2002
02-03099	1P45-F040A failed to close	September 5, 2002
02-03389	NQA finding during quarterly assessment	September 20, 2002
02-03604	Unit 1 Division 3 Battery	October 2, 2002
02-04210	Air block valve	October 22, 2002

Condition Reports Initiated as the Result of NRC Inspection

03-00699	Emergency service water elbow downstream of 1P45F0541B	February 11, 2003
03-00733	Commercial grade dedication process weakness	February 12, 2003
03-00739	Division 3 diesel generator exhaust rupture disc	February 13, 2003
03-00744	Human performance concerns ASME Code	February 12, 2003

Drawings

D-214-004	Electrical Conduit and Tray Separation Criteria	Revision U
D-301-801	30" Pressure Relief Valve	Revision 0
D-301-832	30" Pressure Relief Valve Assembly Division 1	Revision 0
D-301-833	Testable Rupture Disc Latch Assembly	Revision 0
D-302-355	HPCS and Standby Diesel Generator Exhaust, Intake and Crankcase	Revision R
D-302-0621-00000	Emergency Closed Cooling System	Revision HH

Other Documents

	Failure Analysis Report Diesel Generator Rupture Disc	November 22, 2000
J1103754SRLR	Supplemental Reload Licensing Report for Perry Nuclear Power Plant Unit 1 Reload 8, Cycle 9	Revision 1
CERF 1722	Equivalency Evaluation for Replacement of Motor Feed Pump Impellers and Diffuser to Reduce Vibration	Revision 0
VLI-P42	Emergency Closed Cooling System	Revision 6
ARI-H13-P601-20	RHR A	Revision 5
ARI-H14-P601-17	RHR B and C	Revision 5
PTI-P42-P0010	ECC A Loop Total Leakage Verification	Revision 2
1-99-1089	Time Delay for Short Time Function must be changed from Max to Intermediate for proper coordination	September 9, 1999

0-00-1001	Establish same CFA setting that has been approved for Div 1 & 2 Battery Chargers	July 27, 2000
1-00-1039	100-PY-B(Y) Startup Supply to Bus L20 Brkr L1001 Protection Change Pickup Setting from 5400A to 6000A	October 18, 2000
SCR 2-00-1002	Change Pickup Setting from 5400A (9A tap). Change Time Dial setting from 7 to 4.	October 19, 2000
SCR 1-01-0039	Change to Improve the breaker settings to eliminate nuisance trips.	September 7, 2001
Equivalent Change 00-8062	Replace the existing obsolete flow instrumentation 1P45N0270 and 1P45N0271 instrumentation with newer model instruments.	July 17, 2000

Modifications

99-5003	Add a 4- 5 Minute Time Delay to Main & Reactor Feed Pump Turbines For any RCIC Initiations	February 12, 2001
99-8055	Replace DIV 1 DG Sonic Level Instrumentation with Capacitance Sensing Instrumentation	November 12, 2001
00-5004	Reconfigure the Stop/Auto/Start Control Switches	June 5, 2000
00-5018	Replace The Existing Magnetrol Level Switches with Similar Magnetrol Level Switches with Narrower Ranges to Provide Nominal System Leakage Rate of 2.77 GPH and Maintaining the 7 day Supply of Water	Revision 1
00-6009	Replace the existing hinged disc with a metal rupture disc that will burst when exposed to back pressure for the division 3 testable rupture disc	August 22, 2000
00-8092	Repair ESW elbow leak at 3 O'clock position looking east at the ECC B heat exchanger ESW piping	November 7, 2000
01-8043	Install a 1-1/2 " 3000#, SA 105 half-coupling	October 29, 2001
02-0254	Relocate the 1P45K0027 relay from 1H51P0805, Compt. D (F1G04-D) for hydrometer 1P45F0160	November 20, 2002

PRDC-0005	Replacement of Topaz Inverter and SOLA Power Supply Units Resulting in Change to Division 2 battery sizing calculation	June 1, 2000
PRDC-0014	Replacement of Topaz Inverter and SOLA Power Supply Units Resulting in Change to Division 1 battery sizing calculation	June 1, 2000
<u>Purchase Orders</u>		
7046059	Class III piping components	November 6, 2000
7056588	Inconel X-750 valve spring	March 12, 2001
<u>Procedures</u>		
NOP-CC-2003	Engineering Changes	Revision 1
NEI-0373	Processing Engineering Changes	Revision 6
PAP-1403	Control of Setpoints	Revision 9
NEI-0375	Equivalent Replacements	Revision 4
NEI-0420	Procurement Engineering	Revision 0
	Procurement Engineering Technical Evaluation for Parts Procurement/Inspection	Revision 2
NEI-0341	Calculations	Revision 9
NOP-CC-2004	Design Interface Reviews and Evaluations	Revision 0
PAP-1124	Pre-Maintenance and Post-Maintenance/modification Test Program	Revision 0
FTI-F0036	PMT Program Matrix	Revision 2
<u>Work Orders</u>		
00-009369	Implement a design change to install a half-coupling and plug per at risk SMRF 00-8092	November 7, 2000
00-001032	Diesel generator exhaust relief device 30 inch	October 25, 2000

Supplemental Information Request

Document Type	Number(s)	Requesting Inspector
3 Modifications (see Note 1)	00-5004, 99-5003, 99-8055	R. Winter
4 Modifications (see Note 1)	00-8062, 01-8040, 02-0020, 02-0254	R. Daley
1 Modifications (see Note 1)	00-5018	J. Ellegood
2 Modifications (see Note 1)	00-8092, 00-6009	M. Holmberg
Setpoint Changes	0-00-1001, 1-00-1039,1-99-1089	B. Winter
2 Setpoint Changes	1-01-0039, 2-00-1002	R. Daley
Equivalency Evaluation	CERF 01722	J. Ellegood
Commercial Grade Dedication	PO. Number 7056588 S/N Number 91132595	M. Holmberg
Calculations Supporting Modifications	R45-T04	R. Winter
2 Calculations Supporting Modifications	PRDC-0005 R/4 DCC-5 PRDC-0014 R/0 DCC-1	R. Daley
Calculations Supporting Modifications	All calculations supporting modification 00-5018.	J. Ellegood
Calculations Supporting Modification	All calculations supporting modifications 00-6009 and 00-8092.	M. Holmberg
Condition Reports (mod related)	00-2216, 01-2620, 01-3776, 02-00723	R. Winter
Condition Report (mod related)	01-2184, 02-03604	R. Daley
Condition Reports (mod related)	01-2949, 02-01465	J. Ellegood
Condition Reports (mod related)	01-0230, 02-03389, 02-04210	M. Holmberg

Document Type	Number(s)	Requesting Inspector
2 Safety Evaluations	01-0026, 01-0036	R. Winter
1 Safety Evaluation	00-0089	J. Ellegood
2 Safety Evaluations	00-0014, 00-0021	R. Daley
3 Safety Evaluations	01-001, 01-007, 01-018	M. Holmberg
3 safety evaluation screenings	02-01210, 02-00453, 02-00940	R. Winter
1 RAD and 2 safety evaluation screenings	RAD 02-00055, Screens 01-00294, 01-00384	R. Daley
3 safety evaluation screenings and 1 RAD	02-00530; 02-00569; 02-01301; RAD 02-00436	J. Ellegood
2 safety evaluation screenings and 3 RADs	Screening 02-00384 associated with UFSAR change and CR 02-034. Screening for Modifications 01-8034 and 02-0237. RADs 02-01799, 02-01669, 02-01617	M. Holmberg
3 Condition Reports (related to 50.59 evaluation)	02-00272, 01-3683, 01-3747	R. Winter
2 Condition Reports (related to 50.59 evaluation)	01-3269, 01-4365	R. Daley
3 Condition Reports (related to 50.59 evaluation)	02-01422; 01-01332; 02-00272	J. Ellegood
3 Condition Reports (related to 50.59 evaluation)	02-03487, 02-01332, 02-00250	M. Holmberg
Special Test	TXI-0321-002 Noble Chemical Metals Addition	M. Holmberg

Note 1 - Copy of the Modification Package to include; document describing the need for the modification, condition reports associated with the modification or installation of the modification, description of the modification, the safety evaluation or screening, material lists and procurement records, completed modification acceptance tests, procedures and drawings affected/changed due to modification, records confirming applicable environmental qualification (e.g. steam, fire, flooding, seismic) of modified components and applicable UFSAR Sections changed due to the modification. Also, provide a copy of the vendor technical manual associated with the modified equipment.

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
HPCS	High Pressure Core Spray
IMC	Inspection Manual Chapter
LOCA	Loss of Coolant Accident
NMCA	Noble Metal Chemical Addition
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PCT	Peak Clad Temperature
SDP	Significance Determination Process
UFSAR	Updated Final Safety Analysis Report