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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
Reply to a Notice of Violation

Ladies and Gentlemen:

Enclosed is the Perry Nuclear Power Plant staff's reply to the Notice of Violation contained in NRC Inspection Report 50-440/97002, which was transmitted by letter dated April 23, 1997. The Notice of Violation involves a safety evaluation that did not provide an adequate basis for the determination that the change did not increase the probability of a malfunction of equipment or introduce the potential for a malfunction of a different type than any previously evaluated in the Updated Safety Analysis Report.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs at (216) 280-5606.

Very truly yours,

Richard [illegible] for L.W. Myers

for Lew W. Myers

DTG:cw

Enclosure

IE01 1/1

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager

REPLY TO A NOTICE OF VIOLATION

Violation 97002-02

Restatement of the Violation

10 CFR 50.59 requires, in part, that records of changes to the facility include a written safety evaluation which provides the bases for the determination that the change did not increase the probability of a malfunction of equipment or introduce the potential for a malfunction of a different type than any evaluated previously.

Contrary to the above, Safety Evaluation No. 97-001, written on December 6, 1996, to support the change to the facility controlled by Design Change Package 94-0027, Revision 6, did not adequately support your conclusion that the probability of a malfunction was not increased or that the potential for a different malfunction was not introduced.

Issue

In NRC Inspection Report No. 50-440/97-002, the NRC concluded that a violation of 10 CFR 50.59, "Changes, tests and experiments," occurred because the complete basis to support the determination that a design change did not involve an Unreviewed Safety Question (USQ) was not included in the safety evaluation. Safety Evaluation 97-001, prepared on December 6, 1996, and approved on January 17, 1997, was written to support Design Change Package (DCP) 94-0027, which provided for the installation of bypass lines around each Emergency Closed Cooling (ECC) heat exchanger. The bypass lines utilize a three-way, electro-hydraulic modulating Temperature Control Valve (TCV) to divert flow around the heat exchangers during low lake water temperature conditions. Because the TCV is an active component, and the associated bypass line did not previously exist in the system, failure of the TCV could result in a total bypass of the heat exchanger. This possibility should have been explicitly discussed and shown not to be different from malfunctions previously evaluated in the Updated Safety Analysis Report.

Reason for the Violation

The apparent cause of this inadequate safety evaluation is the failure by the preparer and reviewer to ensure the appropriate and relevant details of the modification were documented in the safety evaluation during its development. Because of their engineering expertise and plant specific knowledge, these engineers inappropriately bounded the failure modes of the modification by reliance on the design report and their understanding of the modification, rather than by an explicit justification in the answers to the safety evaluation questions. This resulted in the conclusion that no USQ existed without the basis sufficiently documented in the safety evaluation, as required by Plant Administrative Procedure (PAP)-0305, "Safety Evaluations."

Additionally, the site organization was generally too receptive to using implicit or understood information to provide the basis for the safety evaluation. The design report information that supports modification approval was reviewed along with the safety evaluation. Because some of the information in the design report supports the conclusion that a USQ does not exist, and this information was not explicitly stated in the safety evaluation, the basis for the USQ determination was dependent on the design report. Therefore, a contributing cause of the inadequate safety evaluation was an acceptance of the design report to provide the basis for the USQ determination.

The safety evaluation review and approval process did not identify that this discrepancy existed. The supervision and management involved in the process, including the Plant Operational Review Committee (PORC) and the Safety Evaluation Review Subcommittee (SERS) of the Corporate Nuclear Review Board (CNRB), were surveyed and clearly understood the evaluation results based on their plant specific knowledge of the system's design and operational history. Therefore, oversight by the review and approval process was a contributing factor in missing further opportunities for self identification of the safety evaluation's deficiency.

Corrective Steps Taken and Results Achieved

Safety Evaluation 97-001 is being re-evaluated to determine whether a USQ exists. As a part of the review and approval process, members of the engineering management team, members of PORC, alternates, and the SERS chairman are being made aware of this discrepancy and that heightened awareness should be maintained to ensure that the conclusions reached in safety evaluations are sufficiently documented within the answers to the safety evaluation questions.

Corrective Steps That Will Be Taken To Avoid Further Violations

The 10 CFR 50.59 Training Module will now include this event as a site specific example of a 10 CFR 50.59 violation. The emphasis of this addition to the training module will be to stress the importance of developing safety evaluations that are adequate without use of implicit or understood information, to ensure that the conclusions reached are sufficiently documented within the answers to the safety evaluation questions.

A comprehensive self-assessment will be performed on the 10 CFR 50.59 safety evaluation process to identify areas of improvement to enhance and improve the implementation or quality of the program, as appropriate. Additionally, the assessment will include a review for further enhancements where NRC expectations are communicated to the industry as additional NRC guidance becomes available.

Date When Full Compliance Will Be Achieved

Full compliance will be achieved by June 6, 1997, with the approval of the revision of Safety Evaluation 97-001.

The following table identifies those actions which are considered to be regulatory commitments. Any other actions discussed in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

Commitment
By June 6, 1997, Safety Evaluation 97-001 will be revised, PORC reviewed, and approved by plant management. By December 5, 1997, a comprehensive self-assessment will be completed on the 10 CFR 50.59 safety evaluation process to identify areas of improvement to enhance and improve the implementation or quality of the program, as appropriate. By the next training session, 10 CFR 50.59 Training Module will include this event as a site specific example of a 10 CFR 50.59 violation.