U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-315/88007(DRS); 50-316/88008(DRS)

Docket Nos. 50-315; 50-316 Licenses No. DPR-58; DPR-74

Licensee: American Electric Power Service Corporation
Indiana and Michigan Power Company
1 Riverside Plaza
Columbus, OH 43216

Facility Name: D. C. Cook Nuclear Plant, Units 1 and 2

Inspection At: D. C. Cook Site, Bridgman, Michigan
American Electric Power Service Corporation
Columbus, Ohio

Inspection Conducted: February 8-12, 1988 (Columbus)
February 22-26, 1988 (Bridgman)

Inspectors: R. A. Hasse 3/21/88
Lead Date

R. N. Sutphin 3/21/88
Date

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Date

Approved By: Monte P. Phillips, Chief 3/21/88
Operational Programs Section Date

Inspection Summary

Inspection on February 8-26, 1988 (Reports No. 50-315/88007(DRS); No. 50-316/88008(DRS))

Areas Inspected: This special safety inspection was conducted to determine the adequacy of the licensee's 10 CFR 50.59 safety evaluations. This was an announced inspection conducted under IE Module 92702.

Results: The licensee's safety evaluations were judged to be adequate. The use of a dedicated group to perform these evaluations was considered a major strength. This resulted in generally comprehensive safety evaluations. One weakness noted was the failure to directly address the three criteria defining an unreviewed safety question in the safety evaluation documentation. This led to some doubts in certain instances as to whether or not these criteria had been thoroughly addressed (Paragraphs 2.b.(2).(a), 2.b.(3) and 2.b.(5)). No violations or deviations were identified.

DETAILS

1. Persons Contacted

American Electric Power Services Corporation (AEPSC)

M. Alexich, Vice President, Nuclear Operations
R. Kroeger, Manager, Quality Assurance
**P. Barrett, Manager, Nuclear Safety and Licensing
V. Lepore, Manager, Design Division

Indiana and Michigan Electric Company

*W. Smith, Jr., Plant Manager
*T. Beilman, I&C Superintendent
*T. Postlewait, Superintendent, Technical Engineering
*P. Wycoff, Assistant Supervisor, I&C Production Control

US NRC

*B. Jorgensen, Senior Resident Inspector
J. Heller, Resident Inspector

*Denotes those attending the exit meeting on February 26, 1988.

**Denotes those participation in the exit meeting by telephone.

The inspectors also contacted other licensee personnel as a matter of routine during the inspection.

2. 10 CFR 50.59 Safety Evaluations

a. Purpose and Scope

The purpose of this inspection was to determine if the safety evaluations performed by the licensee pursuant to 10 CFR 50.59 were adequate to identify any unreviewed safety question or required changes to the technical specifications, and if the bases for conclusions were adequately documented. Since 10 CFR 50.59 permits the licensee to make changes to the plant within stated constraints without prior NRC approval, the adequacy of the safety evaluations is essential to maintaining the risk level of the plant. Fundamentally, the adequacy of this process is essential to assuring that the level of public risk is not inadvertently increased.

During this inspection the inspectors reviewed the program for conducting safety evaluations and a sample of the safety evaluations performed for permanent modifications, temporary modifications, operating procedure changes, and a special test.

b. <u>Inspection Results</u>

(1) <u>Program</u>

The conduct of these safety evaluations was governed by two procedures, NS&L QP-7, "Safety Review of Design Changes," and PMI-1040, "Plant Nuclear Safety Review Committee." Salient features of the program were as follows:

- All safety evaluations were being performed by or approved by the Nuclear Safety and Licensing Section (NS&L) of AEPSC. Provisions did exist in PMI-1040 for safety evaluations to be performed by plant personnel; however, this will not be generally implemented until these personnel are trained in this function.

- Personnel at the plant site performing applicability checks (to determine if a safety evaluation is required) received formal training for performing this function. The applicability checks were reviewed by the Plant Nuclear Safety Review Committee (onsite review committee).

- Both documents gave extensive guidance on performing the safety evaluations. The guidance given in NS&L QP-7 did not focus on the basic 10 CFR 50.59 criteria for determining if an unreviewed safety question existed (i.e., probability and consequences of previously analysed accidents, new accidents, and margin of safety of technical specifications). Rather, it focused on fundamental design issues (e.g., single failures, potential common mode failures, seismic qualification, etc.). These issues were then matrixed to provide the basic response to the three criteria which determine whether an unreviewed safety question existed. No explicit guidance was provided in NS&L QP-7 for determining if the margin of safety for a technical specification was reduced. The guidance given in PMI-1040 was focused on the 10 CFR 50.59 criteria; however, the guidance given on determining whether the margin of safety for a technical specification is reduced was marginal, and did not discuss the bases for technical specifications.

- Both the PNSRC and the Nuclear Safety and Design Review Committee (NSDRC) reviewed all safety evaluations.

- The safety evaluations were documented in a Safety Review Memorandum (SRM) issued by the NS&L. These memoranda were generally good analyses of the safety issues attendant to the change as well as any licensing issues involved. As with the procedural guidance noted above, the SRMs neither focus on nor directly addressed the three 10 CFR 50.59 criteria.

The program provided a generally adequate structure for performing safety evaluations and applicability checks. The use of a single organization for performing the safety evaluations was considered a strong point since it engendered an in-depth approach to the task. Licensee management should explore means for addressing the 10 CFR 50.59 criteria more explicitly in the SRMs.

(2) Permanent Modifications - Request For Change (RFC)

Permanent plant modifications involving safety-related systems, safety-interface systems (having safety impact), or requiring significant multi-disciplinary input were controlled by the RFC system. RFCs were controlled by AEPSC. The inspectors reviewed 32 RFC packages. No violations or deviations were identified. One RFC warranted some discussion and one other identified an open item.

(a) RFC-12-2651, "Modification to Safety Injection (SI) Minimum Flow Lines"

A portion of this modification increased the minimum flow (bypass flow) of the SI system and as a result reduced the SI flow into the reactor core. The licensee requested the NSSS to perform a safety analysis for the modification. Two pertinent conclusions were included in the analysis: (1) the peak cladding temperature (PCT) reached during a small break LOCA (SBLOCA) would increase by 87°F; (2) the PCT reached (approximately 1750°F) during a SBLOCA was below the Safety Limit of 2200°F and was bounded by the PCT reached during a large break LOCA (LBLOCA). Based, in part, on these conclusions the SRM issued for the modification concluded that no unreviewed safety question existed; however, because of a required technical specification (TS) change, (4.5.2.h) not related to injection flow, the NSSS safety evaluation was submitted to the NRC as supporting documentation. The change was subsequently approved by the NRC and an SER was issued which did address the reduced SI flow.

The inspector was concerned with the SRM conclusion that no unreviewed safety question existed. The SRM, while complete in other respects, did not address potential impact of the reduced SI flow on the margin of safety of technical specifications. In particular, the basis for TS 3.2.5 (DNB parameters) states that this LCO is established to protect the design DNBR under certain plant transients and accidents. The increase in PCT during a SBLOCA may have impacted this margin.

A second concern involved the statement in the NSSS safety evaluation that while the PCT reached during a SBLOCA increased, it was bounded by the PCT reached during a LBLOCA. The inspector questioned the pertinence of this comparison.

Licensee management should review these items and assure that they are considered in future safety evaluations.

(b) RFC-12-2665, "Chemical and Volume Control System Cross-Tie"

During the early field work on this modification a Condition Report (12-04-85-792) had been written concerning a problem with the pipe welding. The 316 SS pipe had been welded with ER 308L weld wire rather than ER 316 weld wire specified in the weld procedure. The condition report had been closed with corrective action to preclude recurrence of the problem; however, the RFC package did not contain the identification and disposition of those welds completed with the 308L weld wire. This is considered an open item pending licensee identification and disposition of these welds (315/88007-01; 316/88008-01).

(3) Permanent Modifications - Plant Modifications (PM)

Plant Modifications, by definition, are non-safety-related and non-safety-interface and would typically not require a safety evaluation; however, applicability checks are performed for each PM. Setpoint changes are also handled as PMs and may require a safety evaluation.

The inspectors reviewed 17 PMs. No violations or deviations were identified; however, the safety evaluation for one set point change warrants some discussion.

The alarm setpoint for the Unit 1 reactor coolant pump stator winding temperature was increased from 120°C to 135°C under plant modification 01-PM-586. Apparently, three of the four RCPs had been alarming due to higher ambient temperatures which in turn, caused higher stator temperatures. The engineering analysis performed by AEPSC in support of this change noted that the stator winding insulation was rated at 155°C maximum and the 135°C setpoint left margin to the rated temperature. It also contained a discussion of action points (temperature at which action should be taken to reduce stator temperature, temperature at which the pump should be tripped, etc.). The inspector considered the engineering evaluation to be quite adequate and supportive of the setpoint change.

The safety evaluation (SRM) issued for this modification atypically provided very little basis for concluding that no unreviewed safety question existed. The loss of a RCP is an analyzed accident in the FASR and is obviously important to safety. Operating equipment closer to design limits has the potential of increasing the probability that the equipment will fail in service. This issue was not addressed in the SRM, i.e., the potential increase in the probability of failure of equipment important to safety. The inspector concluded, based on the engineering evaluation, that any increase in the probability of RCP failure was minimal at worst; however, the failure to address the issue in the SRM raises the question of whether it was actually considered. This is a good example of the hazard of not explicitly addressing the three 10 CFR 50.59 criteria in the SRM (see Paragraph 2.b.(1)).

(4) Temporary Modifications (TM)

The inspectors reviewed approximately 100 TMs (open and closed). While no violations or deviations were identified, there were several items of concern and one open item.

(a) TM - 19, "Remove Pressurizer Heater From Service"

Two items of concern where identified with this TM. Since 1981, approximately 92 Kw of pressurizer heating capacity had been removed from service. FSAR Table 4.1-4 lists the total heater capacity as 1800 Kw and the capacity required by the Technical Specification is 150 Kw. Thus, considerable margin still exists. The rationale for removing these heaters under a TM is that the capacity is planned to be restored but on a long-term basis.

The first concern involved a 10 CFR 50.59 applicability check performed in support of this TM. A safety evaluation had been properly performed in 1986 concluding that no unreviewed safety question existed (the inspectors agreed with this conclusion).

In 1987, as a result of an upgrade effort for the TM program, applicability checks and safety evaluations were redone for all open TMs. The applicability check completed in 1987 concluded that no safety evaluation for this TM was required. The basis given was that while it changed something described in the FSAR, the change was not safety significant (supported by calculations similar to those used in the original safety evaluation). The inspectors were concerned with the preparer's apparent misunderstanding of the purpose of the applicability check vs. a safety evaluation (as well as the fact that the PNSRC approved the applicability check). Subsequent to, but not as a result of this instance, formal training had been provided to individuals performing applicability checks. This instance underscores the need for careful management oversight and quality verification of 10 CFR 50.59 activities (see Paragraph 2.b.(7)).

The second concern with this TM was more generic. The TM placed the plant in a condition different than that described in the FSAR for a relatively long period of time (though not permanently). Frequently, TMs supporting other activities (e.g., maintenance) are not included in the licensee's annual 10 CFR 50.59 report; however, long-term TMs or those representing fixes should be included. This was discussed with licensee management and they agreed to address this issue in the next revision of the TM procedure (currently in progress) and include TM-19 and other such TMs in the next annual report.

(b) <u>TM - 152</u>

Temporary Modification No. 152 provided for the installation of a non-Class 1E motor on the valve operator for valve 1-WMO-754. The original design called for a Class 1E motor. The non-class 1E motor was of the identical horsepower, frame, current, and thrust of the original equipment design; however, it was supplied as a QA-S motor for use as a spare in Balance of Plant applications, rather than as a QA-N motor suitable for use in Class 1E systems.

The 1-WMO-754 valve, which is in series with a manual valve (ESW-109), can be opened to provide an alternate flow path from the essential service water (ESW) system to the suction side of the east motor-driven auxiliary feedwater pump. Both valves are normally closed, and the ESW system is used as an alternate feedwater supply source only after the preferred source, the Condensate Storage Tank (CST), is exhausted.

A 10 CFR 50.59 evaluation was performed and documented on September 11, 1987, for this TM, and it was concluded that the use of the non-Class 1E motor in the operator of the valve 1-MWO-754 did not constitute an unreviewed safety question as long as there was on duty a dedicated operator familiar with the auxiliary feedwater system during the reactor operational modes (Modes 1, 2, and 3) requiring an operable motor-driven auxiliary feedwater pump. The operator's function would be to manually open valve 1-MWO-754 as well as ESW-109 if the valve motor operator should fail to function. It was judged that this would not significantly increase the time to supply essential service water to the steam generators because of the necessity of having an operator manually opening valve ESW-109, whether the motor-operated valve functioned or not. A notice to that effect was to be posted at the location of this MWO valve control, in the control room. The inspector verified that the notice (No. 2618) was posted as required.

Licensee plans for closeout of this TM were outlined in a letter prepared by AEPSC. The preferred option was to have the installed motor upgraded to Class 1E and the vendor had been contacted to explore this possibility. Failing this, the purchase of a Class 1E motor would be pursued.

The inspectors concluded that the alternative actions taken in this case to mitigate the increased probability of MOV failure and resultant inability to access the ESW for long term cooling was acceptable on a temporary basis, and the safety evaluation was acceptable. In general, mitigative action would not be acceptable since it can change the mode of plant operation, introduce new failure modes and interfaces, and potentially place the plant in an unanalyzed situation. The timely closeout of this TM is considered an open item (315/88007-02).

(5) <u>Special Test</u>

The inspectors reviewed the safety evaluation for one special test, 1 THP SP 141, "Unit 1 Turbine Volumetric Flow Test." Performance of this test required lowering the RCS Tave by as much as 20°F below the normal operating Tave. The licensee requested the NSSS to perform a safety evaluation in support of the test. The results of the safety evaluation indicated the following impacts on the FSAR safety and engineering evaluations:

- The PCT resulting from a LBLOCA increased from 2154°F to 2198.2°F (Safety Limit is 2200°F).
- Short term effects of RCS blowdown are increased due to an increase in mass and energy release rate.
- Peak pressure in the reactor cavity following a cold leg break are increased.
- The peak pressure in the pressurizer enclosure following a double-ended shear of the four inch pressurizer spray line increases from 13.9 Psi to 14.54 Psi (design limit is 80 Psi).

- Peak pressure against the containment shell due to a hot leg break increases from 14.1 Psi to 14.4 Psi (design is 16.6 Psi).

- Peak pressure against the containment shell due to a cold leg rupture increases from 12.1 Psi to 12.35 Psi (design is 16.6 Psi).

- LOCA forces on reactor internals increase by 14 percent.

The evaluation concluded that no unreviewed safety question existed, presumably since no design or safety limit would be exceeded during the analyzed events. The inspectors had some concern with this conclusion because of the decreases in margins, in particular the increase in PCT and its proximity to the safety limit. The licensee should carefully consider such decreases in margin for potential impact on the probability of failure of the fission product boundaries or equipment important to safety during analyzed events. Again, this might have been addressed in the SRM had the three criteria been addressed directly.

(6) Procedure Changes

The inspectors reviewed changes to four operating and emergency operation procedures. No violations or deviations were identified.

(7) Quality Verification

The inspectors reviewed recent QA audits of the 10 CFR 50.59 process. The audits contained no assessments of the quality or technical accuracy of this effort. A discussion with the chairman of the NSDRC indicated that while they did assess the thoroughness of the safety evaluations, they relied on the verification process (e.g., design verification) to assure the technical accuracy. The inspectors identified no generic adequacy problems (but see Paragraph 2.4.(a)); however, this inspection did not include a specific assessment of technical accuracy.

3. Open Items

Open items are matters which have been discussed with the licensee which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Paragraphs 2.b.(2).(b), and 2.b.(4).(b).

4. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on February 26, 1988, and summarized the purpose, scope, and findings of the inspection. The licensee stated that the inspectors had no access to proprietary information.