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December 31, 1997 AEP:NRC:1260G5

Docket Nos.: 50-315
50-316

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Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION RELATED
TO 10 CFR 50.59 PROGRAM QUESTIONS RAISED DURING THE
DECEMBER 22, 1997 PUBLIC MEETING

The purpose of this letter is to provide additional information in response to requests made during the December 22, 1997, public meeting held in Lisle, Illinois, related to our 10 CFR 50.59 program. In our followup letter AEP:NRC:1260G4, dated December 24, 1997, we docketed our commitment to review design changes, procedure changes, and 10 CFR 50.59 screenings for the types of problems identified during the architect engineering (AE) design inspection. Specifically, the request was that we review our 10 CFR 50.59 screenings and evaluations performed on procedures and design changes to evaluate the adequacy of this process.

A self-evaluation was performed to verify the integrity of our 10 CFR 50.59 program. This self-evaluation looked at a sample of recent (January 1996 thru September 1997) Cook Nuclear Plant 10 CFR 50.59 screenings and safety evaluations in light of insights gained during the AE design inspection. Self-evaluation review team members were provided refresher training on the new plant directive related to design bases, licensing bases, and single failure. Each 10 CFR 50.59 review was then performed against this new directive to ascertain the adequacy of the screening and/or safety evaluation. Based on the 10 CFR 50.59 screenings and safety evaluations sampled:

- there are no new operability issues due to the safety evaluation and screening process,
- there are no improperly evaluated 10 CFR 50.59 screenings or safety evaluations that have not been previously identified with corrective actions taken, and
- the screenings and safety evaluations have been properly updated in the updated final safety analysis report (UFSAR) as required.

In addition to the self-evaluation, five safety related design changes, approved during the period that the recirculation sump cover vent hole modification was made (i.e., 1979), were reviewed to determine if design basis attributes were properly maintained during the design change process. As with the 10 CFR 50.59 program self-evaluation, review teams were established. In addition to

reviewing the five design changes in light of the lessons learned from the AE design inspection, each change was evaluated against the following:

1. did the design change meet its functional objectives,

2. was the design change installed in accordance with the intended design,

3. were the appopriate drawings identified and revised to document the as-installed configuration,

4. was anything found that resulted in the design change being outside the design bases, and

5. did the design change adversely impact the operability of any safety related systems, structures or components?

The results of the design change reviews concluded that no instances were found where the operability of safety related systems, structures or components were adversely affected.

Sincerely,

A. Alan Blind for

E. E. Fitzpatrick
Vice President

/vlb

Attachments

c: A. A. Blind
A. B. Beach
MDEQ - DW & RPD
NRC Resident Inspector
J. A. Abramson

ATTACHMENT 1 TO AEP:NRC:1260G5

REVIEW OF 10 CFR 50.59 SCREENINGS AND EVALUATIONS

SELF-EVALUATION OF RECENT COOK NUCLEAR PLANT 10 CFR 50.59 SCREENINGS AND SAFETY EVALUATIONS

INTRODUCTION

During the recent design inspection conducted by NRC, concerns were raised regarding the understanding of design and licensing bases, and the adequacy of the 10 CFR 50.59 program. As a result of these concerns, a commitment was made on December 22, 1997, to perform a review of recent 10 CFR 50.59 products and to report findings to NRC. This attachment describes the review process and provides the results of the self-evaluation of recent 10 CFR 50.59 safety screenings and evaluations. With three exceptions, the safety screenings and evaluations examined were performed from January 1996 to September 1997. Three of the safety evaluations were performed earlier than this time period, although they were selected because they were installed during a time period of interest. Systems included in the review scope were safety related systems, although the residual heat removal (RHR) and component cooling water (CCW) systems were excluded due to their coverage during the design inspection.

A team of fifteen individuals was selected and received refresher training prior to commencing the self-evaluation. To set the proper perspective for the evaluation, training was performed on Directive 800000-DIR-2300-04, "Application and Use of Design Bases, Single Failure Criteria, Engineering Design Bases, and Current Licensing Basis". This procedure incorporated lessons learned from the design inspection with respect to design and licensing bases and single failure criteria.

The primary focus of the evaluation was to determine, based on these lessons learned, if:

- there were any open operability issues due to the safety evaluation and screening process,
- there were any improperly dispositioned 10 CFR 50.59 screenings or safety evaluations that had not been previously identified and corrective actions taken, and
- the screenings and safety evaluations have been properly incorporated into the UFSAR as required.

The conclusion of the evaluation is that no open operability or improper evaluations were discovered. Also, all safety evaluations which required update of the UFSAR had initiated or completed the required UFSAR update. The conclusions are addressed in more detail later in this attachment.

In order to facilitate this evaluation, the following populations of 10 CFR 50.59 screenings and safety evaluations were selected:

- 35 procedural 10 CFR 50.59 screenings, consisting of both change sheets and revisions,
- 11 procedural 10 CFR 50.59 safety evaluations, and
- 25 design change 10 CFR 50.59 safety evaluations and 24 10 CFR 50.59 screenings that correspond to them.

One of the design change safety evaluations was performed prior to January 1996 and the screening was not included with the safety evaluation. This yielded a population of 71 10 CFR 50.59 screenings and safety evaluations to evaluate. The methods for selecting each of the populations of documents are discussed in detail as follows.

SELECTION CRITERIA FOR 10 CFR 50.59 SELF-EVALUATION

Procedural Screening Sheets

Approximately 1180 procedures had change sheets and 463 procedures had revisions for the January 1996 through September 1997 period. These numbers do not reflect actual numbers of issued change sheets and revisions since only the revision in effect is listed. Also, when a procedure is revised, change sheets on the previous revision are subsumed and do not appear on this list.

The sample was obtained by selecting approximately every 230^{th} change sheet to obtain a sample of 4, and approximately every 13^{th} revision to obtain a sample of 31. If this random sample method revealed a revision on a non-safety related system, the nearest safety-related revision was chosen.

Procedural Safety Evaluations

In the case of the procedural safety evaluations, the files for all safety evaluations between 1/1/96 and 9/30/97 were assembled. Approximately every 3^{rd} safety evaluation relating to a procedure and not involving RHR or CCW was chosen. This resulted in about every 8 to 10^{th} document in the safety review files being selected for evaluation.

Design Change Evaluations

A sample of fifteen design change proposals (DCPs) and five temporary modifications (TMs) was randomly selected from the design change population, to ensure a minimum of twenty evaluations for review. In addition, five safety evaluations, corresponding to different revisions of some of the design changes and temporary modifications, were reviewed, thus making a total of 25 safety evaluations.

TEAM SELECTION AND EVALUATION PROCESS

Three teams of qualified individuals were formed to perform the self-evaluation. At the start, two teams evaluated procedural 10 CFR 50.59 issues and one team examined screenings and safety evaluations associated with design changes. After completing the procedural reviews, one Nuclear Safety 10 CFR 50.59 reviewer and one lead auditor supported the design change review effort. The teams were composed as follows:

Procedure Review Team #1	Procedure Review Team #2	Design Change Review Team
QA Lead Auditor	QA Lead Auditor	QA Lead Auditor
Nuclear Safety 10 CFR 50.59 Reviewer	Nuclear Safety 10 CFR 50.59 Reviewer	Nuclear Safety 10 CFR 50.59 Reviewer
Operations Procedure Specialist	Operations Procedure Specialist	Design Engineers (Civil, Electrical, Mechanical, I&C As Needed)

All Nuclear Safety and design engineering personnel involved in this effort were qualified 10 CFR 50.59 reviewers, and procedure specialists were qualified 10 CFR 50.59 screeners. Each review was performed considering the lessons learned from the design inspection regarding single failure and design bases as well as a checklist that was developed to document the adequacy of the screenings and safety evaluations. A sample of the checklist used for this review is attached to this document. A checklist was filled out for each screening and safety evaluation and was signed by the QA lead auditor, Nuclear Safety reviewer and appropriate subject matter expert. Comments from each reviewer were annotated on these sheets and collated after completion of the effort. The Nuclear Safety and design engineering personnel performed independent reviews of screenings and safety evaluations and were available to the lead auditors and others to answer questions regarding the specifics of the design or procedure change. The operations procedure specialists and design engineers served as subject matter experts in this regard.

EVALUATION RESULTS

Screenings

With respect to the screenings, 11 of 60 screenings were found to contain justifications for the "yes" or "no" answers that would be considered less than adequate in light of lessons learned from the design inspection. These inadequacies included things such as:

- not specifically referencing all applicable UFSAR sections or Technical Specifications by number,
- change descriptions that were in less detail than expected today, and
- justifications for a "yes" or "no" which would not be considered detailed enough in light of lessons learned from the design inspection.

The teams, however, concluded that all screenings had been properly performed, had the correct conclusion with respect to whether or not a full safety evaluation was required, and did not represent any operability concerns. Concerns identified relate to desired level of detail and completeness of documentation. To address these shortcomings in future 10 CFR 50.59 screenings, lessons learned will be shared with 10 CFR 10 CFR 50.59 screeners and safety reviewers for training purposes. Also, issues identified in this self-evaluation will be integrated into future regular initial and requalification training sessions for 10 CFR 10 CFR 50.59.

Safety Evaluations

With respect to the safety evaluations, one of 36 safety evaluations did not contain justification that would be considered adequately in light of lessons learned from the design inspection. However, this evaluation was part of a design change that utilized two other supporting safety evaluations for installing the design change and these safety evaluations contained the justification missing from the design change safety evaluation.

Two safety evaluations associated with design changes 12-DCP-0049, Rev. 0 and Rev. 1, "Engineered Safeguards (AES) Filtration System Bypass Damper Replacement" were not properly evaluated, did not identify single failure issues and posed operability concerns. However, condition report CR 97-2491 and LER 97-023 had previously been written to address these issues. This condition report and LER were written prior to this self-evaluation and corrective action has already been taken in these cases. This does not represent a new operability concern or a concern that has not already been identified and corrective action taken.

Safety evaluations which would have required UFSAR update properly identified the need to update the UFSAR in the safety evaluation. Also, it was verified that the proper actions have been taken to initiate an update of the UFSAR in these cases.

No new or previously unidentified operability issues were identified in this self-evaluation. Corrective action has already been taken for a previous issue that was identified in the self-evaluation. All safety evaluations properly identified the need to update the licensing basis when required. With the exception of the two previously identified design change revisions for which corrective action has already been initiated, safety evaluations were properly dispositioned and do not represent unreviewed safety questions. The results of this self-evaluation concluded the operability of the plant and the integrity of the design and licensing basis remains intact with respect to our safety evaluations.

CONCLUSION

This self-evaluation process has considered the lessons learned from the design inspection and demonstrated the integrity of the 10 CFR 50.59 process at Cook Nuclear Plant. The evaluation found no outstanding operability or licensing basis concerns as a result of this 10 CFR 50.59 screening or safety evaluation process. Specifically, this self-evaluation has demonstrated that:

- there are no new operability issues due to the safety evaluation and screening process,
- there are no improperly dispositioned 10 CFR 50.59 screenings or safety evaluations that have not been previously identified and corrective action taken, and
- the screenings and safety evaluations have been properly incorporated in the UFSAR as required.

ATTACHMENT 2 TO AEP:NRC:1260G5
REVIEW OF SELECTED DESIGN CHANGES

REVIEW OF DESIGN CHANGES
APPROVED OCTOBER 6, 1978 - OCTOBER 6, 1979

Purpose

To review a sample of safety related design changes approved during the same period as "Modifications to the Containment Recirculation Sump" (RFC-2361). The criteria for these reviews are as follows:

1. did the design change meet its functional objectives,

2. was the design change installed in accordance with the intended design,

3. were the appropriate drawings identified and revised to document the as-installed configuration,

4. was anything found that resulted in the design change being outside the design bases, and

5. did the design change adversely impact the operability of any safety related systems, structures or components?

Results

The review found that the five design changes reviewed met their functional objectives, were installed in accordance with the intended design, and appropriate drawings were issued. No instances were found where the design change adversely affected the operability of any safety related system, structure or component.

A review of the design change index revealed that RFC-2361 was the only safety related design change performed by that lead engineer during the short time he was employed by American Electric Power. The individual was lead engineer for four other design changes, but they were non-safety related.

Scope

The approval date for RFC-2361 was April 6, 1979. The design changes selected were those having approval dates six months before and after RFC-2361. Ten safety related design changes were approved between October 6, 1978, and October 6, 1979. Five of these were randomly selected for review.

Summary of Reviews

RFC-DC-1-1508: Installation of Seismic Peak Recording Accelerometers

01-RFC-1508 replaced the seismic peak recording accelerometers (SPRA) which had been installed in Unit 1 during construction with a new, more reliable type. Initially, the new SPRAs were mounted at three test locations where they were exposed to heat, cold, radiation, and vibration. The test period of one calibration cycle was specified and, after successful evaluation, the new SPRAs were permanently installed at three permanent locations (CD Emergency Diesel Generator, Containment Spring Line, and Spent Fuel Pit). The design change did not require any special design by American Electric Power. The vendor provided documentation of seismic

performance certification for the SPRA and vendor mounting specifications were used for installation. The calibration procedure was revised to incorporate the new SPRAs.

The review of this design change concluded that the installation was performed as intended, met design bases requirements, and resulted in no nonconforming conditions or deviations. It also determined that the design change has had no adverse impact on operability of any safety-related systems, structures, or components. The review concluded that 01-RFC-1508 was properly handled in accordance with the requirements applicable at the time of installation.

RFC-DC-12-2213: Upgrade Performance of Current Circuits

RFC-DC-12-2213 upgrades the performance of current circuits. This includes replacement of existing cables with cables and parallel conductors to reduce the burden on current transformers (CT's) which are in the relay circuits for 4kV motors. This results in cables which have a higher ampacity and lower resistance. The function of the CT/relay circuits impacted by this design change was not changed. The implementation of this design change ensured the CTs would operate within the manufacturers specifications.

The design change package was reviewed to ensure the appropriate drawings were issued to support the design change and summary reports documented the installation completion. The review concluded that the design change was installed per the intended design and there was no deviation from our design bases as documented in the safety review memorandum. No significant nonconforming conditions were identified as a result of this technical review and there was no adverse impact on safety related systems, structures or components.

RFC-DC-12-2225: Reactor Coolant Pump (RCP) Motor Oil Spillage Protection and Collection System

The purpose of 12-RFC-2225 was to install an additional reactor coolant pump motor oil spillage protection and collection system (OSC) on all RCP motors in both units. This design change was installed to meet commitments to the NRC in response to BTP 9.5-1. Each RCP motor has a 265 gallon lubricating oil reservoir coupled to the RCP motor oil lift system, necessary for proper operation of the motor. The drip pans supplied by the original equipment manufacturer (OEM) were shown to be sufficient to contain ordinary drips of oil. The oil lift system, however, is pressurized during start up and a pressurized oil leak could not be handled by the OEM drip pans and could potentially cause a fire. The OEM designed and provided analysis for the additional oil spill and collection system. Analysis included the appropriate seismic, missile, and high energy line break considerations. The design change also installed level indication for the collection tank with readout to control room. Plant procedures were updated to include surveillance of the integrity of the system and assurance that the system is empty prior to startup.

One minor nonconformance to the design was identified and corrected in 1989, in which a fillet weld size was found to be different than the one shown on the drawing. Even though the existing weld was found to be adequate, it was repaired to comply with the

requirements of the design drawing. This condition was identified and corrected under the corrective action system.

The review concluded that this design change had no adverse impact on the operability of other systems, structures, or components. The review also concluded that the design change met the design bases.

RFC-DC-12-2229: Installation of Fire Protection Hose Reels

This design change added additional hose station capability at access areas to the control room cable spreading room, the auxiliary cable vault, the switchgear room cable vault and the containment penetration cable tunnels. The design change was initiated in 1978 as a part of our response to BTP 9.5-1 and to meet commitments to the NRC to improve manual fire fighting capability in safety related areas of the plant. Three revisions to the design change were made with the final installation being completed in 1985. One of these revisions was a result of a corrective action report issued in 1984 that identified that fire protection piping installed under revision 0 was not adequately supported to seismic design criteria. This was subsequently reported to the NRC as LER 86-003.

The review of the complete RFC-2229 package determined that the final installed design change met its intended functional objectives, was installed in accordance with the issued design and the appropriate drawings were identified and documented as having been updated. The review found no conditions adverse to the quality of safety related systems structures or components, other than those resolved by the revision. There was no adverse impact on the operability of safety related systems, structures or components and the design change met its design bases. The review included a review of the original copy of the design change package and subsequent revisions by fire protection and design engineers.

RFC-DC-12-2276: Provide Capability to Stop, Start, and Control Voltage Speed of the Emergency Diesel Generators (EDGs) from the Local Subpanel

RFC-DC-12-2276 provides the capability for local control of the EDGs on both units 1 and 2 in the event of control room uninhabitability. The stop, start, and control functions of each EDG are provided from a local subpanel in each EDG room. This was accomplished by installing an additional auxiliary subpanel in each EDG room and mounting the control functions on the new subpanel. A switch for a transfer circuit to isolate the diesel control from the control room was mounted on the new subpanel. The start and stop (trip) function for the EDGs already existed on the original EDG local subpanel and the scope of the design change was changed to reflect this.

This review determined that the design change met its intended functional objectives, and met its design bases. The engineering and design was developed and appropriate drawings were issued. The review further concluded that there was no adverse impact on the operability of safety related systems, structures, and components. The review included a walkdown of the subpanels to verify component installation, bolting arrangement, adequate emergency lighting, and ease of future maintenance activities. No nonconforming conditions

were identified as a result of this technical review of RFC-DC-12-2276 that would affect the operability of the EDGs.

A condition report was generated on December 11, 1997, that is related to this design change. During troubleshooting of the EDG manual voltage regulator, a discrepancy between the electrical drawing and the actual wiring configuration for the local panel was noted. The drawing shows the correct terminal connections for the voltage adjustment potentiometers, but the wires were reversed. While this discrepancy could have caused confusion in the setup of the device, it does not affect the operability of the EDG.