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OPPD
Omaha Public Power District

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December 3, 2002
LIC-02-0137

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References:
1. Docket No. 50-285
2. Letter from OPPD (D. J. Bannister) to NRC (Document Control Desk), Fort Calhoun Station Unit No. 1 License Amendment Request, Low Pressure Safety Injection System Allowed Outage Time, dated October 8, 2002 (LIC-02-0097)

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, "Low Pressure Safety Injection System Allowed Outage Time," – Additional Information

In Reference 2, Omaha Public Power District (OPPD) submitted an Application for Amendment of Facility Operating License to revise the Fort Calhoun Station (FCS) Unit No. 1 Technical Specifications (TS). In a telephone discussion with Mr. A. B. Wang (NRC Project Manager) on November 14, 2002, OPPD verbally communicated its intention to provide additional discussion and justification for the proposed amendment. Attached please find the responses to NRC questions supporting the low pressure safety injection system (LPSI) allowed outage time (AOT) extension amendment.

If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833. No commitments are made to the NRC in this letter.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on December 3, 2002)

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/RRL/rrl

A001

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Attachment: Responses to NRC LPSI AOT Questions

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska
Winston & Strawn

ATTACHMENT

Responses to NRC LPSI AOT Questions

Responses to NRC LPSI AOT Questions

Question 1
PRA quality - Was your probabilistic risk assessment (PRA) reviewed by the Combustion Engineering Owners Group (CEOG)? What do you do to maintain and assure the quality of your PRA?

Response 1

In March 1999, the Fort Calhoun Station PRA was peer reviewed by a team of PRA engineers from Westinghouse, four other utilities and a PRA consultant. This peer review was the first conducted in accordance with the CEOG implementation of the nuclear industry peer review process as documented in NEI 00-02, *Probabilistic Risk Assessment (PRA) Peer Review Process Guidance.* The following paragraphs briefly discuss the conclusions of that peer review.

The peer review team found the Fort Calhoun Station PRA to be effective for assessing planned plant maintenance and operations configurations and evaluating future plant design changes. The PRA was also found to be adequate for other applications which are supported by deterministic insights and plant expert panel input. The review did identify some areas of weakness in the PRA that should be considered in any application. The review also identified several areas of strength in the Fort Calhoun Station PRA.

The review team found the Fort Calhoun Station PRA to be strong in the areas of initiating event identification and containment performance analysis. OPPD had a particularly good treatment of the containment reliability analysis.

The reviewers recommended that the plant dependency analysis be upgraded. As the result of an in-depth investigation of dependencies, one missed dependency for the diesel-driven auxiliary feedwater pump, FW-54, was identified and corrected. Improvement in the documentation of the dependency matrix was also recommended. This activity was tracked by a configuration control program and was integrated in the Revision 3 PRA model used for severe accident mitigation alternatives (SAMA) assessment.

PRA quality is maintained in accordance with procedure PED-SEI-37, "Probabilistic Risk Assessment Configuration Control". This procedure describes the PRA inputs, such as plant modifications and equipment failure history, which are reviewed and compared against the PRA model. The PRA model is typically revised once per operating cycle, and more frequently if warranted by major changes. The PRA model currently in use is Revision 5.

Question 2
What considerations did you make for external events?

Response 2
The Fort Calhoun Station Individual Plant Examination for External Events (IPEEE) was submitted to the NRC on June 30, 1995. The core damage frequency (CDF) contributions from high winds and tornados, and from toxic hazards, were each <1.0% as shown in Table 1.4.1 of the submittal. Since extending the low pressure safety injection (LPSI) pump AOT would not affect the relative contributions from tornados and toxic hazards, they are not addressed in the discussion below.

Seismic
The FCS design basis requires meeting single failure criteria for seismic events up to the design basis earthquake (0.17g). Existing controls for maintaining the design basis of the plant are in place and are implemented to ensure the design basis is not compromised when removing equipment from service. Such practices help keep the seismic risk-significance of plant configurations low.

Higher seismic hazard levels (> 0.1g) are sufficiently low in expected frequency that the probability of such an earthquake during a maintenance activity period of one week is of the order of 1E-6 or less. Seismic hazard levels greater than 0.1g have a frequency of approximately 7E-5/year or lower based on Electric Power Research Institute (EPRI) curve SLFC-93-1421. Hence, seismic hazard levels greater than 0.1g are not expected to have a significant impact on the decisions regarding acceptable plant configurations. Nevertheless, knowledge of seismic risk drivers at lower g-levels (< 0.1g) is useful in considering the seismic risk-significance of certain anticipated plant configurations. This is due largely to possible impact on non-design basis equipment that may not be protected by existing seismic controls. For this reason, the impact of <0.1g seismic accelerations is explicitly and routinely quantified by the PRA model as part of the 10 CFR 50.65(a)(4) risk assessment process.

Fire
The probability of plant fires is not assessed for distinct plant activities such as LPSI pump maintenance. However, the affect of fire risk upon an extended Technical Specification AOT can be evaluated qualitatively. Within the context of the power operation PRA, LPSI pumps serve three functions: injection following a large-break loss of cooling accident (LOCA); alternate hot leg injection following a large-break LOCA; and, establishment of shutdown cooling. The third function, establishment of shutdown cooling, is the only function applicable to fire PRA since fires concurrent with large-break LOCA's are not considered.

Shutdown cooling is included in the fault tree logic for two fire core damage sequences: transient initiating event with failure of long-term decay heat removal (sequence at TX), and transient-induced RCP seal LOCA with failure of long-term decay heat removal (sequence at TQ2X). Review of the fire IPEEE model and results led to the following conclusions. First, many of the fire areas involve consequential failure of shutdown cooling, in which case LPSI pump outages are moot. Second, most of the remaining fire areas have no direct impact upon shutdown cooling. For these fire areas, individual LPSI pump outages are of low risk significance. Finally, there are a few fire areas that involve consequential failure of one LPSI pump, in particular fires in the LPSI pump rooms. These fire areas are not risk significant since they have no affect upon main or auxiliary feedwater, and since they degrade but do not fail once-through-cooling.

Fire safety is further enhanced by the implementation of the following procedures:

- Standing Order G-103, "Fire Protection Operability Criteria and Surveillance Requirements";
- Standing Order G-58, "Control of Fire Protection System Impairments."

External floods

As with fire, the LPSI function of interest is shutdown cooling, and the probability of external floods is not assessed for distinct plant activities such as LPSI pump maintenance. However, the relationship between external floods and an extended LPSI pump AOT can be evaluated qualitatively.

The consequences of external floods can generally be placed into two categories. The moderate floods cause loss of off-site power and failure of some structures, systems and components (SSC) in the turbine building. The large floods cause loss of off-site power and failure of numerous risk-significant SSC's. Since the large floods also cause failure of shutdown cooling, only the moderate floods will be evaluated further.

Moderate floods are caused by rising level of the Missouri river, a process that is relatively slow and predictable. Consequently, there is generally ample time available to place the plant in a stable condition and restore unavailable equipment. There are also opportunities for alternate means of decay heat removal, such as long-term refilling of the emergency feedwater storage tank, or use of the containment spray pumps for shutdown cooling (although directed by procedure, this option requires reactor coolant system temperature <120 °F and the pressurizer manway open). It is expected that Fort Calhoun Station will eventually adopt hot shutdown as a high river level Technical Specification end state (CE-NPSD-1186-A, Rev 00, "Technical Justification for the Risk Informed Modification to Selected Required Action End States for CEOG Member PWRs", Westinghouse Electric Company, October 2001), further reducing the importance of shutdown cooling. For these reasons, extension of the LPSI pump AOT would not have a significant impact upon external flood risk.

Question 3
What was done for Tier 2?

Response 3
The Maintenance Rule 10 CFR 50.65(a)(4) process provides sufficient configuration control for LPSI pump outages. The (a)(4) process includes solving the PRA model with

the software tool Equipment Out of Service (EOOS), so that any risk-significant configurations are automatically identified. EOOS also identifies risk-significant initiating events, and maintenance activities are restricted accordingly. Since the (a)(4) process explicitly models configuration changes that could increase the risk significance of a LPSI pump outage, no additional Tier 2 restrictions are needed.

The process for on-line maintenance activities is described in Standing Order SO-M-101, "Maintenance Work Control". This procedure applies to both planned and emergent maintenance activities. SO-M-101 describes responsibilities for conduct of the (a)(4) process, addresses both quantitative and qualitative evaluation of maintenance activities, and provides criteria for determining when consideration of risk management actions is warranted. Examples of typical risk management actions are provided. SO-M-101 also provides a trigger for the Plant Review Committee to review maintenance activities. Risk assessments for power operation are both quantitative and qualitative, the quantitative portion being supported by the software tool EOOS.

Additional guidance for the risk assessment process is provided in FCSG-19, "Performing Risk Assessments". The guidance supports SO-M-101 in that the standing order specifies the requirements, and the FCSG provides detailed guidance for complying with the requirements. The majority of the detailed guidance deals with the operation of the EOOS software.