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402/636-2000

July 21, 1992

LIC-92-0145

Mr. Martin J. Virgilio
Assistant Director for Region IV and V Reactor Projects
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rock ille, MD 20852

REFERENCE: Docket No. 50-285

SUBJECT: Request for Waiver of Compliance from the Provisions of Technical Specification 3.17(3)(iii)3

Dear Mr. Virgilio:

The Omaha Public Power District (OPPD) respectfully requests a one-time waiver of compliance from the provisions of Technical Specification 3.17(3)(iii)3. The requirements of this specification are:

> "*(iii) Unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 3-13 during the shutdown subsequent to any of the following conditions:...*
>
> *3. A loss-of-coolant accident requiring actuation of the engineered safeguards...*"

The sample size of the first sample inspection specified in Table 3-13 is as follows:

> "*A minimum of 300 tubes per steam generator*"

Discussion

On July 3 - 4, 1992, Fort Calhoun Station experienced a Reactor Coolant System (RCS) pressure transient which resulted in Pressurizer Safety Valve RC-142 opening and not reseating. The initial leak rate from RC-142 was greater than 120 gpm resulting in partial actuation of the engineered safeguards (i.e., no safety injection tank discharge, low pressure safety injection or containment spray actuation occurred). Using the nominal three-inch size for the open safety valve, the break size was calculated to be 0.049 ft^2 which constitutes a small break Loss of Coolant Accident (LOCA) as defined in USAR Section 14.15. Therefore, the criteria of the subject Technical Specification is applicable.

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The origin of the subject Technical Specification requirement is based on Regulatory Guide 1.83 Revision 1 (July 1975) which states that Steam Generator tubes should be inspected following a LOCA requiring actuation of engineered safeguards. Fort Calhoun Station Technical Specifications, as taken from Standard ABB Combustion Engineering (ABB/CE) Technical Specifications, state that the inspection shall be performed.

Basis

As required by NRC General Design Criteria, Fort Calhoun Station's USAR Section 4.3.4 identifies design transient cycles for which the Steam Generators were designed. The July 3 - 4, 1992 event was enveloped by these transients. In addition, the transient is within the Nuclear Steam Supply System (NSSS) design basis of the USAR. The peak RCS pressure was approximately 2430 psia which is less than design pressure (2500 psia) and significantly less than 110% of design pressure (2750 psia). In accordance with USAR Section 4.3.4, Fort Calhoun Station Steam Generators are analyzed to withstand abnormal conditions resulting from 40 cycles of turbine loss of load without immediate reactor trip. As of July 20, 1992, Fort Calhoun Station has experienced only three such events.

ABB/CE has concluded that during the recent event, thorough mixing of the safety injection water with the RCS occurred, and the bulk temperature of the RCS was representative of the bulk temperature seen in the Steam Generators. The temperature seen by the Steam Generators was similar to that expected during a normal plant trip. Consequently, the loads (e.g., tensile and/or compressive stresses generated by differential temperature) introduced by this temperature transient are small. Previous analyses performed by ABB/CE have shown that the integrity of the Steam Generator tubes would be maintained even under the severe loads imposed by a double-ended guillotine break LOCA. The RCS pressure transient from the recent small break LOCA event is significantly less than that of the guillotine break and the resultant loads imposed by this pressure transient would not challenge the Steam Generator tube integrity. Throughout the recent LOCA event, RCS pressure was maintained greater than pressure in the secondary system imposing no abnormal differential load on the Steam Generator tube bundle.

Results of four consecutive eddy current exams performed since 1985, including the most recently completed during the 1992 Refueling Outage, have shown the Fort Calhoun Station Steam Generators to be in Technical Specification Category C1 (no action required). Consequently, the condition of the Steam Generator tubes when the LOCA occurred was excellent with no technical reason to expect that the July 3 - 4, 1992 temperature and pressure transient would have propagated any existing indications in the Steam Generator tubes. This is further supported by a hydrostatic test of the Reactor Coolant System to 2150 psia for three hours which was performed on July 20, 1992 to ensure Steam Generator tube integrity. No leakage through the Steam Generator tubes occurred as verified by comparison of secondary system samples taken prior to July 3, 1992, during the current shutdown, and samples taken after the July 20, 1992 hydrostatic test.

Routine monitoring of Steam Generator integrity will proceed in accordance with existing plant Standing Orders for potential leakage and secondary side radiological activity. The RCS leakrate determination is performed daily with a sensitivity of approximately ±0.05 gpm and the results are reviewed by Operations and Engineering personnel. Further, the PRC is required by procedure to review these results and associated action plans when total leakrate exceeds 0.2 gpm for three consecutive days. This is well below the Fort Calhoun Station Technical Specification limit of 1.0 gpm. Primary-to-secondary leakage is monitored by weekly samples and analysis capable of detecting 1E-7 µCi/gm I-131. This assures that the Technical Specification limit of 1.0 gpm for both Steam Generators is met.

Conclusion

OPPD requests timely review and approval of this request to preclude a delay in startup from the July 3 - 4, 1992 shutdown. Performance of the Steam Generator tube examination would require cooldown of the RCS, mobilization of contractor personnel to perform the examination and a resultant minimum delay of approximately 12 - 14 days in return to power operations. Additionally, the requirement to perform Steam Generator tube inspections following a LOCA with engineered safeguards actuation is a requirement that is not technically justified in this situation. The Steam Generator inspection would result in significant radiation exposure to personnel involved in this type of inspection. The inspection would require an additional cooldown and heatup cycle to the RCS, and the inspection would have to take place in its entirety with the RCS at mid-loop, which is not desirable from a shutdown risk standpoint.

This waiver request could not have been reasonably avoided as the July 3 - 4, 1992 event invoked the Technical Specification requirement. Although a more timely identification of this Technical Specification requirement would have precluded the need for an expedited NRC review of this waiver request, OPPD would have still requested this waiver based on the justification provided in this letter.

Approval of the waiver of compliance will not involve a significant hazards consideration or adverse environmental consequences because a significant transient was not seen by the Steam Generator tubes which could have caused any existing indications in the tubes to propagate or any new degradation to occur. It is thus concluded that the requested waiver will not affect the health and safety of the public. This proposed waiver has been reviewed and approved by the Plant Review Committee.

Sincerely,

W. G. Patton
for W. G. Gates
Division Manager
Nuclear Operations Division

WGG/ljj

c: LeBoeuf, Lamb, Leiby and MacRae
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