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the southern electric system

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Docket No.: 50-364

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

Joseph M. Farley Nuclear Plant - Unit 2
Request for Enforcement Discretion

Ladies and Gentlemen:

As a result of a misapplication of the technical specification requirements regarding F*, steam generator tubes with axial indications have been left in service which do not meet the technical specification requirements. As discussed per telephone-conference with NRC staff on the evening of April 22, 1996, enforcement discretion was requested and approved to allow continued operation of FNP Unit 2 within conditions of the relevant L* parameters bounded by the six tubes discussed during the conference. Operation of Unit 2 under NOED was approved until NRC approval of a one time cycle specific L* technical specification change. Attachment 1 provides the basis for Notice of Enforcement Discretion for Technical Specification 4.4.6.4.11. L* criterion allows a steam generator tube to remain in service with bands of axial degradation in the tube sheet region provided sufficient non-degraded tubing remains to satisfy regulatory guidance concerning structural and leakage integrity. The proposed FNP cycle specific emergency technical specification associated with L* is similar to a previously NRC approved V. C. Summer nuclear plant submittal.

If there are any questions, please advise.

Respectfully submitted,

DH Morey

Dave Morey

Sworn to and subscribed before me this 23rd *day of* April *, 1996.*

Martha Gayle Dow
Notary Public
My Commission Expires: November 1, 1997

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Attachments:

1. Request for Enforcement Discretion
2. Marked Up Proposed Technical Specification Pages

cc: Mr. S. D. Ebneter, Region II Administrator
Mr. B. L. Siegel, NRR Senior Project Manager
Mr. T. M. Ross, FNP Sr. Resident Inspector
Dr. Donald E. Williamson, State Department of Public Health

Attachment 1

Request for Enforcement Discretion

Request for Enforcement Discretion

1. The Technical Specification that will be violated.

Farley Unit 2 Technical Specification 4.4.6 requires that F* tubes; i.e., tubes with degradation equal to or greater than 20%, less than 1.79 inches below the top of the tubesheet or the bottom of the roll transition, whichever is lower; be repaired based on the steam generator technical specifications. Tubes with degradation greater than 40% below the F* distance which remain in service are considered F* tubes. As a result of the misapplication of the technical specification requirements, tubes with assumed depths greater than 40% were left in service less than 1.79 inches below the top of the tubesheet or bottom of the roll transition, whichever is lower. Provided below is the data for each tube left in service and discussed with the NRC on April 22,1996:

S/G	Row	Col	RPC Inspection Length in.	L* Length (RPC to crack [illegible]) in.	TOTAL SRE inches	Axial Crack Length in.	Inclination Angle Degrees	Number of distinct indication found
B	19	45	>3.1	-1.45	5.58	0.23	14	3
C	07	42	>3.1	-1.77	2.30	0.39	12	5
C	11	63	>3.1	-1.50	2.07	0.36	15	4
C	10	64	>3.1	-1.73	3.87	0.38	08	3
C	08	65	>3.1	-1.72	4.22	0.32	12	3
C	13	65	>3.1	-1.63	2.38	0.20	08	1

2. Circumstances surrounding the situation, including root causes, the need for prompt action, and identification of any relevant historical events.

This event was discovered because of questions asked during a pilot INPO Steam Generator evaluation. Westinghouse developed and site approved inspection procedures that had been independently reviewed but failed to properly determine the F* distance as defined by Unit 2 technical specifications. These procedures were in use for several cycles. This NOED request could not have been foreseen in advance since no prior knowledge of an event of this type is known to have occurred in the industry. The root cause for this event is personnel error in that inadequate procedures were developed and approved for FNP steam generator inspections. Prompt action is needed to avoid undesirable transients potentially associated with unit shutdown due to forced compliance for non safety significant issues. This event has never occurred at FNP.

3. Safety basis for the request, including an evaluation of the safety significance and potential consequences of the proposed course of action.

Safety Analysis

Operation with 6 or more tubes that fail to meet the F* technical specification but meet the criteria for L* as described below is acceptable. The basis for this is provided in SNC submittal "Joseph M. Farley Nuclear Plant Unit 2 L* Tubesheet Region Plugging Criterion" dated April 22, 1996. All structural integrity and leakage requirements of draft Reg. Guide 1.121 continue to be met for the tubes in question. Below is an excerpt from the "Joseph M. Farley Nuclear Plant Unit 2 L* Tubesheet Region Plugging Criterion" technical specification submittal dated April 22, 1996 to support our safety basis for continued operation. It should be noted that the values of L* criterion in the "Joseph M. Farley Nuclear Plant Unit 2 L* Tubesheet Region Plugging Criterion" dated April 22, 1996, bound the values of the Unit 2 cycle 11 specific L* criterion associated with this submittal. The reason for this is that this submittal is intended to support Enforcement Discretion allowing operation with tubes exceeding the currently approved F* technical specification criteria but are conservative with respect to the proposed L* technical specification submitted April 22, 1996, thereby providing sufficient safety margin.

Unit 2 Cycle 11 Specific L* Criterion

Parameter	L*criterion for cycle 11	Base L* criterion
Minimum distance of SRE	2.07 inches	1.84 inches
Maximum number of distinct degradation areas in a band	5	15
Maximum inclination angle of single band	15 degrees	30 degrees
Maximum crack length	.39 inches	.5 inches
Minimum distance of SRE from the bottom of the transition roll to the top of the indication	1.45 inches	.56 inches

Provided below is a excerpt from the "Joseph M. Farley Nuclear Plant Unit 2 L* Tubesheet Region Plugging Criterion" technical specification submittal dated April 22, 1996 to support our safety basis for continued operation. This excerpt combined with the margins noted above form the basis for this enforcement discretion.

DESCRIPTION OF THE AMENDMENT REQUEST

The L* criterion assesses indications of degradation within the roll expanded region of steam generator tubes based upon location of the indication, the axial length of the indication and the inclination angle of the indications. The tubes to which the criteria are applied must meet the following restrictions:

- Each tube must be inspected using the RPC probe or other advanced NDE inspection method for a minimum distance of 3.1 inches below the bottom of the roll transition or the top of the tubesheet, whichever is lower. This is termed the inspection region.

- A minimum length of sound roll expansion of 0.50 inch plus allowance for eddy current measurement uncertainty extending down from the bottom of the roll expansion or the top of the tubesheet, whichever is lower, must be established. This is termed L* length.

- For tubes where a single degradation band is evidenced within the inspection region, an aggregate distance of sound roll expansion of 1.74 inch minimum plus allowance for eddy current measurement uncertainty, must be established. This aggregate distance includes the L* length. Single band degradation is limited to 0.50 inch axial length, with an inclination angle of not more than 30 degrees from the vertical, and limited to 15 distinct degradation areas within the band. SNC has administratively limited the number of distinct degradation areas in the band to 15, although the Westinghouse support documentation justifies up to 30 such areas.

- For tubes where multiple degradation bands (maximum of two) are detected within the inspection region, an aggregate distance of sound roll expansion of 1.87 inch plus allowance for eddy current measurement uncertainty, must be established. Multiple degradation bands are limited to 0.5 inch axial length each, with an inclination angle of no more than 30 degrees from the vertical, and limited to 15 distinct degradation areas within each band. SNC has administratively limited the number of distinct degradation areas in the band to 15, although the Westinghouse support documentation justifies up to 30 such areas.

- The L* criterion is limited to 600 tube ends per steam generator.

If any of these items is not validated, the tube must be repaired.

The total length of aggregate sound roll expansion needed to prevent pullout during all plant conditions for the single and multiple band degradation cases is termed the pullout load reaction length (PLRL). This term is equivalent to the L* distance.

EVALUATION

The proposed license amendment (Technical Specification change) addresses the action required when degradation has been detected in the top portion (within the F* distance) of the mechanically expanded portion of steam generator tubes within the tubesheet region. The presence of the tubesheet enhances the integrity of degraded tubes in that region by precluding tube deformation beyond the expanded outside diameter. The roll expansion of the tube into the tubesheet provides a barrier to significant leakage for through wall cracking of the tube in the expanded region. L* criterion approved for implementation at similar plants include provisions which account for the support of the tubesheet in the portion of the tubes below the top of the tubesheet. The previously approved (at Farley Unit 2 and similar plants) F* criterion was not established to address degradation in the top portion of the tube expansion.

F*/L* Plugging Criteria Comparison

As stated previously, the L* criterion is essentially an extension of the existing F* methodology. WCAP-1[illegible]6 Rev. 2 provides the basis for the F* criterion currently licensed at Farley Unit 2. Both criteria (F* and L*) provide for structural integrity of the tube during all plant conditions by verification of SRE lengths. The SRE lengths provide for resistance load capabilities such that the most limiting postulated pressure developed

end cap loading defined in Regulatory Guide 1.121 is balanced. In the F* criterion, a SRE length of 1.54 inch plus allowance for eddy current uncertainty measurements must be established below either the bottom of the roll transition, or the top of tubesheet, whichever is lower. The F* distance of 1.54 inch is established based on faulted condition loadings and represents the limiting RG 1.121 loading. The F* criterion precludes primary to secondary leakage during all plant conditions. The L* criterion addresses tubes which contain axial or nearly axially oriented eddy current indications within the F* length. The L* criterion includes a minimum length of SRE immediately below the bottom of the roll transition or top of tubesheet, whichever is lower, which precludes significant leakage during all plant conditions. This distance plus allowance for eddy current measurement uncertainty is termed L* length. The SRE length above a crack is summed with the SRE length below the crack to ensure structural integrity during all plant conditions. This distance of total aggregate SRE is termed L* distance. In both the F* and L* evaluation bases, circumferential separation of the tube is postulated below either the F* or L* distance. Verification of the SRE F* or L* distances provides for structural integrity even for the postulated condition of a tube separation below the F* or L * distance. Tube rupture can occur if a F* or L* tube is postulated to separate and lift out of the tubesheet due to the end cap load. Verification of the F* or L* distance prevents tube lift from occurring and therefore prevents the possibility of a tube rupture even for a tube postulated to become separated below the F* or L* distance.

L* Evaluation Discussion

The proposed change designates a portion of the tube for which tube degradation of a defined type does not necessitate remedial action except as dictated for compliance with tube leakage limits as set forth in the Farley Nuclear Plant Unit 2 Technical Specifications. As noted above, the area subject to this change is in the expanded portion of the tube within the tubesheet of the steam generators. The length of mechanical expansion required to resist significant leakage for all normal operating and postulated accident conditions is designated L* length. Tubes whose integrity is ascertained using L* length will not contribute significantly to offsite dose. In addition to determining that the expansion of the tube within the L* length is not degraded, a determination of the condition of tube degradation in the portion of the tube below the L* length must be made to verify that the degraded tube has sufficient structural integrity.

The proposed amendment would modify Technical Specification 3/4.4.6 "Steam Generator Surveillance Requirements" which provides tube inspection requirements and acceptance criteria to determine the level of degradation for which the tube may remain in service. The proposed amendment would add definitions required for the alternate plugging criteria and prescribe the portion of the tube subject to the criteria. The proposed Technical Specification changes accompany this analysis.

ANALYSIS

Conformance of the proposed amendments to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:

1. Operation of the Farley Nuclear Plant Unit 2 steam generators in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The supporting technical evaluations of the subject criteria demonstrate that the presence of the tubesheet enhances the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. The result of the hardroll of the tube into the tubesheet is an interference fit between the tube and the tubesheet. Tube rupture can not occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated LOCA loadings.

The type of degradation for which the L* criterion has been developed (cracking with an axial or near axial orientation) has been found not to significantly reduce the axial strength of a tube. An evaluation including analysis and testing has been done to determine the strength reduction for axial loads with simulated axial and near axial cracks. This evaluation provides the basis for the acceptance criteria for tube degradation subject to the L* criterion.

The SRE L* length is sufficient to preclude significant leakage from tube degradation located below the L* length. The existing Technical Specification leak rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. Any leakage from the tube within the tubesheet at any elevation in the tubesheet is fully bounded by the existing steam generator tube rupture analysis included in the Farley Nuclear Plant Final Safety Analysis Report. A conservative leakage allowance for each L* tube is provided to determine the impact of L* criterion upon offsite doses in the event of a postulated double ended guillotine break of the main steam line outside of containment, but upstream of the main steam line isolation valves. Since Farley Unit 2 has implemented the Interim Plugging Criteria (IPC) for ODSCC at the tube support plates, projected steam line break (SLB) leakage at the end of the next successive operating cycle must be evaluated. Per Generic Letter 95-05, plants implementing the IPC can utilize SLB leakage limits higher than the originally assumed 1.0 gpm primary to secondary leakage value provided an analysis of offsite doses consistent with the Standard Review Plan methodology is performed. This analysis performed for the Farley Unit 2 plant indicates that primary to secondary leakage of 11.2 gpm in the faulted loop (0.1 gpm in the intact loops) will result in offsite doses at the site boundary of less than 10% of the 10 CFR 100 guidelines. The total projected SLB leakage from all leakage sources must remain below this value. Per attachment 4 addressing the L* methodology, the number of tube ends to which L* criterion can be applied is limited to 600 per steam generator. Using a bounding SLB leakage allowance per L* tube, the SLB leakage component from 600 L* tube ends will be less than 0.33 gpm in the faulted loop. The proposed alternate plugging criterion does not adversely impact any other previously evaluated design basis accident. As the current Unit 2 IPC SLB leakage has been calculated to be less than 2 gpm in the faulted loop, a SLB leakage margin of over 9 gpm is provided for this cycle.

As noted above, tube rupture and pullout is not expected for tubes using the L* criterion. In addition to the L* length, a minimum length of SRE below the identified degradation must be established. The aggregate L*distance of SRE provides the structural integrity to prevent tube pullout. Conservatively, it is assumed that the degraded band length does not provide any support in resisting tube pullout.

Therefore SNC concludes that Operation of the Farley Nuclear Plant Unit 2 steam generators in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed L* criterion does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to result in an accident initiated outside of the region of the tubesheet expansion. The structural integrity of L* tubes will be maintained during all plant conditions. Any hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. If it is postulated that a circumferential separation of an L* tube were to occur below the PLRL, tube structural and leakage integrity will be maintained during all plant conditions. Verification of the L* distance of non-degraded tube roll expansion prevents the postulated separated tube from lifting out of the tubesheet during all plant conditions. Verification of the L* criterion prevents tube displacement of any magnitude, and therefore, postulated axial cracks existing a minimum of 0.5 inch from either the bottom of the roll transition or top of tubesheet, whichever is lower, from migrating out of the tubesheet.

Therefore, SNC concludes that the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

The use of the L* criterion has been concluded to maintain the integrity of the tube bundle commensurate with the requirements of draft Regulatory Guide 1.121 under normal and postulated accident conditions. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The L* length has been verified by testing to be greater than the length of roll expansion required to preclude significant leakage during normal and postulated accident conditions. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the FSAR accident analyses. The L*distance provides for structural integrity during all plant conditions.

Implementation of the L* criterion will decrease the number of tubes which must be taken out of service with tube plugs or repaired with sleeves. Both plugs and sleeves reduce the

RCS flow margin, thus implementation of the L* criterion will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

Therefore, SNC, concludes based on the above, it is concluded that the proposed change does not result in a significant reduction in a loss of margin with respect to plant safety as defined in the Final Safety Analysis Report or the bases of the FNP technical specifications.

CONCLUSION

Based on the preceding analysis, it is concluded that operation of the Farley Nuclear Plant Unit 2 steam generators in accordance with the proposed amendment does not involve a significant hazards consideration as defined in 10 CFR 50.92.

4. The basis for the conclusion that the noncompliance will not be of potential detriment to the public health and safety and that neither an unreviewed safety question nor a significant hazard consideration is involved.

Based on the allowable steam generator primary-to-secondary leakage rate calculated in support of the alternate repair criteria, the radiological consequences of a release outside containment will continue to remain within a small fraction of the guideline values of 10 CFR 100.

5. The basis for the conclusion that the noncompliance will not involve adverse consequences to the environment.

Application of the L* criterion on a one time cycle specific basis for 6 or more tubes will not involve any significant change in the types of effluents that may be released offsite and no significant increase in the individual or cumulative occupational radiation exposure. Therefore, this request for enforcement discretion does not involve any irreversible environmental consequences.

6. Proposed compensatory measures.

Procedural guidance for primary-to-secondary leakage will be reviewed with on shift licensed crews. Furthermore, steamline and feedline break procedures will be reviewed with on shift licensed operating personnel. FNP has the N16 primary to secondary monitoring system.

7. Justification for the duration of the noncompliance.

The justification for the duration of the noncompliance is based on the fact that existing structural and leakage integrity requirements for the 6 or more tubes of concern continue to meet regulatory guidelines of draft Regulatory Guide 1.121 for the remainder of the operating cycle based on L* criterion. To shutdown and perform plugging operations for these tubes could result in undesirable transients. In addition for Farley Unit 2, the probability of a steam line break occurring prior to the Unit 2 Fall 1996 outage is low. Consequently, operation with the flaws of concern in service is acceptable until a Technical Specification amendment can be submitted by SNC and approved by the NRC.

8. Review by the Plant Operations Review Committee.

This request for enforcement discretion has been reviewed and approval has been recommended by the organization tasked to advise the General Manager - Nuclear Plant on all matters related to nuclear safety at Farley Nuclear Plant, i.e., the Plant Operations Review Committee.

9. Satisfaction of NOED criteria.

This NOED is intended to avoid undesirable transients as a result of forcing compliance with a license condition and, thus, minimizing potential safety consequences and operational risks. The structural and leakage integrity regulatory requirements continue to be met, as discussed in 3 above. Consequently, shutdown to inspect the steam generator tubes will not provide any additional margin of safety; however, it will result in unnecessary operational transients without any benefit. Southern Nuclear believes that it has met the criteria for NOED as provided NRC Inspection Manual Part 9900: 10 CFR Part 2 Appendix C Enforcement Discretion.

10. Marked-up Technical Specification pages showing the proposed changes.

A copy of the referenced marked up FNP technical specification pages is provided in attachment 2.

11. Prior adoption of approved line-item improvements to the technical specifications of the improved technical specifications would not have obviated the need for the NOED request.

This issue is included in the on going efforts to develop a steam generator integrity rule and is not addressed by line item improvements.

12. Additional information requested by NRC Staff.

At this time no additional information has been requested by the staff.

Attachment 2

Marked Up Proposed Technical Specification Pages