Attachment 3

Revised Technical Specification Pages

for the

Interim Plugging Criteria

<u>Unit 2</u>

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

3. Degraded Tube means a tube, including the sleeve if the tube has been repaired, that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing a defect is defective.

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. This definition does not apply to the area of the tubesheet region below the F* distance in the F* tubes. For a tube that has been sleeved with a mechanical joint sleeve, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging. For a tube that has been sleeved with a welded joint sleeve, through wall penetration greater than or equal to 37% of sleeve nominal wall thickness in the sleeve between the weld joints requires the tube to be removed from service by plugging. At tube support plate intersections, the repair limit for the Ninth Operating Cycle is based on maintaining steam generator tube serviceability as described below:

   a. An eddy current examination using a bobbin probe of 100% of the hot and cold leg steam generator tube support plate intersections will be performed for tubes in service.

   b. Degradation within the bounds of the tube support plate with bobbin voltage less than or equal to 1.0 volt will be allowed to remain in service.

   c. Degradation within the bounds of the tube support plate with a bobbin voltage greater than 1.0 volts will be repaired or plugged except as noted in 4.4.6.4.a.6.d below.

d. Indications of potential degradation within the bounds of the tube support plate with a bobbin voltage greater than 1.0 volt but less than or equal to 3.6 volts may remain in service if a rotating pancake coil probe (RPC) inspection does not detect degradation. Indications of degradation with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

7. Unserviceable describes the condition of a tube or sleeve if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.6.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube.

9. Tube Repair refers to mechanical sleeving, as described by Westinghouse report WCAP-11178, Rev. 1, or laser welded sleeving, as described by Westinghouse report WCAP-12672, which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. For the Ninth Operating Cycle only, primary-to-secondary leakage through all steam generators shall be limited to 450 gallons per day and 150 gallons per day through any one steam generator.

For subsequent cycles, 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM UNIDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

3/4.4.6 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired.

For the Ninth Operating Cycle only, the repair limit for tubes with flaw indications contained within the bounds of a tube support plate has been provided to the NRC in Southern Nuclear Operating Company letter dated February 20, 1992. The repair limit is based on the analysis contained in WCAP-12871, Revision 2, "J. M. Farley Units 1 and 2 SG Tube Plugging Criteria for ODSCC at Tube Support Plates." The application of this criteria is based on limiting primary-to-secondary leakage during a steam line break to less than 1 gallon per minute. Primary-to-secondary leakage during this cycle only is limited to 150 gallons per day per steam generator during normal operation.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. If a sleeved tube is found to have through wall penetration of greater than or equal to 31% for the mechanical sleeve and 37% for the laser welded sleeve of sleeve nominal wall thickness in the sleeve, it must be plugged. The 31% and 37% limits are derived from R.G. 1.121 calculations with 20% added for conservatism. The portion of the tube and the sleeve for which indications of wall degradation must be evaluated can be summarized as follows:

a. Mechanical

1. Indications of degradation in the entire length of the sleeve must be evaluated against the sleeve plugging limit.

2. Indication of tube degradation of any type including a complete guillotine break in the tube between the bottom of the upper joint and the top of the lower roll expansion does not require that the tube be removed from service.

3. The tube plugging limit continues to apply to the portion of the tube in the entire upper joint region and in the lower roll expansion. As noted above the sleeve plugging limit applies to these areas also.

4. The tube plugging limit continues to apply to that portion of the tube above the top of the upper joint.

b. Laser Welded

1. Indications of degradation in the length of the sleeve between the weld joints must be evaluated against the sleeve plugging limit.

2. Indication of tube degradation of any type including a complete break in the tube between the upper weld joint and the lower weld joint does not require that the tube be removed from service.

3. At the weld joint, degradation must be evaluated in both the sleeve and tube.

4. In a joint with more than one weld, the weld closest to the end of the sleeve represents the joint to be inspected and the limit of the sleeve inspection.

5. The tube plugging limit continues to apply to the portion of the tube above the upper weld joint and below the lower weld joint.

F* tubes do not have to be plugged or repaired provided the remainder of the tube within the tubesheet that is above the F* distance is not degraded. The F* distance is equal to 1.79 inches and is measured down from the top of the tubesheet or the bottom of the roll transition, whichever is lower in elevation. Included in this distance is an allowance of 0.25 inch for eddy current elevation measurement uncertainty.

Steam generator tube inspections of operating plants have demonstrated the capability to reliability detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to 10 CFR 50.73 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision to the Technical Specifications, if necessary.

Attachment 4

Significant Hazards Consideration Evaluation

in support of the

Interim Plugging Criteria

Joseph M. Farley Nuclear Plant Unit 2

Steam Generator Tube Support Plate Interim Plugging Criteria

Significant Hazards Consideration Analysis

Introduction

In a letter dated January 29, 1992, the NRC Staff indicated that they were unable to approve a Technical Specification amendment concerning use of a steam generator tube support plate alternate plugging criteria in time for use in the Unit 2 spring of 1992 outage. The Staff indicated a willingness to discuss an interim plugging criteria for use in the Unit 2 Ninth Operating Cycle which would be more conservative than the current proposed alternate plugging criteria but less so than Farley's current Technical Specification. As a result, Southern Nuclear is proposing an interim plugging criteria, developed by adding additional conservatisms to the alternate plugging criteria.

Description of the Amendment Request

As required by 10 CFR 50.91(a)(1), an analysis is provided to demonstrate that the proposed license amendment to implement the interim plugging criteria for tube support plate elevations for Farley Unit 2 involves no significant hazards. The interim plugging criteria involves a correlation between eddy current bobbin probe signal amplitude (voltage) and the tube burst and leakage capability.

Specifically, crack indications with bobbin probe voltages less than or equal to 1.0 volt, regardless of indicated depth, do not require remedial action if postulated steam line break leakage can be shown to be acceptable. A sampling program would also be implemented to ensure other forms of degradation are not occurring at the tube support plates and that cracks are not being masked at tube support plates by other factors.

The proposed amendment would modify Technical Specification 3/4.4.6 "Steam Generators," and its associated bases, and Technical Specification 3/4.4.7 "Reactor Coolant System Leakage" . The steam generator plugging/repair limit will be modified to clarify that the appropriate method for determining serviceability for tubes with outside diameter stress corrosion cracking at the tube support plate is by a methodology that more reliably assesses structural integrity. The operational leakage requirement will be modified to reduce the total allowable primary-to-secondary leakage for any one steam generator from 500 gallons per day to 150 gallons per day.

The Technical Specifications and the associated Bases will indicate that these changes are applicable only for the Unit 2 Ninth Operating Cycle.

EVALUATION

Steam Generator Tube Integrity Discussion

In the development of the interim plugging criteria, R.G. 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and R.G. 1.83, "Inservice Inspection of PWR Steam Generator Tubes," are used as the bases for determining that steam generator tube integrity considerations are maintained within acceptable limits. R.G. 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 2, 14, 15, 31, and 32 by reducing the probability and consequences of steam generator tube rupture through determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service by plugging or repair. This regulatory guide uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code. For the tube support plate elevation degradation occurring in the Farley steam generators, tube burst criteria are inherently satisfied during normal operating conditions by the presence of the tube support plate. The presence of the tube support plate enhances the integrity of the degraded tubes in that region by precluding tube deformation beyond the diameter of the drilled hole. It is not certain whether the tube support plate would function to provide a similar constraining effect during accident condition loadings in Farley Unit 2. Therefore, no credit is taken in the development of the plugging criteria for the presence of the tube support plate during accident condition loadings. Conservatively, based on the existing data base, burst testing shows that the safety requirements for tube burst margins during both normal and accident condition loadings can be satisfied with bobbin coil signal amplitudes less than 6.2 volts, regardless of the depth of tube wall penetration of the cracking. R.G. 1.83 describes a method acceptable to the NRC staff for implementing GDC 14, 15, 31, and 32 through periodic inservice inspection for the detection of significant tube wall degradation.

Upon implementation of the interim plugging criteria, tube leakage considerations must also be addressed. It must be determined that the cracks will not leak excessively during all plant conditions. For the interim tube plugging criteria developed for the Farley Unit 2 steam generator tubes, no leakage is expected during normal operating conditions even with the presence of through-wall cracks. This is the case as the stress corrosion cracking occurring in the tubes at the support plate elevations in the Farley steam generators are short, tight, axially oriented macrocracks separated by ligaments of material. No leakage during normal operating conditions has been observed in the field for crack indications with signal amplitudes less than 7.7 volts. Relative to the expected leakage during accident condition loadings, the limiting event with respect to primary-to-secondary leakage is a postulated steam line break (SLB) event. Laboratory data for pulled tubes and model boiler specimens show limited leakage for indications under 10.0 volts during a postulated SLB condition.

Additional Considerations

The proposed amendment would preclude occupational radiation exposure that wou'd otherwise be incurred by plant workers involved in tube plugging or repair operations. The proposed amendment would minimize the loss of margin in the reactor coolant flow through the steam generator in LOCA analyses. The proposed amendment would avoid loss of margin in reactor coolant system flow and, therefore, assist in demonstrating that minimum flow rates are maintained in excess of that required for operation at full power. Reduction in the amount of tube plugging required can reduce the length of plant outages and reduce the time that the steam generator is open to the containment environment during an outage.

ANALYSIS (3 FACTOR TEST)

In accordance with the three factor test of 10 CFR 5C.92(c), implementation of the proposed license amendment is analyzed using the following standards and found not to: 1) involve a significant increase in the probability or consequences for an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety.

Conformance of the proposed amendment 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 Farley Unit 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Testing of model boiler specimens for free standing tubes at room temperature conditions show burst pressures as high as 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 30 volts. Burst testing performed on pulled tubes with up to 10 volt indications show burst pressures in excess of 5900 psi at room temperature. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the R.G. 1.121 criterion requiring the maintenance of a margin of three times normal operating pressure differential on tube burst if through-wall cracks are present. Based on the existing data base, this criterion is satisfied with bobbin coil indications with signal amplitudes less than 6.2 volts, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the data. The 1.0 threshold volt criteria provides an extremely conservative margin of safety to the structural limit considering expected growth rates of ODSCC at Farley. Alternate crack morphologies can correspond to 6.2 volts so that a unique crack length is not defined by a burst pressure to voltage correlation. However,

relative to expected leakage during normal operating conditions, no field leakage has been reported from tubes with indications with a voltage level of under 7.7 volts (as compared to the 1.0 volt proposed interim tube plugging limit).

Relative to the expected leakage during accident condition loadings, the accidents that are affected by primary-to-secondary leakage and steam release to the environment are Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station Auxiliaries, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. Of these, the Major Secondary System Pipe Failure is the most limiting for Farley Unit 2 in considering the potential for off-site doses. The offsite dose analyses for the other events which model primary-to-secondary leakage and steam release from the secondary side to the environment assume that the secondary side remains intact. The steam generator tubes are not subjected to a sustained increase in differential pressure, as is the case following a steam line break event. This increase in differential pressure is responsible for the postulated increase in leakage and associated offsite doses following a steam line break event. Upon implementation of the interim plugging criteria, it must be verified that the expected distribution of cracking indications at the tube support plate intersections are such that primary-to-secondary leakage would result in site boundary dose within the current licensing basis for Unit 2, 1 gallon per minute during a steam line break event. Data indicate that a threshold voltage of 2.8 volts would result in through-wall cracks long enough to leak at SLB conditions. Application of the proposed plugging criteria requires that the current distribution of a number of indications versus voltage be obtained during the Unit 2 Eighth Refueling Outage. The current voltage is then combined with the rate of change in voltage measurement to establish an end of cycle voltage distribution and, thus, leak rate during SLB pressure differential. If it is found that the potential SLB leakage for degraded intersections planned to be left in service exceeds 1 gallon per minute, then additional tubes will be plugged or repaired to reduce SLB leakage potential to 1 gallon per minute or less.

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 interim tube support plate elevation steam generator tube plugging criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions). The bobbin probe signal amplitude plugging criteria is established such that operational leakage or excessive leakage during a postulated steam line break condition is not anticipated.

SNC will implement a maximum leakage rate limit of 150 gpd per steam generator to help preclude the potential for excessive leakage during all plant conditions upon application of the plugging criteria. The R.G. 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of three against bursting at normal operating pressure differential. A voltage amplitude of 6.2 volts for typical OD SCC corresponds to meeting this tube burst requirement at the lower 95% uncertainty limit on the burst correlation. Alternate crack morphologies can correspond to 6.2 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at three times normal operating pressure differential and SLB conditions are about 0.42 inch and 0.84 inch, respectively. Normal leakage for these crack lengths would range from 0.11 gallons per minute to 4.5 gallons per minute, respectively, while lower 95% confidence level leak rates would range from about 0.02 gallons per minute to 0.6 gallons per minute, respectively.

An operating leak rate of 150 gpd will be implemented in application of the tube plugging limit. This leakage limit provides for detection of 0.4 inch long cracks at nominal leak rates and 0.6 inch long cracks at the lower 95% confidence level leak rates. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for SLB conditions at leak rates less than a lower 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

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

The use of the interim tube support plate elevation plugging criteria at Farley Unit 2 is demonstrated to maintain steam generator tube integrity commensurate with the requirements of R.G. 1.121. R.G. 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability of the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of OD SCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The most limiting effect would be a possible increase in leakage during a steam line break

event. Excessive leakage during a steam line break event, however, is precluded by verifying that, once the criteria is applied, the expected end of cycle distribution of crack indications at the tube support plate elevations would result in minimal, and acceptable primary to secondary leakage during all plant conditions and, hence, help to demonstrate radiological conditions are less than a small fraction of the 10 CFR 100 guideline.

In addressing the combined effects of LOCA + SSE on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to either the LOCA rarefaction wave and/or SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, a detailed leak-before-break analysis was performed and it was concluded that the leak-before-break methodology (as permitted by GDC 4) is applicable to the Farley Unit reactor coolant system primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed at Farley Unit 2 and were found to be of insuffici nt magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Farley Unit 2, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Based on recent analyses results, no tubes near wedge locations are expected to collapse or deform to the degree that secondary to primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a LOCA + SSE is expected to be less than that currently allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-

primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to tighten under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing the R.G. 1.83 considerations, implementation of the tube plugging criteria is supplemented by 100% inspection requirements at the tube support plate elevations having OD SCC indications, reduced operating leak rate limits, eddy current inspection guidelines to provide consistency in voltage normalization, and rotating pancake coil inspection requirements for the larger indications left in service to characterize the principal degradation mechanism as OD SCC.

As noted previously, implementation of the tube support plate elevation plugging criteria will decrease the number of tubes which must be taken out of service with tube plugs or repaired. The installation of steam generator tube plugs would reduce the RCS flow margin, thus implementation of the interim plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any bases of the plant Technical Specifications.

CONCLUSION

Based on the preceding analysis, it is concluded that using the TSP elevation bobbin coil probe signal amplitude interim steam generator tube plugging criterion for removing tubes from service or repairing tubes at Farley Unit 2 is acceptable and the proposed license amendment does not involve a Significant Hazards Consideration Finding as defined in 10 CFR 50.92.

Attachment 5

1. WCAP-12871, Revision 2 - J.M. Farley Units 1 and 2 SG Tube Plugging Criteria for ODSCC at Tube Support Plates (Proprietary).

2. WCAP-12872, Revision 2 - J.M. Farley Units 1 and 2 SG Tube Plugging Criteria for ODSCC at Tube Support Plates (Non-Proprietary).