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A Duke Energy Company

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November 17, 1999

U. S. Nuclear Regulatory Commission
Document Control Desk
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Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Proposed Technical Specification Change Related to Steam Generator Tubing Surveillance
(TSCR 99-12)

Reference: Letter, WR McCollum (Duke) to USNRC, Steam Generator Tube End Anomalies - Interpretation of Technical Specification 5.5.10, § 8, dated September 7, 1999

Technical Specification 5.5.10, § e.6, requires that steam generator tubes that exceed the repair limit be repaired by sleeving or rerolling, or be removed from service. Based on discussions with the NRC to resolve TS interpretation discrepancies described in the referenced letter, Duke Energy Corporation (Duke) has concluded Units 1 and 3 were operating contrary to the above requirements of TS 5.5.10 at 1610 hours on November 15, 1999. Since this condition would lead to a shutdown of Oconee Units 1 and 3, Duke held a conference call with the NRR and Region II staff at 1615 hours on November 15, 1999, requesting enforcement discretion. At 1705 hours on November 15, 1999, the NRC granted enforcement discretion for Units 1 and 3. The request for enforcement discretion did not apply to Unit 2 since it is currently shutdown for a refueling outage. Oconee committed to submit a proposed Technical Specification change applicable to all units to resolve this issue by November 17, 1999. This submittal fulfills that commitment.

A001

Pursuant to 10 CFR 50.90, Duke hereby requests an amendment to the Technical Specifications (TS) for Oconee Nuclear Station. This proposed change would modify the TS 5.5.10, § e.6, definition of Repair Limit, by adding a provision to exclude certain Steam Generator (SG) tube defect indications.

Some Oconee SG tubes exhibit Tube End Anomalies (TEAs) in Eddy Current (EC) data developed by the SG tube surveillance program. A TEA is an indication of a potential defect located at the SG tube end. To better characterize these TEAs, Duke has performed testing, analysis of SG EC data, and analysis of data from EC testing on mockups of the tube sheet. Duke has concluded that TEAs beyond the outer, primary, surface of the tube sheet clad surface that exhibit axial defects have little potential to develop primary to secondary leakage during normal operation or the limiting design basis event. Further, it is highly unlikely for such a SG tube end to become unserviceable per TS 5.5.10, § e.7, prior to the next inspection regardless of the indicated depth of any apparent axial defect.

Attachment 1 provides a proposed amendment to the ONS Technical Specifications. Attachment 2 provides a marked up copy of the current Technical Specifications. Attachment 3 provides the Technical Justification for the proposed amendment, and Attachment 4 provides the No Significant Hazards Consideration. The Environmental Impact Analysis is provided in Attachment 5.

The proposed Technical Specification amendment has been reviewed and approved by both the Plant Operations Review Committee and the Nuclear Safety Review Board. The UFSAR will be updated, as necessary, to reflect the changes associated with this submittal.

It is Duke's understanding that enforcement discretion will remain in effect until the staff completes its review and approves the proposed changes provided in this license amendment. Unit 2 is currently scheduled to enter Mode 4 from its current refueling outage on December 9, 1999. Thus, Duke requests prompt attention to this matter in order to prevent an unnecessary delay in the restart of Unit 2.

A copy of this application is being forwarded to the South Carolina Department of Health and Environmental Control for their review and, as appropriate, subsequent consultation with the staff.

If there are any questions regarding this submittal, please contact Larry Nicholson at 864-885-3292.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Attachments (5)

xc: Mr. D. E. LaBarge, ONRR
Project Manager

Mr. L. A. Reyes
Regional Administer, Region II

Mr. M. C. Shannon
Senior Resident Inspector

Mr. V. R. Autry
DHEC

AFFADAVIT

W. R. McCollum, Jr., being duly sworn, states that he is Site Vice President of Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this revision to the Oconee Nuclear Station License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., Site Vice President

Subscribed and sworn to before me the 17th day of November, 1999.

Notary Public: Robert C. Douglas

My Commission Expires: August 13, 2009
Date

Seal

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATONS

REVISED PAGES

Remove Pages	Insert Pages
5.0-17	5.0-17

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

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

3. Degraded Tube means a tube or a sleeve containing imperfections ≥ 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 repair limit. A tube or sleeve containing a defect is defective.

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness. Axial tube imperfections of any depth observed between the primary side surface of the tube sheet clad and the end of the tube are excluded from this repair limit.

   The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

   The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the Topical Report, BAW-2303, Revision 3.

7. Unserviceable describes the condition of a tube 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 5.5.10.d.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

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ATTACHMENT 2

TECHNICAL SPECIFICATONS

Marked Up Pages

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

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

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INSERT FROM FOLLOWING PAGE

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8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

Insert 1 to TS 5.5.10, § e.6, on page 5.0-17

Axial tube imperfections of any depth observed between the primary side surface of the tube sheet clad and the end of the tube are excluded from this repair limit.

ATTACHMENT 3

TECHNICAL JUSTIFICATION

Background

In early May 1998, operating experience data based on events at Arkansas Nuclear One (ANO) were received by Duke Energy Corporation (Duke). This information indicated that previous Eddy Current (EC) indications classified as TEAs had exhibited primary-to-secondary leakage at ANO, thus indicating they were in the pressure boundary. As a consequence, in 1998, Duke redefined TEAs and developed an analysis methodology and guidelines capable of distinguishing between anomalies and indications in the tube sheet clad.

Discussions of the above issue with NRC personnel at NRR and Region II led Duke to incorrectly conclude that a consensus existed regarding axial TEA indications identified between the tube end and primary surface of the tube sheet clad. Specifically, Duke concluded that these indications were not part of the pressure boundary and could therefore be excluded from the TS inspection requirements. On September 7, 1999, Duke submitted the suggested interpretation letter which said, in part:

> "...the portion of a SG tube end that extends beyond the top of the cladding is not part of the pressure boundary since it is beyond the point of exit from the SG secondary side. The SG tube end beyond the top cladding is therefore excluded from the SG tube inspections."

On November 10, 1999, in a telephone conference with the NRC staff, the NRC suggested that the above position may be contrary to the requirements of TSs. Specifically, TS 5.5.10, SG Tube Surveillance Program, says in part:

> e.6. <u>Repair Limit</u> means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable

prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

The NRC indicated that the above 40% repair criteria was applicable to the TEAs such that operation with the TEAs was contrary to TS. Oconee's TS require inspection of SG tubes from point of entry completely to the point of exit.

In a subsequent call with the NRC, on November 15, 1999, the NRC informed Duke that they had concluded that TS 5.5.10, § e.8 required the SG tubes to be inspected from point of entry completely to point of exit. Although Duke had inspected the SG tubes in Units 1, 2 and 3 completely from end to end, a methodology was established that allowed no repair of indications of potential SG tube defects between the tube end and the primary surface of the cladding.

Description Proposed TS Change

In order to remove the literal compliance ambiguity, the following change (italicized) is proposed to TS 5.5, Programs and Manuals, Section 5.5.10, SG Tube Surveillance Program:

e.6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness. *Axial tube imperfections of any depth observed between the primary side surface of the tube sheet clad and the end of the tube are excluded from this repair limit.*

The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet

area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the Topical Report, BAW-2303, Revision 3.

Technical Justification

Background

Inspections at plants with B&W designed steam generators have revealed indications near the ends of the expansion roll of the tubes' upper tube-to-tubesheet joints. These indications, originally defined as TEAs, were initially believed to be in the non-pressure boundary portion of the tube. During another B&W plant's outage of 1998, a reverse pressure bubble test identified leakage from two tubes, one in each generator. One of the leaking tubes at this plant had been identified as having a tube end anomaly during the previous inspection. Subsequent inspection of the leaking tubes using updated EC analysis guidelines indicated that some of these indications were in the portion of the tube adjacent to the tubesheet cladding and therefore are within the pressure boundary. The indications were identified as mixed mode cracking (axial and circumferential) in one steam generator and an axial crack in the other steam generator. Both of these indications were in the upper roll region below the primary surface of the upper tubesheet clad. These crack-like indications located between the primary face of the tubesheet clad and the tubesheet clad to carbon steel interface are referred to as Tube End Cracks (TECs). These types of indications have been identified in NRC Information Notice 98-27, "Steam Generator Tube End Cracking" (dated July 24, 1998). It is noted that no laboratory examination data on tube-to-tubesheet rolled joints is available which would verify the indications are actually cracks. The indications are believed to be cracks based on the EC response and bubble test results.

ATTACHMENT 3

In 1998, based on re-analysis of EC data, some TEA indications were confirmed to extend below the primary surface of the tube sheet clad and were in service at Unit 1 and Unit 3. These indications were re-classified as TEC indications. In 1998, Duke Energy Corporation requested and received both a Notice of Enforcement Discretion and a subsequent TS amendment from the NRC to allow the affected plants to operate the remainder of their current fuel cycles with TEC indications in service. The NRC granted these petitions with the commitment that the TEC indications would be repaired or plugged during the next outage of sufficient duration. All known TEC indications at Unit 1 and Unit 3 have since been repaired.

Characteristics of Tube End Cracks (TECs)

Based on a review of the EC data for tubes with TEC indications, the indications are typically characterized as crack-like and axially oriented. They are located between the primary surface of the tubesheet clad and the tubesheet clad to carbon steel interface. Circumferential indications have also been identified as well as a small number of volumetric indications. Multiple axial indications and combinations of axial and circumferential indications have also been identified.

Based on the EC data, the TECs are believed to initiate on the inside surface of the tube. They are typically short, axially oriented, and located in the rolled portion of the tube near the heat affected zone created by the tube-to-tubesheet weld. While no laboratory examination data on TECs is available, these indications have been verified as through wall cracks based on bubble tests performed at other B&W plants. The indications are believed to be cracks based on the EC response. The rolling process and weld create residual stress that may make the material more susceptible to Primary Water Stress Corrosion Cracking (PWSCC). For this reason, it is believed that the TECs are PWSCC initiated. The Oconee steam generators have been bubble tested during the last five refueling outages. Through wall TECs were detected in the Oconee Unit 1A Once Through Steam

Generator (OTSG), however, it should be noted that this OTSG has a flush welded upper tubesheet.

Characteristics of Tube End Anomalies (TEAs)

TEAs, as defined after the 1998 development of a TEC definition, are axial inside diameter (ID) indications located between the tube end and the primary face of the tubesheet clad. No laboratory examination data on TEAs is available and no TEAs have been identified with bubble testing. Therefore, these indications have not been confirmed to be actual degradation in the tube. The rolling process and weld create residual stress that may make the material more susceptible to Primary Water Stress Corrosion Cracking (PWSCC). For this reason, it is possible that the TEAs are PWSCC initiated. The Oconee steam generators have been bubble tested during the last five refueling outages for a total of ten tests. Leakage has not been identified from any TEA indication.

Safety impact of TEA indications

Based on industry experience of Inconel 600 weld metal, it has been concluded that the tube-to-tubesheet weld (wire type 82T) is not likely to crack and that the weld would not be affected by the TEAs. If the tubing next to the seal weld is cracked due to PWSCC, the crack growth should slow as the remaining stresses from the weld process are relieved. It should be noted the steam generators were full bundle stress relieved after the seal weld was installed. A crack would not be expected to penetrate into the weld material. History of PWSCC failures of Inconel 600 material (control rod drive nozzles and pressurizer nozzles) in pressurized water reactors shows that the cracks typically initiate on the ID of the nozzle in the heat affected zone (created by the application of the weld) and propagates through the nozzle wall. The crack would initiate at the highest stress location and follows the stress profile. A crack would not initiate at the weld. Based on industry experience, stress corrosion cracking (SCC) susceptibility of Inconel 600 materials can be ranked as follows:

ATTACHMENT 3

1) Wrought Inconel 600 is most susceptible to SCC
2) Alloy 182 (wire type) weld metal is next
3) Alloy 82 (wire type) weld metal is least

The rankings provide additional evidence that the Inconel 600 tube is more likely to crack than the alloy 82 weld material.

The only observed cracking of Inconel 600 weld material in pressurized water reactors has occurred in repair welds. These repair welds were installed as part of the replacement of an Inconel 600 nozzle that had failed due to PWSCC with an alloy 690 nozzle. For the repairs, portions of the original weld material were left in place. Although the cause of these weld failures is not known, it is believed the source may have been poor welds.

This conclusion that the original weld does not crack is also supported by bubble tests of steam generator tubes performed at various plants. The tests clearly showed that the leakage was from a crack in the tube wall in or near the heat affected zone caused by the weld. These indications were below the primary face of the tubesheet clad and in the tube wall adjacent to the clad. There was no evidence of leakage from the weld and no evidence of cracks in the weld.

The location of a TEA precludes the possibility of burst since a differential pressure cannot be developed across the tube end above the primary face of the clad. Therefore, the primary safety concern for primary-to-secondary leakage is postulated leakage during an accident condition. The axial loads and increased pressure differential during a postulated accident condition have the potential to increase the primary-to-secondary leakage, compared to the normal operating conditions. This can only occur if the TEAs that are left in service experience indication growth such that they extend beyond the primary surface of the tubesheet clad at some point during the operating cycle.

TEA indications are not predicted to contribute to primary-to-secondary leakage since the weld will remain functional and the TEA does not extend into the clad. A direct path

from the primary side to the secondary side does not exist. This is supported by the ten nitrogen bubble tests that have been performed at Oconee with no indication of leakage from a TEA.

Primary-to-secondary leakage during normal operation is monitored according to the plants' TSs to ensure that any leakage remains less than the acceptable limit. Total operational leakage from TEAs left in service is zero.

More than 2500 TEAs in the three Oconee Units have been reviewed for growth beyond the primary face of the tubesheet clad. No TEA has been identified that grew into the cladding region of the tubesheet during one cycle of operation. Side by side comparisons of TEAs in successive outages indicate no apparent growth in length. Therefore, growth into the clad and completely throughwall is not expected.

BAW-2346P Rev. 0 Alternate Repair Criteria for Tube End Cracking in the Tube-To-Tubesheet Roll Joint of Once Through Steam Generators was reviewed for comparable growth studies for TECs. If the TEAs are assumed to be due to PWSCC with the stress component being generated by the seal weld process, it is reasonable to assume that the growth rate of a TEA would be similar to that of a TEC. TEC growth rates were assessed in BAW-2346P, Rev. 0. The growth rate indicated in the report is 0.0135 inches per 1.37 EFPY in one steam generator and 0.0 growth in another steam generator. The data is from another B&W plant with a similar operating history. This data supports a zero length growth rate.

Since the apparent growth rate is zero and no leakage has been observed during nitrogen bubble testing, the leakage from a TEA at MSLB conditions is expected to be zero even if some growth into the cladding is observed. To further verify this conclusion, available information was input into the Tubeworks computer code to assess TEAs for MSLB leakage.

The leak rate calculations in Tubeworks are based on the EPRI leak rate equations and standard fracture mechanics

calculations of crack opening areas. The leak rate equations, describing flow as a function of crack opening areas and crack lengths, are used for both axial and circumferential cracks. The calculated leak rates are good upper bounds to actual data on stress corrosion cracks.

The EPRI leak rate equations were developed from leak rate calculations performed with the PICEP two phase flow algorithm. These equations are described in "Depth Based Structural Analysis Methods for Steam Generator Circumferential Indications" EPRI TR-107197-P1, November, 1997. Crack opening area calculations are based on results in the EPRI Ductile Fracture Handbook, NP-6301-D, June 1989, and experimental crack opening area measurements. Measurements and theoretical calculations showed excellent agreement. The calculated leak rates are an upper bound to leak rates for stress corrosion cracks.

The conservative nature of the leak rate calculations is particularly evident at short crack lengths. For fatigue cracked specimens, it is very difficult to observe leakage from cracks less than about 0.15 inches in throughwall length at normal operating conditions. At throughwall lengths less than about 0.05 inches, leakage does not exist for steam line break or normal operating conditions. This is supported by the data in PWR Steam Generator Tube repair Limits: Technical Support Documents for Expansion Zone PWSCC in Roll Transitions", EPRI NP-6864-L and related studies such as "Evaluation of Leak and Burst Characteristics of Roll Transitions containing PWSCC", EPRI RP S406-7. For very short crack lengths (<0.1 inches), the reasonable engineering approach is to assume no leakage at main steam line break conditions.

The leak rates predicted using the EPRI methodologies are good upper bound estimates of leakage through cracks in the tubing freespan. The calculated leak rates are even more conservative since the effect of the tubesheet roll is not modeled in the calculations. The presence of the tubesheet roll will decrease the actual leakrate through a flaw. For comparison purposes the leakrate through a 0.5 inch long crack was calculated using EPRI methodologies and compared

to the leak test data in BAW-2346P Rev. 0. This report contains leakrate information from testing simulated tubesheet flaws. Using the tolerance limit material properties at normal operating temperatures, the calculated leak rate for a 0.5 inch freespan crack at 2640 psi is 17 gpm. This is compared to the maximum observed leakage through a 0.5 inch simulated tubesheet flaw of 0.00753 gpm as reported in BAW-2346P, Rev. 0.

Assuming a 0.0135 inch growth rate, 100 % throughwall, and no leakage reduction due to the tube to tubesheet roll, the leakage through these flaws can be conservatively calculated. Using the lower tolerance limit material properties at normal operating temperatures, this results in a MSLB leakrate of 0.17 gpm for 10,000 indications 0.0135 inches long.

Due to the conservative nature of the leak rate calculations, the reduction in leakage due to the tubesheet roll, and considering the fact that no TEAs have exhibited growth into the clad, the leakage from TEAs is expected to be zero during MSLB conditions. It has been noted that even for fatigue cracked specimens, it is very difficult to observe leakage from cracks less than about 0.15 inches in throughwall length at normal operating conditions. At throughwall lengths less than about 0.05 inches, leakage does not occur for steam line break and normal operating conditions. This supports the conclusion that TEAs will not leak at steam line break conditions even in the unlikely event that the TEAs experience growth such that they extend beyond the primary surface of the tubesheet clad at some point during the operating cycle.

Summary

TEAs are indications in the tube end above the primary face of the clad. These indications are axial in nature. The true nature of these indications has not been determined but it is possible that the most likely cause is PWSCC.

ATTACHMENT 3

TEAs that are left in service were evaluated for contribution to primary-to-secondary leakage during a postulated worse case accident condition. This contribution is expected to be zero. Therefore, operational limits on the number of tubes with TEAs, or a limit on leakage from tubes having TEAs, is unnecessary. Burst and failure by bending are precluded by the location of TEAs.

ATTACHMENT 4

SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

No. This evaluation addresses the potential effects of not applying the repair criteria to steam generator tubes. Tube End Anomalies (TEAs) are eddy current indications between the outer tubesheet cladding surface and the tube end. As described in the Technical Justification, operating with some steam generator tubes with Tube End Anomalies (TEAs) in Units 1, 2 and 3 does not increase the probability of any accident evaluated in the Safety Analysis Report (SAR) because this condition is not an accident initiator. Operation with TEAs will not adversely affect the ability to mitigate any FSAR described accident since it has been demonstrated that indications in areas defined as TEAs do not represent a risk of pressure boundary leakage. Therefore, the leakage requirements for steam generator integrity for the most limiting event, a Main Steam Line Break (MSLB), are satisfied with no compensatory actions required.

There is no physical change to the plant Structures, Systems or Components (SSCs) or operating procedures. Neither electrical power systems, nor important to safety mechanical SSCs will be adversely affected. The steam generators have been evaluated as operable for normal and accident conditions. There are no shutdown margin, reactivity management, or fuel integrity concerns. There is no increase in accident initiation likelihood, therefore analyzed accident scenarios are not impacted.

ATTACHMENT 4

There is no adverse impact on containment integrity, radiological release pathways, fuel design, filtration systems, main steam relief valve setpoints, or radwaste systems.

There is no increase in accident initiation likelihood or consequences, therefore analyzed accident scenarios are not impacted.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

There is no increased risk of unit trip, or challenge to the reactor protection system or other safety systems. There is no physical effect on the plant, i.e., none on Reactor Coolant System (RCS) temperature, boron concentration, control rod manipulations, core configuration changes, and no impact on nuclear instrumentation. There is no increased risk of a reactivity excursion. No new failure modes or credible accident scenarios are postulated from this activity. The MSLB scenario has been evaluated and the potential for damage to the steam generator tubes is not increased.

(3) Involve a significant reduction in a margin of safety.

No function of any important-to-safety SSC will be adversely affected or degraded as a result of continued operation. No safety parameters, setpoints, or design limits are changed. There is no adverse impact to the nuclear fuel, cladding, RCS, or required containment systems. Therefore, the margins of safety as defined in the bases to any Technical Specifications are not reduced as a result of this change.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

ATTACHMENT 5

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10 CFR 51.22 (b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the No Significant Hazards Consideration evaluation that is contained in Attachment 4.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   The proposed amendment will not significantly change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

   The proposed will not significantly increase the individual or cumulative occupational radiation exposure.

In summary, the proposed amendment request meets the criteria set forth in 10 CFR 51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.