Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

M. S. Tuckman
Senior Vice President
Nuclear Generation
(704)382-2200 Office
(704)382-4360 Fax

DUKE POWER

September 30, 1994

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

Subject: McGuire Nuclear Station
Docket Nos. 50-369, 50-370
Replacement Steam Generator Proposed Tech Spec Amendment

Pursuant to 10 CFR 50.90, attached are proposed revisions to the Technical Specifications related to the replacement of the steam generators at McGuire Units 1 and 2. Revisions to topical reports DPC-NE-3002, FSAR Chapter 15 System Transient Analysis Methodology, and DPC-NE-3000, Thermal-Hydraulic Transient Analysis Methodology, which were required for analyses supporting these changes were submitted on July 18, 1994 and August 9, 1994 respectively.

The following are attached: Attachment 1, Accident Evaluation for the Steam Generator Replacement; Attachment 2, Marked up Technical Specification Changes; Attachment 3, Technical Justification; Attachment 4, No Significant Hazards Evaluation; and Attachment 5, Replacement Steam Generator Topical Report. The Replacement Steam Generator Topical Report is attached for information during the review of the Technical Specification changes related to the replacement steam generators.

The requested changes to the Technical Specifications are marked up on the current pages. On July 18, 1994, a proposed amendment for both the Catawba and McGuire Nuclear Stations was submitted which split Technical Specifications into two separate unit-specific volumes. Duke anticipates approval of the split prior to approval of the replacement steam generator submittal. When approval of the July 18, 1994 submittal is received, new marked up Technical Specification pages will be submitted. This will resolve any difficulties related to a Unit's Technical Specifications applicability, since each Unit will be contained in a separate volume.

Duke Power is requesting review and approval of these changes by the beginning of the McGuire Unit 1 replacement outage, which is currently scheduled to start November 1, 1995. Duke is asking that the McGuire Unit 2 changes be reviewed and approved concurrently with the McGuire and Catawba (submitted under separate cover, September 30, 1994) Unit 1 changes, and that they become effective upon replacement of the steam generators. This request reflects current replacement

9410070181 940930
PDR ADOCK 05000369
P PDR

A001
1/1

schedules, Duke Power will keep the staff informed of any changes in desired approval dates. If we can be of assistance in your review, please call Mary Hazeltine at (704) 382-6111.

Pursuant to 10 CFR 50.91(b)(1), a copy of this amendment request has been provided to the appropriate North Carolina state officials.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

cc: Mr. V. Nerses, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 14H25, OWFN
Washington, D. C. 20555

Mr. S. D. Ebneter, Regional Administrator
U.S. Nuclear Regulatory Commission - Region II
101 Marietta Street, NW - Suite 2900
Atlanta, Georgia 30323

Mr. G. F. Maxwell
Senior Resident Inspector
McGuire Nuclear Station

R. C. Jones, Branch Chief
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 8 E23, OWFN
Washington, D. C. 20555

Dayne Brown, Director
Division of Radiation Protection
P. O. Box 27678
Raleigh, North Carolina 27611-7687

Mike S. Tuckman, being duly sworn, states that he is Senior Vice President of Duke Power Company; that he is authorized on the part of said company to sign and file with the U. S. Nuclear Regulatory Commission these revisions to the McGuire Nuclear Station License Nos. NPF-9 and NPF-17; and, that all statements and matters set forth therein are true and correct to the best of his knowledge.

M. S. Tuckman

Mike S. Tuckman

Subscribed and sworn to before me this 30th day of September, 1994.

Mary P. Nelms

Notary Public

My Commission expires: JAN 22, 1996

bxc: G. A. Copp
T. R. Niggel
G. B. Swindlehurst
C. F. Robinson
S. A. Gewehr
J. E. Snyder
D. B. Mayes
MC 1201.37-28
File: GS-801.01

Attachment 1

I. Introduction

Currently, the steam generators in place at both McGuire units are Westinghouse Model "D" type preheat steam generators. The tube degradation levels in the generators has affected the reliability of the units. Therefore, these generators are scheduled to be replaced with feedring steam generators designed by Babcock & Wilcox International. The major design differences in the feedring steam generator with respect to the preheat design include the following.

- There are approximately 2000 more tubes of a slightly smaller diameter.
- The tube bundle is about 8 feet taller.
- The SG liquid mass at full power is approximately 20,000 lbm greater.

The above steam generator design differences result in the following thermal-hydraulic changes:

- The total primary system volume is increased by about 10%.
- The effective tube bundle heat transfer area is increased by approximately 60%.
- The full power programmed Tavg for McGuire is reduced by about 3°F.

In order to determine the effects of the steam generator replacement and to ensure that the thermal performance during hypothetical incidents is not degraded, each FSAR accident analysis has been evaluated.

II. Transients Reanalyzed

The following thermal-hydraulic system transients are reanalyzed in order to ensure that the acceptance criteria continue to be met for any cases in which the feedring steam generator might result in a more severe challenge to the criteria. All analyses are performed consistent with NRC-approved methodology or, in the case of items A, B, K and N below, methodology submitted for approval (DPC-NE-3004, Mass and Energy Release and Containment Response Methodology, DPC-NE-3002, FSAR Chapter 15 System Transient Analysis Methodology, and DPC-NE-3000, Thermal-Hydraulic Transient Analysis Methodology).

A. <u>Mass and energy release for postulated loss of coolant accidents inside containment (6.2.1.3)</u>

This event is analyzed to ensure that the peak containment pressure limit is not exceeded. Since the Reactor Coolant System volume will be greater, the total mass released into containment will be greater. In addition, during the depressurization of the RCS, the steam generators actually function as heat sources. Since the

feedring steam generator full power liquid mass is greater than that of the Model D steam generators, the total energy available for removal by the RCS is increased. Both of these effects have the potential to yield more severe mass and energy release results.

B. Mass and energy release for postulated secondary system pipe ruptures inside containment (6.2.1.4)

This event is analyzed to ensure that the peak containment temperature limit is not exceeded. A key phenomenon in this analysis is SG tube bundle uncovery, since this initiates the release of superheated steam into containment. Since the feedring steam generator design has tubes that are significantly taller than those in the Model D generators, the potential exists for earlier bundle uncovery.

C. Feedwater system malfunction causing an increase in feedwater flow (15.1.2)

This ANS Condition II event is analyzed to show that DNB does not occur. The transient involves an increase in core power resulting from an overcooling by the secondary system. The impact of the increased heat transfer area of the feedring steam generator is investigated in this reanalysis.

D. Excessive increase in secondary steam flow (15.1.3)

This ANS Condition II event is analyzed to show that DNB does not occur. The transient involves an increase in core power resulting from an overcooling by the secondary system. The impact of the increased heat transfer area of the feedring steam generator is investigated in this reanalysis.

E. Inadvertent opening of a steam generator relief or safety valve (15.1.4)

This ANS Condition II event is reanalyzed to confirm that the transient response is bounded by the steam system piping failure event.

F. Steam system piping failure (15.1.5)

This ANS Condition III & IV event is analyzed to the more stringent Condition II criterion of ensuring that DNB does not occur. This transient involves an increase in core power resulting from an overcooling by the secondary system. The impact of

the increased heat transfer area of the feedring steam generator is investigated in this reanalysis.

G. Turbine trip (15.2.3)

This ANS Condition II event is analyzed to show that peak primary and secondary system pressures do not exceed the applicable limits. The increased heat transfer area of the feedring steam generator improves the ability of the secondary system to remove primary system heat and, therefore, potentially results in a more severe secondary side pressurization.

H. Loss of non-emergency AC power to the station auxiliaries (15.2.6)

This ANS Condition II event is reanalyzed to demonstrate the adequacy of the natural circulation cooling in the modified reactor coolant loop configuration.

I. Loss of normal feedwater flow (15.2.7)

This ANS Condition II event is reanalyzed to confirm that the transient response is bounded by the turbine trip event.

J. Feedwater system pipe break (15.2.8)

This ANS Condition IV event is analyzed to demonstrate the capability of the secondary system to effectively cool the reactor core. While the increase in heat transfer area would tend to improve the transient results, other factors, such as the relocation of the main feedwater nozzle and the removal of the nozzle flow restrictor, necessitate the reanalysis.

K. Reactor coolant pump shaft seizure - locked rotor (15.3.3)

This ANS Condition IV event is analyzed to show that the peak primary system pressure does not exceed the applicable limit and to determine the percentage of fuel rods that experience DNB. Although the results of the transient analysis are insensitive to the secondary system, a proposed change to the transient analysis methodology (DPC-NE-3002, Rev. 1) necessitates the reanalysis of the DNB transient.

In addition, the radiological consequences are reanalyzed due to the differences in the thermal-hydraulic parameters of the feedring steam generator. Specifically, since the feedring generator design has tubes that are significantly taller than those in the Model D generators, the potential exists for a longer period of tube bundle uncovery, which negatively impacts the offsite dose calculation.

L. Steam generator tube rupture (15.6.3)

This ANS Condition IV event is analyzed to show that a) DNB does not occur, b) the calculated offsite doses do not exceed the acceptance criterion, and c) SG overfill is avoided. When the tube rupture is no longer covered with liquid, a significant reduction in the iodine partition factor occurs; therefore, tube bundle uncovery is an important phenomenon in the offsite dose analysis. Since the feedring steam generator design has tubes that are significantly taller than those in the Model D generators, the potential exists for a longer period of tube bundle uncovery. Other factors associated with the feedring steam generator which potentially impact the transient results are the reduced tube diameter and the revised SG level setpoints.

M. Loss of coolant accidents (15.6.5)

A LOCA analysis, applicable to McGuire Units 1 & 2, has been performed by B&W Nuclear Technologies (BWNT). The analysis supports operation of the Duke units with the feedring steam generators. Methodology employed in the analysis is in accordance with 10CFR50.46 and 10CFR50 Appendix K and is documented in topical reports BAW-10174, Revision 2 and BAW-10168P, Revision 3. The LOCA evaluation considered both large and small breaks.

N. Postulated secondary system pipe rupture outside containment

This event is analyzed to ensure that the Doghouse equipment qualification temperature limit is not exceeded. A key phenomenon in this analysis is tube bundle uncovery, since this initiates the release of superheated steam. Since the feedring steam generator design has tubes that are significantly taller than those in the Model D generators, the potential exists for earlier bundle uncovery.

III. Transients Reanalyzed For Offsite Dose Only

The radiological consequences are reanalyzed for the following events due to the differences in the thermal-hydraulic parameters of the feedring steam generator.

Specifically, since the feedring generator design has tubes that are significantly taller than those in the Model D generators, the potential exists for a longer period of tube bundle uncovery, which negatively impacts the offsite dose calculation. The two events listed are Chapter 15 transients for which fuel failures are postulated to occur.

A. Single RCCA withdrawal (15.4.3.d)

B. Spectrum of RCCA ejection accidents (15.4.8)

IV. Transients Not Reanalyzed

For the following transients reanalysis is not required, since either a) the analysis is unaffected by the steam generator replacement, b) any changes will not adversely impact the analysis results, or c) the transient is bounded by a more limiting transient of the same ANS Condition which is being reanalyzed.

A. Peak Reverse Differential Pressure, Containment Subcompartment & Minimum Containment Pressure Analyses (6.2.1.1, 6.2.1.2, & 6.2.1.5)

Three types of containment analyses presented in the FSAR are not reanalyzed. The first type is the short-term or blowdown peak containment pressure analysis following a LOCA or SLB, including subcompartment pressurization analyses. These analyses simulate the compression of the initial air mass in containment immediately following the pipe rupture and lasting seconds. The first few seconds of the LOCA or SLB are not affected by steam generator replacement, and no reanalysis is necessary. The second type of FSAR containment analysis that is not reanalyzed is the peak reverse differential pressure analysis. The results of the analysis as shown in the FSAR maintain a margin of a factor of six to the acceptance criteria. The replacement steam generators would only introduce a small change in these results. Due to the large margin in the current analysis results no reanalysis is necessary. The third type of FSAR containment analysis that is not reanalyzed is the minimum containment backpressure analysis. This analysis is used as a boundary condition for the LOCA peak clad temperature analysis. Since steam generator replacement will result in an increase in the primary coolant volume, the minimum containment backpressure will be higher. Therefore the current minimum backpressure analysis will remain valid, and no reanalysis is necessary. The results of and conclusions made based on these FSAR analyses are not affected by steam generator replacement.

B. Feedwater system malfunction causing a reduction in feedwater temperature (15.1.1)

This ANS Condition II event is bounded by the increase in feedwater flow event. Therefore, a quantitative analysis of this transient is not required.

C. Loss of external load (15.2.2)

This ANS Condition II event is bounded by the turbine trip event. Therefore, a quantitative analysis of this transient is not required.

D. Inadvertent closure of main steam isolation valves (15.2.4)

This ANS Condition II event is bounded by the turbine trip event. Therefore, a quantitative analysis of this transient is not required.

E. Partial loss of forced reactor coolant flow (15.3.1)

The transient results of this ANS Condition II event, which is analyzed to show that DNB does not occur, are insensitive to the secondary system. None of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

F. Complete loss of forced reactor coolant flow (15.3.2)

The transient results of this ANS Condition II event, which is analyzed to show that DNB does not occur, are insensitive to the secondary system. None of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

G. Reactor coolant pump shaft break (15.3.4)

This ANS Condition IV event is bounded by the locked rotor event. Therefore, a quantitative analysis of this transient is not required.

H. Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition (15.4.1)

The transient results of this ANS Condition II event, which is analyzed to show that peak primary system pressure does not exceed the applicable limits and that DNB does not occur, are insensitive to the secondary system. None of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

I. Uncontrolled RCCA bank withdrawal at power (15.4.2)

The transient results of this ANS Condition II event, which is analyzed to show that peak primary and secondary system pressures do not exceed the applicable limits and that DNB does not occur, are insensitive to the secondary system. None of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

J. RCCA misoperation (15.4.3)

The transient results of these ANS Condition II and III events, which are analyzed to show that DNB does not occur or to determine the percentage of fuel rods that experience DNB, are insensitive to the secondary system. For those analyses in which the secondary side is modeled, none of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid. Note the offsite dose calculation for the Single RCCA withdrawal event (15.4.3.d) is being revised (see Section III above).

K. Startup of an inactive reactor coolant pump at an incorrect temperature (15.4.4)

The transient results of this ANS Condition II event, which is analyzed to show that DNB does not occur, are insensitive to the secondary system. None of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

L. CVCS malfunction that results in a decrease in boron concentration in the reactor coolant (15.4.6)

This ANS Condition II event is analyzed to ensure that the dilution is terminated prior to a loss of shutdown margin. The results of the transient analysis are insensitive to the secondary system. In addition, the increase in RCS volume due to the greater number and length of the steam generator tubes will have a beneficial impact on the transient results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

M. Spectrum of RCCA ejection accidents (15.4.8)

This ANS Condition IV event is analyzed to show that the peak fuel pellet enthalpy and the peak primary side pressure do not exceed the acceptance criteria, and to determine the percentage of fuel pins exceeding the DNBR limit. Due to the rapid nature of the transient, the secondary side is not modeled. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNBR results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid. Note the offsite dose calculation for the RCCA ejection event is being revised (see Section III above).

N. Inadvertent operation of ECCS during power operation (15.5.1)

The results of this ANS Condition II event, which is analyzed to show that neither pressurizer overfill nor DNB occur, are insensitive to the secondary system. For the analysis in which the secondary side is modeled, none of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

O. <u>Inadvertent opening of a pressurizer safety or relief valve (15.6.1)</u>

The results of this ANS Condition II event, which is analyzed to show that DNB does not occur, are insensitive to the secondary system. None of the steam generator level RPS or ESFAS trip functions are challenged, and steam generator mass remains relatively constant throughout the event. In addition, the reduction in the RCS loop average temperature will have a beneficial impact on the transient DNB results. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

Attachment 2

Technical Specification Changes

Specification	Description of Change
Table 2.2-1	revise low-low steam generator water level reactor trip setpoint
Table 3.3-4	revise high-high steam generator water level setpoint for turbine trip and feedwater isolation revise low-low steam generator water level setpoint for auxiliary feedwater actuation
3/4.4.4	delete repair methods which will no longer be applicable after the replacement of the steam generators and clarify initial surveillances
3/4.4.6.2	change primary-secondary leakage limit
Table 3.7-3	reduce steam line safety valve lift settings
5.4.2	revise Reactor Coolant System Volume
6.9.1.9	update revision numbers on topical reports