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

Hal B. Tucker
Senior Vice President
Nuclear Generation
(704)382-4531

DUKE POWER

November 13, 1992

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

Subject: McGuire Nuclear Station Units 1 and 2
Docket Nos. 50-369, 370
Catawba Nuclear Station Unit 1
Docket Nos. 50-413
Methodology for Analysis of the Primary Coolant Loops for
Steam Generator Replacement, Supplement 1

Gentlemen:

Our original letter dated October 15, 1992, described the proposed methodology for the analysis of the primary coolant loops at McGuire Nuclear Station Units 1 & 2 and Catawba Unit 1 for steam generator replacement. Since the B & W replacement steam generators have a slightly higher center of gravity and a greater mass than the original Westinghouse steam generators, the NSSS primary coolant loops must be reanalyzed.

In a telecon on October 28, 1992, with your Mr. R. E. Martin and Mr. A. Lee, it was requested that certain clarifications be made to the original letter. Please find Attachment 1 which contains the clarifications incorporated into the original letter along with references as indicated on page 5. In addition, the questions with Duke's responses during the course of the conversation are included on pages 6-9.

As requested in our previous letter in support of our current steam generator replacement schedule, a response indicating acceptance is desired from the NRC by December 1, 1992, since the development of analysis models are already underway by B & W Nuclear Services. Should there be any questions concerning this proposal or if additional information is required, please contact David V. Ethington at (704) 382-6033.

Very truly yours,

D L Rehn

D. L. Rehn, General Manager
Nuclear Services

A001
1/1

dve/paramets1

Attachments

xc (w/att):

S. D. Ebneter
Regional Administrator, Region II

T. A. Reed, ONRR

R. E. Martin, ONRR

P. K. Van Doorn
Senior Resident Inspector (MNS)

W. T. Orders
Senior Resident Inspector (CNS)

Attachment 1

Duke Power Company
Methodology for Analysis of the Primary Coolant Loops at McGuire Nuclear Station Units 1&2 and Catawba Unit 1 for Steam Generator Replacement

Introduction

The existing Westinghouse model D2/D3 steam generators installed in McGuire Nuclear Station Units 1 & 2 and Catawba Unit 1 are suffering from tube degradation problems. Duke Power has decided that the most cost-effective solution to these problems is to replace the steam generators in each unit. Accordingly, an order has been placed with B&W International for twelve replacement steam generators. The B&W replacement steam generators have a slightly higher center of gravity and a greater mass than the original Westinghouse steam generators. These changes require that the NSSS primary coolant loops be reanalyzed. The reanalysis effort consists of a parametric study of the reactor coolant system response before and after steam generator replacement. The parametric study will be performed by Babcock & Wilcox Nuclear Services Inc. (BWNS) The intent of this parametric study is to show that the original design basis analysis of the reactor coolant loop is conservative and therefore is still valid. A new design basis will not be created for the reactor coolant loop.

Reactor Coolant Loop Analysis

The loading analysis of the primary coolant loop will be divided into two phases with each using a different loop model. The first phase involves constructing a loop model with the original steam generator similar to that used by Westinghouse. Gravity and seismic loads will be applied to this model consistent with those used by Westinghouse. The stiffnesses of the component supports provided to Westinghouse for the original loop analysis will be used in this model. Seismic excitation will be provided by the floor response spectra at the various elevations at which the NSSS component supports are attached to the Reactor Building Interior Structure. Structural damping in the NSSS math model will be the same as that used by Westinghouse in the original analysis. The analytical results from this model will be used to "benchmark" the analysis assumptions and input data by comparison of the output results with the original Westinghouse results. The intent of this analysis is to show that the structural properties of the reactor coolant loop model (mass, stiffness, and boundary conditions) and the analytical techniques applied to it will give results comparable to those provided by Westinghouse before the next phase of the NSSS analysis. Exact duplication should not be expected due to differences in modeling techniques, analytical methods, computer codes, etc. The loop model will be checked for only one of the three units which will undergo steam generator replacement. This will be sufficient to validate the analysis approach as the three reactor coolant systems are nearly (but not exactly) identical.

The second phase of the analysis involves modification of the math model constructed in phase one to incorporate the replacement steam generators. In this model, more advanced analytical techniques will be used. The NSSS primary coolant loop will be linked to the structural model of the Reactor Building Interior Structure using springs which represent the individual component supports. Seismic excitation will be

provided by the ground response[1] spectra rather than from the floor response spectra at the various elevations at which the component supports are attached to the Interior Structure. This will reduce the seismic input through the elimination of some of the effects of spectra broadening in floor response spectra.[2] ASME Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2 and 3 Piping" will be used as allowed by Regulatory Guide 1.84 to reduce the seismic input into the primary coolant loop even further. This code case was previously evaluated for McGuire and Catawba and is included in the plant FSARs (Section 3.7.1.3) to provide acceptable alternative damping values for piping.[3] Steady state thermal loads and gravity loads reflecting the new steam generator will also be considered.

The analysis of the phase two model will be performed using all applicable loadings and load combinations. New displacements, forces and moments, piping stresses, and response spectra will be calculated. The static and seismic analysis results will be compared to those from the phase one model with the original Westinghouse steam generators (referred to herein as the "baseline analysis"), in order to show that the pipe stresses, component loads and component support loads are still valid. If the results of the loop piping containing the replacement steam generators are less than that of the baseline analysis model, the stress reports need only be updated to reference the BWNS calculations. If the results of the loop piping containing the replacement steam generators are greater than those in the baseline analysis, the original Westinghouse design reports will be reviewed and updated as necessary.

A flow chart which shows the basic steps that will be followed in the reactor coolant system analysis is included as Attachment 2.

Pipe rupture loads due to a double-ended guillotine break in the primary loop are to be eliminated due to the use of leak-before-break criteria which was previously approved by the NRC for the reactor coolant systems at both McGuire and Catawba.[4] Pipe rupture loads due to breaks on the residual heat removal, accumulator, and pressurizer surge lines will still be considered on the model, as well as loads due to a break in the main feedwater, main steam, and any other applicable secondary side systems.

The pipe rupture analysis results for the component supports will be compared to the original component support loads. If the new loads are bounded by the original results, the original stress reports will be considered valid. It is considered likely that this will be the case due to the margins introduced by the use of leak-before-break criteria. If the component support loads calculated by BWNS exceed the original support loads, then the component stresses will be shown to satisfy the requirements of the applicable component Design Specification, the ASME Code edition to which the component was procured, and the plant specific FSAR and either new or revised stress reports will be issued. A final check will be made to verify that the analyses supporting the use of leak-before-break criteria are still valid for the revised loop analysis results.

---

[1]The Reactor Building basemats for both McGuire and Catawba are founded on rock, consequently, soil/structure interaction is not a consideration.

[2]Refer to NRC Question #7 at end of this document for additional information.

[3]Reference 10 and NRC Question #5 at the end of this document provide additional information.

[4]References 11 and 12 provide documentation of NRC staff acceptance of leak-before-break for McGuire and Catawba.

McGuire and Catawba are similar plants but they are not identical. Therefore the modeling and analysis approaches described above must be performed for each plant. The basic techniques to be used in the analysis are consistent with those already described in the FSAR for each plant. Some changes may be required due to the use of different computer programs, etc. The FSARs will be updated to reflect these and any other changes.

Reactor Coolant Loop Components

The replacement steam generators will be designed, fabricated, and tested by B&W to meet the requirements of Duke Power Company Design Specifications MCS-1201.01-00-0005 and CNS-1201.01-00-0004, Section III of the 1986 ASME Code, and Chapter 5 of the plant specific FSARs. All existing piping and components of the NSSS will be reviewed based on the Code edition to which they were procured.

The reactor vessel and reactor coolant pumps will be reviewed in a manner similar to the reactor coolant loop piping. The baseline (phase one piping math model described above) analysis nozzle loads on each piece of equipment will be compared with the nozzle loads from the original Westinghouse primary loop analysis [illegible]re that the baseline analysis is valid. The nozzle loads from the analysis containing the replacement steam generators are then compared to the nozzle loads from the baseline analysis. If the nozzle loads from the analysis containing the replacement steam generators are less than the baseline analysis nozzle loads, the existing stress reports will be updated to reference the BWNS calculations and no further work will be required. If the nozzle loads from the analysis containing the replacement steam generators are greater than the baseline analysis nozzle loads, the original Westinghouse stress reports will be reviewed and updated as necessary.

All existing reactor coolant loop components will be reviewed to determine the impact of the new thermal transients associated with the replacement steam generators. A fatigue evaluation will be included in this review if necessary, based on a comparison between the new and original design transients.

Reactor Coolant Loop Component Supports

The loads from the BWNS analysis of the reactor coolant loop on the component supports will be compared to the Duke Power design loads for each support. If the new loads are less than the design loads, the calculations will only be updated to reference the BWNS calculations. The supports will be analyzed to evaluate any loads which increase beyond the design loads. Any necessary support modifications will be made.

Reactor Building Structure

If the loads from the BWNS analysis of the reactor coolant loop on any of the component supports increase beyond the design loads for that support then the support will be reanalyzed as stated above. The embedment loads and building structure loads obtained from the support reanalysis will be compared to the design loads for the embedment and structure. If the new loads are less than the design loads, the calculations will only be updated to reference the BWNS calculations. Any of the new loads on the NSSS support embedments and building structures which increase beyond the design loads will require that the embedment and building structure be evaluated.

The increased weight of the replacement steam generators will be incorporated into the seismic analysis of the interior structure for each plant. New Reactor Building floor response spectra will be generated using methods described in Section 3.7 of the plant FSARs. These spectra, along with the associated frequencies and mode shapes, will then be compared to the design frequencies, mode shapes and spectra to ensure no significant changes have occurred in the building response.

Branch Piping Lines
The displacements, rotations and response spectra at branch line nozzle locations on the reactor coolant loop and components from the BWNS analyses will be compared to those movements and spectra used for the design of the branch piping systems. If the movements and spectra are less, the piping analysis calculations need only be updated to reference the BWNS calculations. If the movements or spectra increase, the branch lines will be evaluated and reanalyzed as necessary.

Any branch lines off of the reactor coolant loop or secondary side lines which are rerouted to accommodate the steam generator replacement will be reanalyzed.

Conclusion
A parametric analysis of the reactor coolant piping at McGuire Nuclear Station Units 1 & 2 and Catawba Nuclear Station Unit 1 provides adequate assurance that the design of the primary coolant loop, major components, component supports, branch piping systems and the building structures remain bounded by the original plant design bases after the steam generator replacement.

References

1) Final Safety Analysis Report, McGuire Nuclear Station

2) Final Safety Analysis Report, Catawba Nuclear Station

3) NUREG-0422, Safety Evaluation Report related to operation of McGuire Nuclear Station Units 1 and 2 including supplements

4) NUREG-0954, Safety Evaluation Report related to operation of Catawba Nuclear Station Units 1 and 2 including supplements

5) ASME Code Case N-411-1, Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1

6) Regulatory Guide 1.84, Design and Fabrication Code Case Acceptability, ASME Section III, Division 1, Revision 28

7) NUREG-0800, Standard Review Plan

8) Duke Power Company Specifications MCS-1201.01-00-0005 and CNS-1201.01-00-0004 (in preparation)

9) 1986 ASME Code, Section III, Division 1

10) May 17, 1990 correspondence from K.N. Jabbour (NRC) to H.B. Tucker (DPC), Docket Nos. 50-413 and 50-414, concerning direct generation of response spectra at Catawba Nuclear Station (TACs 67359/67360)

11) May 8, 1986 correspondence from B.J. Youngblood (NRC) to H.B. Tucker (DPC), Docket Nos. 50-369 and 50-370, concerning elimination of large primary loop pipe ruptures at McGuire Nuclear Station

12) April 7, 1987 correspondence from K.N. Jabbour (NRC) to H.B. Tucker (DPC), Docket Nos. 50-413 and 50-414, concerning elimination of large primary loop pipe ruptures at Catawba Nuclear Station

13) Final Rule, 10 CFR Part 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Federal Register, Vol. 51, No. 70, p.12502, April 11, 1986

14) Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants

15) Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants

16) BWSPAN User's Manual, Revision G, NPD-TM-35

Questions from 10/28/92 Duke Power-NRC Tele-Conference

A telephone conference call was held on October 28, 1992 between representatives of Duke Power Company and the NRC. Several questions were brought up by the NRC during the course of this conversation. Answers to those questions are provided below.

1) Question: Are pipe rupture loads included in the "baseline" analysis?

Response: No. The purpose of the "baseline" analysis is to demonstrate that the analysis assumptions and structural properties of the primary loop model (mass, stiffness, boundary conditions) will produce seismic results comparable to those obtained by Westinghouse. Therefore only gravity and seismic loads will be applied to the "baseline" model.

2) Question: How will we obtain Westinghouse loads used in the original analysis if they are required?

Response: Duke Power has negotiated an agreement with Westinghouse which will give us access to information that we might deem necessary.

3) Question: What is the basis of comparison between the "baseline" analysis and the Westinghouse analysis?

Response: The "baseline" analysis will be considered acceptable if the results are within ±10% of the Westinghouse results.

4) Question: How will discrepancies in the "baseline" analysis (the ±10% variation mentioned in question #3) be accounted for in the comparison between the "baseline" analysis and the final analysis?

Response: Duke Power and BWNS will reconcile any discrepancies in the analysis results as needed. If the final analysis results are greater than the "baseline" analysis results then the final results will be compared directly to the original Westinghouse results or to the ASME Code design limits to certify acceptability.

5) Question: What is the justification for using ASME Code Case N-411-1 damping? How was its use approved previously? Are the requirements of Regulatory Guide 1.84 satisfied?

Response: ASME Code Case N-411-1 damping will be applied subject to the restrictions of Regulatory Guide 1.84, Revision 28. The use of this Code Case was previously approved for use with direct generation methods for response spectra in Duke Power Company's snubber reduction program. No conditions on the use of the Code Case other than compliance with Regulatory Guide 1.84 were cited. The approval of the Code Case for use with the direct generation of response spectra

is documented in Reference 10. The restrictions of the Regulatory Guide are to be met as follows.

(1) The Code Case will be used consistently for all piping in the primary loop model. Regulatory Guide 1.61 damping values will be used for all other components and structures. Duke Power is in compliance with RG 1.61 per Section 3.7.1 of the Catawba SER (Reference 4). The McGuire SER does not specifically address this topic (Reference 3).

(2) The damping will be used only for response spectrum analyses. The seismic design response spectra used for the design of all seismic Category I structures, systems, and components was reviewed in Section 3.7.1 of the Catawba SER (Reference 4) and found to comply with the recommendations of Regulatory Guide 1.60.[5] The McGuire SER does not specifically address this topic (Reference 3).

(3) Supports with existing gaps will be reviewed for the effects of any increased piping movements.

(4) The piping systems to be analyzed using the Code Case do not contain pipe supports designed to dissipate energy by yielding.

(5) None of the piping systems to be analyzed are subject to stress corrosion cracking.

6) Question: How will different damping values be applied to the different components in the primary loop model?

Response: BWNS will analyze the NSSS using response spectrum methods and composite modal damping. Damping values used in the model will be from Code Case N-411-1 for the piping elements and from Regulatory Guide 1.61 for the remainder of the model elements. (Section 3.7.1 of the plant specific FSARs permits the use of N-411-1 damping. See the previous question for a discussion of Code Case applicability.) Restrictions on the use of the Code Case, given in Revision 28 of Regulatory Guide 1.84, require the consistent use of N-411-1 damping for the piping and the concurrent use of RG 1.61 damping for other model elements.

The composite modal damping will be applied using the "bar strain energy damping method" in the BWNS proprietary computer code BWSPAN (Reference 16). Damping for model elements other than piping will be assumed constant for all frequencies while damping for piping will be frequency dependent per Code Case N-411-1. The bar strain energy method of damping is recommended for use in Revision 2 of the Standard Review Plan for Section 3.7.3, "Seismic Subsystem

[5]Catawba and McGuire were licensed for the same design response spectra.

Analysis" through its reference to SRP Section 3.7.2 (it is referred to as composite modal damping in that section.) This same method is recommended in Section N-1233.2 of Appendix N of Section III of the ASME Boiler and Pressure Vessel Code (Reference 9 – referred to as subregional modal damping there.) In both the SRP and Appendix N there are two techniques for applying this method: mass weighted modal damping and stiffness weighted modal damping. BWSPAN uses stiffness weighted modal damping.

7) Question: It was stated that the use of the ground (or basemat) spectra instead of floor spectra will reduce seismic input through the elimination of the effects of spectra broadening in the floor spectra. How or why is this beneficial?

Response: BWNS routinely uses a coupled model in its NSSS analyses. This approach is supported by Revision 2 of Standard Review Plan Section 3.7.2, "Seismic System Analysis", which states in Section II.3.b that the reactor coolant system "is considered a subsystem but is usually analyzed using a coupled model of the reactor coolant system and primary structure." Coupling of the interior structure and NSSS models reduces seismic input in the following ways.

(1) The four synthetic time histories used to generate the floor spectra were developed such that a ground response spectra generated from these time histories envelopes the Regulatory Guide 1.60 spectra at most frequencies (see Section 3.7.1 of supplement 1 to the Catawba SER – Reference 4.) The smooth RG 1.60 ground response spectra was generated to be inherently conservative. The use of peak broadened floor response spectra, which would be required input for a decoupled NSSS model, would add unnecessary seismic energy to the analysis.

(2) The decoupled model cannot account for the offsetting effects of the loop on the interior structure and vice versa: the interior structure model only accounts for the NSSS mass, not the NSSS stiffness. Stated another way, if the NSSS were included in the decoupled interior structure model its stiffness would tend to reduce the spectra at the floor elevations, thus reducing the input spectra to the loop model. This "offsetting effect" is a natural part of the coupled model.

8) Question: Will the generation of new floor spectra be a separate time history analysis?

Response: The additional mass of the new steam generators will be incorporated into the interior structure seismic model. New floor response spectra will be generated for comparison with the design floor response spectra in order to ensure that no significant changes have occurred in the building response. The new spectra will be generated by either time

history or direct generation methods as permitted in Section 3.7 of the McGuire and Catawba FSARs.

9) Question: Provide supporting documentation on the approval of leak-before-break by the NRC staff for McGuire and Catawba.

Response: Leak-before-break was approved for McGuire Nuclear Station in the May 8,1986 correspondence from B. J. Youngblood (NRC) to H.B. Tucker (DPC), Docket Nos. 50-369 and 50-370. Approval for Catawba Nuclear Station was granted in the April 7,1987 correspondence from K.N. Jabbour (NRC) to H.B. Tucker (DPC), Docket Nos. 50-413 and 50-414. Both pieces of correspondence recognize Duke Power compliance with the April 11, 1986 final rule (51 FR 12502) amending 10 CFR Part 50, Appendix A, GDC 4 (Reference 13).

10) Question: How will traceability be maintained for the plant design basis?

Response: Duke Power will ensure that all calculations, specifications, stress reports and any other items essential to the documentation of the plant design basis are revised or updated to reflect any impact due to steam generator replacement.

11) Question: Provide more specifics on the design specification, ASME Code edition and FSAR sections applicable to the design of the replacement steam generators.

Response: The replacement steam generators will be designed, fabricated, and tested by B&W to meet the requirements of Duke Power Company Design Specifications MCS-1201.01-00-0004 and CNS-1201.01-00-0005, Section III of the 1986 ASME Code, and Chapter 5 of the plant specific FSARs. All existing piping and components of the NSSS will be reviewed based on the Code edition in force at the time of their procurement.

12) Question: Provide more details on the loop models (i.e. boundary conditions, damping, etc.).

Response: BWNS is currently in the process of constructing the loop models and writing a topical report describing the analytical procedures that they intend to employ in their analysis. Further information will be available when these activities are complete.

Attachment 2

NSSS Primary Loop Analysis Methodology