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October 15, 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

Gentlemen:

Pursuant to 10CFR 50.4, attached is 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.

The reanalysis effort consists of a parametric study which will show that the original design basis analysis of the reactor coolant system is conservative and will therefore remain valid after the replacement steam generators are installed. Since the proposed methodology has not been previously used by Duke Power Company, it is requested that the NRC review and indicate acceptance of the proposal included in Attachment 1, in support of the current steam generator replacement schedule by December 1, 1992.

Should there be any questions concerning this proposal or if additional information is required, please contact David V. Ethington at (704) 382-6633.

Very truly yours,

H. B. Tucker, Senior Vice President
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Attachments

xc (w/att):

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Regional Administrator, Region II

T. A. Reed, ONRR

R. E. Martin, ONRR

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Senior Resident Inspector (MNS)

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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. 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 almost (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 top of basemat response 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 the effects of spectra broadening in floor response spectra. 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) as acceptable alternative damping values. 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 then 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 then the original Westinghouse design reports will be checked and upgraded 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. 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 obviously bounded by the original results then 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, the component stresses will be shown to satisfy the requirements of the Design Specification, ASME Code, and 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.

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 analyzed and designed by B&W to meet the requirements of the Design Specification, the ASME Code, and the FSAR.

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 to ensure 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 then 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 then the original Westinghouse stress reports will be checked 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. No work will be necessary otherwise.

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 or spectra increase then the branch lines will be evaluated and reanalyzed as necessary; otherwise, no work will be required. 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.

Attachment 2

NSSS Primary Loop Analysis Methodology