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August 28, 2000

U. S. Nuclear Regulatory Commission
Document Control Desk
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Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Reactor Coolant Loop Re-Analysis

The purpose of this letter is to request NRC review and approval of the methodology that will be used for the reactor coolant loop re-analysis for Oconee as part of steam generator replacement.

On October 28, 1999, representatives of Duke Energy Corporation (Duke) and their contractor, SGT, met with representatives of the NRC staff to discuss Duke's plans for replacement of the Once Through Steam Generators (OTSGs) at Oconee Units 1, 2, and 3 beginning in the Fall of 2003. As discussed in that presentation, the replacement OTSGs (ROTSGs) are essentially identical to the existing OTSGs except that the shells are manufactured with a higher strength forged material that results in reduced weight and the original skirt support will be replaced with a pedestal support that will facilitate installation and future Inservice Inspection. Replacement of the OTSGs will necessitate re-analysis of the reactor coolant loop structural design basis. This re-analysis is considered necessary to demonstrate that the reactor coolant loop stresses are not exceeded by the introduction of the ROTSGs.

Since the original 1960's vintage structural analysis computer codes are no longer available, Framatome Technologies (FTI) has been contracted to perform a complete re-analysis of the reactor coolant loop using current analytical methods. This approach is consistent with licensing support provided for other utilities in support of steam generator replacement. This analysis will be similar to the re-analysis that was done for Catawba Unit 1

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and McGuire Units 1 and 2 in conjunction with their steam generator replacement projects. The McGuire/Catawba loop re-analysis methodology was reviewed and approved by the NRC in an SER dated April 8, 1993. For McGuire and Catawba, the loop re-analysis was used to demonstrate that the analysis in their respective UFSARs remained conservative with the installation of the replacement steam generators. Unlike McGuire and Catawba, the updated loop analysis for Oconee will be included in the UFSAR as the analysis of record pursuant to 10CFR50.71(e).

The methodology to be used by FTI for re-analysis of the Oconee reactor coolant loops is described in Attachment 1. It is requested that the NRC review and approve the proposed methodology by December 31, 2000.

If there are any questions, please contact Robert Sharpe at (704) 382-0956.

Very truly yours,

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Attachment

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Attachment 1
Duke Energy Corporation
Oconee Nuclear Station
Methodology for Analysis of the Reactor Coolant Loop
For Steam Generator Replacement

1.0 Introduction

The Once Through Steam Generators currently in service in Oconee Nuclear Station (ONS) Units 1, 2, and 3 will be replaced with new steam generators of a near identical design. The replacement once through steam generators (ROTSGs) will incorporate a number of material changes that will reduce the operating weight of each ROTSG. The ROTSGs will be supported on a pedestal rather than on the current skirt arrangement. In 1985, the NRC approved the elimination of the dynamic effects of large break loss of coolant accidents (LOCAs). These changes necessitate the re-analysis of the Oconee reactor coolant loop using current methodologies. Design details for the ROTSGs will be provided by Babcock & Wilcox Canada (BWC).

2.0 Approach

The purpose of the structural analysis is to demonstrate that the design basis requirements for the piping, components, and supports are still met with the ROTSGs in the system. This will be demonstrated in one of two ways:

1. By showing that the loads acting on the piping, components, and supports do not increase above design basis loads when the ROTSG is introduced into the Reactor Coolant System (RCS).

2. By showing that the RCS stresses, which are present after the ROTSG is introduced into the RCS, continue to meet the allowable stresses dictated by the applicable design codes.

3.0 RCS Structural Model Development

Full structural models of the ONS Unit 1 and ONS Units 2 and 3 Reactor Coolant Systems (RCSs) will be developed using the

Framatome Technologies (FTI) structural code BWSPAN. A separate model of Unit 1 is necessary due to the fact that Unit 1 has Westinghouse reactor coolant pumps (RCPs) and Units 2 and 3 have Bingham RCPs. These models will include the RCS components, RCS piping, component supports, Control Rod Drive Mechanisms (CRDMs), Service Support Structure (SSS), the ROTSG internals, and the Interior Concrete Structure (ICS).

- The RCS components that will be modeled include the Reactor Vessel (RV), RCP assembly (pump, motor stand and motor), ROTSG and pressurizer. Centerline models will be used to represent the components. Local flexibility of each component at the RCS piping connections will be included in the model and will be calculated using methods developed by P.P. Bijlaard (Reference 3).

- The RCS piping that will be modeled includes the hot legs, lower cold legs, upper cold legs and surge line. Centerline models will be used to represent the piping. Nominal dimensions will be used in the model, however, as-built dimensions may be used in the stress analysis of the piping. Piping attached to the RCS, excluding the surge line, will be de-coupled from the RCS model on the basis of the guidance given in Welding Research Council Bulletin 300, "Technical Position on Industry Practice" (Reference 4).

- The RCS component supports that will be modeled include the RV support skirt, ROTSG pedestal, steam generator upper supports (SGUS), RCP supports and restraints, and the pressurizer support frame. The support skirt/pedestal will be represented as centerline models. Each of the SGUSs will be represented as a set of five springs, one for each "spur" of the SGUS. The RCP snubbers and spring hangers will be represented as springs. The structural steel members which support the RCP hangers and snubbers will be modeled as beams with the appropriate cross section properties. The structural steel members which make up the pressurizer support frame will be modeled in a similar fashion.

- The CRDMs will be represented in the structural model as a vertical beam having cross section properties calculated by considering the properties of all of the CRDMs together. The

SSS is also modeled as a single vertical beam which runs parallel to the CRDM beam. The properties of this beam are representative of the SSS as a whole. The CRDM and SSS beams are connected by a spring which represents the CRDM clamps.

- The ROTSG internals (tubes, tube support plates (TSP), tubesheets and wrapper) will also be represented in the structural model as a series of beams with springs to represent the interfaces between the TSPs and the wrapper and shell.

- The ICS will be represented in the structural model as a series of beams having the cross sectional properties dictated in the original analysis of the ONS containment buildings (Reference 5). An "isolated" model of the ICS, similar to the model described in Reference 5, will be developed in BWSPAN. Frequencies and mode shapes will be calculated for this isolated model and these will be compared to the results obtained in the original analysis as a means of benchmarking the model. After the isolated model has been successfully benchmarked, it will be inserted into the model to be used in the loading analysis of the RCS.

Nodes will be placed at all current (pre-ROTSG) whip restraint locations.

Current modeling techniques will be used in the development of the RCS structural model. These include the use of uniform mass distribution, explicit modeling of structural steel beams and frames, coupling of the RCS and ICS, modeling of the entire RCS, element specific damping (different damping assigned to different elements in the same model), and others.

4.0 Development and Analysis of the RCS, ROTSG and Reactor Building Hydraulics Models

Current modeling techniques will be used in the development of the RCS, ROTSG and reactor building hydraulics models. These include the use of current discharge correlations (Modified Zaloudek-Moody, for example) and the subcompartment modeling

techniques discussed in NUREG 0609 (Reference 9) and Standard Review Plan Section 6.2.1.2 of NUREG 0800 (Reference 10).

The RCS, ROTSG secondary side and reactor building initial conditions will be those at 100% power for the High Energy Line Break Accident (HELBA) analyses. Temperatures and pressures will be taken from the ROTSG Certified Design Specification (CDS) (Reference 1) or other documentation provided by BWC, whichever is available at time of analysis.

4.1 RCS Hydraulics Model and Analysis

A thermal hydraulic model consisting of a network of fluid control volumes and flow paths will be developed for the complete RCS using the FTI hydraulic code, CRAFT2. The model, after being initialized to normal steady state pressures and temperatures, will be used to determine hydraulic forcing functions (force time histories) at changes in area and flow direction, which would occur during rapid depressurization after a postulated LOCA. The force time histories will be oriented to the global coordinate system for input to the structural model using the FTI computer code, BWHIST.

Because the NRC has accepted the application of Leak-Before-Break (LBB) methodology at the ONS units (Reference 14), breaks in the large bore primary piping will not be considered. The hydraulics analyses of the RCS will consider the controlling breaks which remain after application of LBB:

- Decay heat line break at the hot leg nozzle.

- Two terminal end surge line breaks - one at the hot leg nozzle and one at the pressurizer nozzle.

- Stress-induced intermediate surge line break - FTI will establish and analyze the controlling stress-induced intermediate break.

- Core flood line break at the RV nozzle.

Note that the surge line break at the pressurizer nozzle will be considered because it is the controlling break for the

pressurizer support frame and that the core flood line break at the RV nozzle will be considered because it is the controlling break in the RV cavity.

Mass and energy release data generated in the analyses for these breaks will be saved for later use in the Asymmetric Cavity Pressure (ACP) loading analysis of the RCS. Other data from these analyses, such as temperatures, pressures and flow rates, will be used to evaluate Jet Impingement (JI) loading of the RCS.

4.2 ROTSG Hydraulics Model and Analysis

A thermal hydraulic model consisting of a network of fluid control volumes and flow paths will be developed for the ROTSG using the FTI hydraulic code, CRAFT2. The model, after being initialized to normal steady state pressures and temperatures, will be used to determine hydraulic forcing functions (pressure time histories), at changes in area and flow direction in the generator, which occur during rapid depressurization after a postulated secondary side HELBA. The pressure time histories will be converted to force time histories by multiplying them by the appropriate area, and will be oriented to the global coordinate system for input to the structural model using the FTI computer code, BWHIST.

The ROTSG hydraulics model will account for the flow restrictor in the main steam outlet nozzle. The flow restrictor is designed to prevent rapid steam generator depressurization by choking the steam flow should a steam line break occur and will minimize the pressure drop loading on the generator and its internals.

Hydraulics analyses of the ROTSG will consider "Single" and "Double" Main Steam Line Breaks (SMSLB and DMSLB) and Main Feedwater Line Breaks (MFWLB) HELBAs. The SMSLB is postulated at the generator nozzle for any one steam line. The DMSLB is postulated at the juncture of the two main steam lines from a given generator such that both main steam lines blow down. The MFWLB is postulated at the juncture of the 14 inch riser pipe and the 14 inch tee in the feedwater header. These breaks are the controlling secondary side breaks.

Mass and energy release data generated in the analyses for these breaks will be saved for later use in the ACP loading analysis of the RCS. Other data from these analyses, such as temperatures, pressures, flow rates, and steam quality, will be used to evaluate Jet Impingement (JI) loading of the RCS. Note that there are no credible RCS targets for the DMSLB, so a JI analysis will not be performed for this break.

4.3 Reactor Building Hydraulics Model and Analysis

A thermal hydraulic model consisting of a network of fluid control volumes and flow paths will be developed for the reactor vessel and steam generator cavities within the reactor building. Component, equipment and structural steel volumes in the cavities will be accounted for, that is, they will be subtracted from the control volumes representing the cavities. This model, with mass and energy release data from the RCS and ROTSG hydraulic analyses, will be used to determine pressure time histories in the control volumes using the FTI hydraulic code, COMPAR2. These pressure time histories constitute ACP loading on the RCS components. The pressure time histories will be converted to force time histories by integration over appropriate surface areas of the components, and will be oriented to the global coordinate system for input to the structural model using the FTI computer code, BWHIST.

ACP loading will be generated for all of the primary side breaks listed in Section 4.1. ACP loading will also be generated for the SMSLB and MFWLB secondary side breaks as listed in Section 4.2. The DMSLB is outside of the steam generator cavity and therefore it generates no appreciable ACP loading on the RCS components.

Note that this ACP analysis is not intended to determine maximum building pressures. Therefore, only the controlling breaks which remain after LBB are considered and the time span of the analysis is limited to that needed to describe the ACP loads.

4.4 Jet Impingement and Thrust Analysis

The geometry of the primary and secondary side breaks listed in Sections 4.1 and 4.2 will be evaluated in order to identify credible RCS component targets for jet impingement.

Temperatures, pressures, flow rates, steam quality and other data from the RCS and ROTSG hydraulic analyses will be used to determine the JI forces acting on the RCS components which are credible targets. ANSI Standard 58.2 (Reference 12) methodology will be used to calculate the JI forces. JI loading will be calculated for all of the breaks listed in Sections 4.1 and 4.2 except the DMSLB which occurs outside of the steam generator cavity and therefore has no potential to impinge on any RCS components.

Reaction forces (thrust) acting at the break will also be calculated. Thrust load time histories will be provided for all of the primary and secondary side breaks listed in Sections 4.1 and 4.2 except for the DMSLB: the DMSLB is remote from the RCS such that the thrust loads at the break would not be felt by the RCS.

5.0 Structural Loading Analysis

FTI will consider the following load cases in the loading analysis of the RCS:

- Pressure: Design and operating (as appropriate),
- Deadweight: 100% power operating weight,
- Thermal Expansion: 0, 8 and 15 and 100% power, reactor trip,
- Seismic: Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE),
- High Energy Line Break Accident (HELBA).

All loading analyses will be performed using the BWSPAN model of the RCS and will use properties at 100% operating conditions unless otherwise noted.

5.1 Pressure Loading

Design (2500 psig) or operating (2155 psig) pressure will be considered in the ASME Section III stress calculations, as appropriate.

5.2 Deadweight Loading

The mass of the modeled components and their internals, entrained fluid and insulation will be considered as distributed mass in the deadweight analysis. Mass of the component supports, such as the support skirts and structural steel beams, will be considered as distributed or lumped mass, as appropriate. The mass of the ICS will be considered as lumped mass in the model in keeping with the original ICS analysis (Reference 5). Other mass supported by the RCS, such as whip restraints and feedwater headers, will be considered as lumped mass if it is deemed significant.

5.3 Thermal Expansion

Four thermal expansion load cases will be considered for normal operating conditions at 0%, 8%, 15% and 100% power. The 0% and 8% power conditions will be considered to provide a lower bound for thermal stress ranges in the fatigue calculations. The 15% thermal expansion case will be considered because the cold leg temperature is highest at this power level. The RCS temperatures used for each expansion analysis will be taken from the CDS for the ROTSG or from documentation provided by BWC, whichever is available at the time of analysis.

The upset condition reactor trip transients in the current ONS Functional Specification (Reference 6) will be reviewed to identify the bounding overtemperature transient for the hot and cold leg. This bounding transient will be analyzed.

5.4 Seismic Loading

Both Operating Basis and Safe Shutdown Earthquake scenarios will be analyzed. Note that the ONS Design Base Earthquake will be referred to as "OBE" and the ONS Maximum Hypothetical Earthquake will be referred to as "SSE."

Seismic loading analysis will be performed using the response spectrum method. Because the ICS will be included in the model, seismic excitation (basemat response spectra) will be applied at the base of the ICS model along the RCS's three global axes.

Per Section 3.7 of the ONS UFSAR (Reference 2), modal combination will be by Square-Root-Sum-of-the-Squares (SRSS). The cutoff frequency for the seismic analyses will be 33 Hz. The contribution of those modes beyond the cutoff frequency will be accounted for using the technique outlined in Standard Review Plan 3.7.2 of NUREG 0800 (Reference 8).

Damping will be taken from Section 3.7 of the ONS UFSAR: 0.5% for piping (OBE and SSE), 1% for components (OBE and SSE), 2% for steel and reinforced concrete supports (OBE and SSE), and 5% for the ICS (OBE and SSE).

Earthquake direction combination will be 2-Dimensional and the results from both the X+Y and Y+Z combinations will be reported, where X and Z are the two horizontal earthquakes. The X (or Z) earthquake results will be combined with the Y earthquake results by absolute summation.

5.5 HELBA Loading

For a given pipe break, the internal forcing function, ACP, JI and thrust time histories which result from the RCS, ROTSG and reactor building hydraulics analyses will be applied to the structural model of the RCS. The breaks considered are listed in Section 4.0. The LOCA restraints on the RCP assembly, the hot leg whip restraint, and the cold leg whip restraint will not be considered active in the HELBA loading analyses.

There is no commitment to specific damping values for HELBA in the ONS UFSAR and there are no known issues relative to HELBA which are compensated for through the use of conservative damping. Therefore, the SSE damping values given in Regulatory Guide 1.61 (Reference 7) will be used: 3% for equipment and piping >12 inch OD, 2% for piping <= 12 inch OD, 4% for welded steel structures, 5% for pre-stressed concrete structures and 7% for bolted steel and reinforced concrete structures.

5.6 Load Combinations

Load combinations for use in the ROTSG nozzle and support skirt load comparison (Section 8.0) will be per the ROTSG CDS. Load combinations for use in the load comparisons for the remaining RCS component nozzles (Section 10.0) and the RV support skirt (Section 11.0) will be those defined in Section 3.9.3.1.1 of the ONS UFSAR.

Load combinations for use in the load comparisons/stress analyses for the SGUS and RCP supports (see Section 11.0) will be per the original design calculations. Load combinations for use in the RCS pipe stress calculations (see Section 9.0) will cover both the load combinations described in Section 3.9.3.1.1 of the ONS UFSAR and those load combinations described in the prescribed design code, the 1983 Edition of Section III to the ASME Boiler and Pressure Vessel Code.

In cases where combination of SSE and HELBA loads is called for, the combination will be by SRSS.

6.0 Amplified Response Spectra Generation

Amplified Response Spectra (ARS) will be generated at all branch line nozzle locations on the RCS primary piping as well as at the auxiliary feedwater, main feedwater, main steam lines, and lower tubesheet drain nozzle locations on the ROTSG. ARS will also be generated at other locations on the ROTSG. Response spectra will be provided for both OBE and SSE at 0.5, 1, 2 and 5% damping.

The spectra will be generated using FTI's response spectrum generator code, RESPECT. RESPECT reads frequency and mode information from the output files generated by BWSPAN during the seismic loading analysis. In addition, basemat time histories for the three earthquake directions are input by the user. RESPECT will generate response spectra in the three coordinate directions at any "non-fixed" node in the BWSPAN model for each direction of earthquake. Each point selected will have three spectra for each of the three coordinate directions (one from the X earthquake, one from the Y earthquake and one from the Z earthquake). The minimum frequency intervals suggested in Reg.

Guide 1.122 (Reference 13) will be met or smaller intervals will be used. These three spectra will be combined by SRSS of accelerations at each frequency. ARS will be peak broadened by +/- 10% in accordance with Section 3.7.2.4.1 of the ONS UFSAR.

7.0 RCS Loading Specification

FTI will tabulate displacements and loads at key locations throughout the RCS resulting from the analysis of the load cases described in Section 5.0. Displacements will be provided at branch nozzle and whip restraint locations. Loads will be reported at the primary nozzles (including both of the surge line nozzles) and at the RCS supports. Tabulation will be performed using FTI's specification writer, BWSPEC. Note that the loads being presented here are actual analysis results and not allowables. The stress report summary updates discussed later will place the highest analyzed load, that is, the "allowable load," into the summary document.

The ARS generated for the Operating Basis and Safe Shutdown Earthquakes will also be placed in this document.

8.0 Confirm the ROTSG Design Loads and Force Time Histories

FTI will make the following comparisons to ensure that the design loads given in the ROTSG CDS (Reference 1) envelop the actual loads, which result from the loading analysis:

Primary nozzle loads from the FTI loading analysis and secondary nozzle loads supplied by Duke will be compared to the design loads given in the ROTSG CDS.

- Loads acting on the ROTSG from the upper and lower supports will be compared to the design loads given in the CDS.

- Seismic loads on the ROTSG internals at the locations where the internals attach to the wrapper and/or shell will be compared to the design loads in the CDS.

- Force time histories on the ROTSG internals, which result from the FTI MSLB loading analysis, will be compared to the design time histories in the CDS.

If the loads or force time histories resulting from the FTI loading analyses exceed those in the ROTSG CDS, they will be forwarded to the ROTSG designer (BWC) for consideration.

9.0 Primary Piping Stress and Fatigue Analysis

The design code for the RCS piping will be changed from the 1968 Edition of USA Standard B31.7 to Subsection NB of Section III of the 1983 Edition of the ASME Boiler and Pressure Vessel Code (no addenda). Stress calculations will be performed using the simplified pipe stress equations as given in Article NB-3600 of the 1983 Code. A code reconciliation will be performed to assess the impact of this change. The basic material allowable stresses used will be the lower of those in the 1968 B31.7 or those in the 1983 ASME Boiler and Pressure Vessel Code. Stress allowable factors (1.8 for Level B primary stress check, for example) will be taken from the 1983 ASME Boiler and Pressure Vessel Code.

Once the code reconciliation is complete, stress analysis of the primary loop piping (including the surge line) will be performed. This will be accomplished using FTI codes BWSPAN and T3PIPE. BWSPAN will be used to calculate stresses due to pressure and mechanical loads at all piping locations. T3PIPE will be used to calculate run-branch stresses at branch connections and to calculate fatigue stress and usage factors at all locations. Required T3PIPE inputs include: pipe geometry, run pipe loads (taken from the loading specification described in Section 7.0), branch pipe loads (limit loads calculated by FTI) and thermal radial gradients/thermal discontinuity stresses (taken from existing analyses, hand calculated or generated using verified computer codes). Appropriate stress intensification factors will be used for all geometries.

The combined loads used in the Leak-Before-Break (LBB) stress calculations will be calculated using the individual loads from the loading specification described in Section 7.0. These will

be compared to the combined loads used in the existing (pre-ROTSG) LBB calculations to confirm that LBB requirements continue to be met with the ROTSG in the RCS.

10.0 Primary Nozzle and Lug Load Comparisons/Stress Analysis

Loads generated by the FTI loading analysis for the RV and RCP primary nozzles, the CRDM nozzles, and the surge line nozzles will be compared to the existing (pre-ROTSG) design loads. FTI analysis loads will be taken from the loading specification described earlier and existing design loads will be taken from the original stress reports for the component in question. Where the FTI analysis loads are higher, stress and fatigue analysis will be performed in accordance with the original stress reports and the original design codes. The original design codes are: 1965 Edition of Section III of the ASME Boiler and Pressure-Vessel (B+PV) code with addenda through Summer 1967 for the RV/pressurizer/Unit 1 RCP and 1968 Edition of Section III of the ASME Boiler and Pressure Vessel (B+PV) code with addenda through Summer 1970 for the Units 2 and 3 RCPs.

Comparison of the FTI analysis loads on the ROTSG primary nozzles to the design loads contained in the ROTSG CDS is discussed in Section 8.0. FTI will perform the as-built analysis of the ROTSG primary nozzles after final machining for fit-up has been completed. This analysis will calculate stress and fatigue according to the ROTSG design code using mechanical loads from the FTI loading analysis. Peak stresses calculated during the design of the ROTSG should not change due to the as-built condition of the nozzles, that is, peak stresses will not be regenerated for the as-built analysis.

The FTI analysis loads on the pressurizer support lugs will be compared to the design loads given in the FTI stress report for the pressurizer. If the FTI analysis loads are higher, stress analysis will be performed in accordance with the original stress report and the design code.

11.0 Equipment Support Loads Comparison/Stress Analysis

Loads, taken from the FTI loading analysis, which act on the supports listed below, will be compared to existing (pre-ROTSG) design loads:

- RV skirt and its embedded steel
- ROTSG lower support embedded steel
- ROTSG upper support and its embedded steel
- RCP hangers/snubbers and their embedded steel

The existing design loads will be taken from the original stress report in the case of the RV support skirt and from the original design calculations in the case of the remaining supports and embedded steel.

Stresses will be calculated by FTI's structural analysis code, BWSPAN, for the pressurizer support frame and the RCP support beams according to the rules given in Subsection NF to Section III of the ASME Boiler and Pressure Vessel Code. A code reconciliation will be performed to assess the impact of the change from the original design code, 6th Edition of the AISC Manual of Steel Construction, to the 1983 Edition of Subsection NF. Load combinations will be performed as described in Section 5.0.

12.0 Update Stress Report Summaries

FTI will update the Stress Report Summaries it has developed for ONS under the B&W Owners Group program. Stress Report Summaries have been developed for the RCS piping, RV, CRDM, RCP, OTSG and pressurizer.

Loads on nozzles and support points generated in the FTI loading analysis described in Section 5.0 will be added to the summary documents if they exceed the loads currently listed. Stresses calculated as part of the RCS qualification described in Sections 9.0 through 11.0 will be added to the summary documents

in all cases. Note that actual loads resulting from the FTI loading analysis of the RCS will be contained in the loading specification described in Section 7.0.

13.0 Computer Codes

BWHIST:

An FTI developed code which converts pressure time histories generated by CRAFT2 or COMPAR2 into force time histories by integrating the pressure over the area to which it is being applied.

BWSCAN:

An FTI developed code which performs B31.7 and Section III (ASME Boiler and Pressure Vessel Code) stress and fatigue calculations for Class 1 and 2 piping. BWSCAN can also calculate stresses for linear type supports according to Subsection NF of Section III. Load inputs are taken from BWSPAN output files.

BWSPAN:

An FTI developed code which performs structural analysis of piping and structural systems. Deadweight, thermal expansion, response spectrum, time history and thermal stratification loading can be analyzed. Output includes displacements, loads, accelerations and displacement time histories, as appropriate.

BWSPEC:

An FTI developed code which tabulates displacements, pipe and structure loads, support loads and spring loads for selected locations using output from a BWSPAN analysis. Tabulations can be made for static, response spectrum and time history load cases.

COMPAR2:

An FTI developed code which performs hydraulics analysis of fluid systems (generally containment cavities). The system is modeled as a series of control volumes and flow paths such that the behavior of a pressure wave caused by a pipe break can be predicted. Pressure time histories can be obtained for any structure included in the model.

CRAFT2:

An FTI developed code which performs hydraulics analysis of fluid systems (generally piping or components). The system is modeled as a series of control volumes and flow paths such that the behavior of a pressure wave caused by a pipe break can be predicted. Pressure time histories can be obtained at changes in area or changes in flow direction.

P91232:

An FTI developed code which calculates through-wall gradient temperatures and stresses given pipe or nozzle geometry and thermal characteristics (time dependant fluid temperature and film coefficients or flow rates).

RESPECT:

An FTI code which generates Amplified Response Spectra (ARS) given the frequency and mode characteristics of the system in question (from BWSPAN) and the acceleration time history applicable to the base of the structure. RESPECT is generally used to generate seismic ARS at the branch nozzle locations in a model of a piping system.

T3PIPE:

An FTI code which performs Class 1 pipe stress and fatigue calculations per Section III of the ASME Boiler and Pressure Vessel Code. Capabilities include run/branch calculations. Inputs include a description of the pipe geometry, applied loading input and peak stress input (thermal radial gradient and thermal discontinuity stresses).

12.0 References

1. Duke Power Company Specification OSS-0279.00-0001, Revision 0, "Bid Specification for Replacement Steam Generators."

2. Updated Final Safety Analysis Report for the Oconee Nuclear Station through the December 31, 1998, Update.

3. "Stresses from Radial Loads and External Moments in Cylindrical Pressure Vessels," P.P. Bijlaard, Welding Journal, Volume 34, 1955.

4. "Technical Position on Industry Practice," Welding Research Council Bulletin 300, December, 1984.

5. Duke Document OSC-7298, Revision D1, "Seismic Analysis of the Internal Structure."

6. FTI Document 18-1130828-04, "Functional Specification, Reactor Coolant System."

7. USNRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 0, October 1973.

8. USNRC Standard Review Plan Section 3.7.2 of NUREG 0800, "Seismic System Analysis," Revision 2, August 1989.

9. USNRC NUREG 0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981.

10. USNRC Standard Review Plan Section 6.2.1.2 of NUREG 0800, "Subcompartment Analysis," Revision 2, July 1981.

11. Code of the Federal Register, Part 50, Section 50.59, "Changes, Tests and Experiments."

12. ANSI Standard 58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," 1988.

13. USNRC Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," Revision 1, February 1978.

14. NRC letter dated December 12, 1985 to the B&W Owners Group approving Topical Report BAW-1847, Rev 1, "Leak-Before-Break Evaluations of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS."