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Subject: Oconee Nuclear Station,
Units 1, 2, and 3
Docket Nos. 50-269, 270, and 287
Responses to Request for Additional Information
Reactor Coolant Loop Re-Analysis

By letter dated August 28, 2000, Duke requested that the NRC review and approve the methodology that is being used for the reactor coolant loop re-analysis for Oconee as part of steam generator replacement.

On October 23, 2000, the NRC transmitted a Request for Additional Information. Attachment 1 provides Duke's responses to these questions.

On January 25, 2001, a conference call was held to provide clarification to the NRC staff on the loop re-analysis methodology. From this discussion, two additional questions were raised. Attachments 2 and 3 provide responses to these questions.

The loop re-analysis has been performed by Framatome ANP (FRA-ANP). It was expected that FRA-ANP could reproduce the original Bechtel Amplified Response Spectra (ARS) for the attachment points on the Interior Concrete Structure and the Once Through Steam Generators, as described in Section 6.0 of the August 28, 2000 letter. However, the methods used by Bechtel in the original analysis to generate the ARS are not available through FRA-ANP certified computer programs used today. Therefore, FRA-ANP instead performed a comparative analysis as described in Section 6.0 of Attachment 4. Attachment 4 is an update that replaces the previously submitted Methodology for Analysis of

A001

the Reactor Coolant Loop for Steam Generator Replacement in its entirety.

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

Very truly yours,

W. R. McCollum, Jr.
Site Vice-President,
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xc: (w/attachments)

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Attachment 1

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

Response to NRC Request for Additional Information Dated October 23, 2000

1. *Per your submittal, you indicated that breaks in large bore primary piping has not been considered because the NRC has approved BAW-1847, Rev. 1. Provide the following information to demonstrate that the analyses and results of BAW-1847, Rev. 1 still bounds the plant-specific applications of Oconee, Units 1, 2, and 3 after the steam generator replacement:*

Response

Changes in the Leak Before Break (LBB) loadings, as a result of the current loop re-analysis with the replacement steam generator (RSGs) and/or changes in piping/weldment materials that will be utilized, will be evaluated for Oconee Units 1, 2, and 3. If the results are not bounded by the current analysis, an LBB submittal will be made that will summarize the results of this evaluation with comparison to the results of BAW-1847, Rev 1. A firm schedule is not yet available for this evaluation since it depends on future decisions for piping and weldment materials for installation of the RSGs. It is expected that this evaluation may be available by the end of 2001.

2. *Assess the impact on applicability of BAW-1847, Rev. 1 to Oconee, Units 1, 2, and 3 due to any change of loading (dead weight, transients, OBE, and SSE) caused by the steam generator replacement.*

Response

A comparison of the LBB load sets post steam generator replacement to those used in the design basis LBB evaluation will be performed (see Section 9.0 of Attachment 4).

3. *Assess the impact on applicability of BAW-1847, Rev. 1 to Oconee, Units 1, 2, and 3 due to any piping material (base metal and weld) degradation and aging that might have occurred during the past 15 years of operation of Oconee, Units 1, 2, and 3. This evaluation should include the change of the pipe size (dimension) and the change of material properties such as the flow stress, Ramberg-Osgood parameters, and the fracture toughness (J or K_{ic}) due to degradation and aging.*

Response

In 1998, Framatome ANP (FRA-ANP) performed an LBB evaluation of cast austenitic stainless steel (CASS) pump casing nozzles (discharge and suction nozzle) to address the effects of thermal aging on the RCS primary piping for the license renewal application of Oconee Units. The results of this evaluation are summarized in an RAI response (RAI 5.4.1-1) and briefly discussed below.

Test data obtained by Argon National Laboratory (ANL) [O. K. Chopra and W. J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U.S. Nuclear Regulatory Commission, Washington DC, May 1994], indicate that prolonged exposure of CASS to reactor coolant operating temperatures can lead to reduction of fracture toughness by thermal embrittlement. The fracture toughness curves for the ferritic base metal and ferritic weld metals used in the Reactor Coolant System piping leak-before-break analysis were compared to the lower-bound fracture toughness curves of Oconee reactor coolant pump CASS materials (i.e., statically cast CF8 and CF8M) from the ANL report. The fracture toughness curve of the lower-bound CASS material is below the fracture toughness curves used in the Reactor Coolant System piping leak-before-break analysis. Therefore, the assumption in BAW-1847, Revision 1, that the fracture toughness of the ferritic piping and ferritic weldments bounds the fracture toughness of CASS materials cannot be supported.

A flaw stability analysis was performed using the lower-bound CASS fracture toughness curves from the ANL report cited above to show acceptability of leak-before-break for the Reactor Coolant System main coolant piping for the period of extended operation. The most limiting material and location used in the Reactor Coolant System piping leak-before-break analysis (i.e., BAW-1847) was determined to be the base metal material of the straight section of the 28-inch cold leg pipe. Both the suction and discharge nozzles of the reactor coolant pump casings are attached to the 28-inch cold leg pipes and have similar geometry and loading applied to them as the limiting location used for the leak-before-break analysis. The discharge and suction nozzles of the reactor coolant pump casings were evaluated for leak-before-break using lower-bound CASS fracture toughness properties.

Bounding 10 gpm leakage crack sizes for the reactor coolant pump suction and discharge nozzle were determined using a methodology that is consistent with that reported in BAW-1847, Revision 1. The leakage crack length (twice the leakage flaw size) for the suction nozzle was determined to be 8.62 inches and the leakage crack length for the discharge nozzle was determined to be 8.86 inches. A flaw stability analysis for the reactor coolant pump inlet and exit nozzles was conducted, and the discharge nozzle was found to be limiting. The maximum applied J value at the discharge nozzle, for the 10 gpm leakage flaw size, was determined to be 0.510 kips/in. The margin on flaw size was determined to be 2.4, which is greater than the required margin of 2 in accordance with SRP 3.6.3.

The LBB loads comparison task (see response to question 1) will identify whether or not the loadings at the suction and discharge nozzles as a result of steam generator replacement are bounded by the existing loadings applicable at these locations. If they are not bounded, a re-evaluation of the pump casing nozzles will have to made.

4. *Summarize your effort in Tables similar to Table 4-11 and Table 4-12 of BAW-1847, Rev. 1. Remember that the staff is not asking you to repeat the analyses of BAW-1847, Rev. 1 but to apply your engineering judgement at every critical step to estimate the final safety factors for the primary piping of Oconee, Units 1, 2, and 3*

Response

A limit load analysis for the critical base metal and weldment locations in the RCS primary piping hot leg and cold leg piping will be performed. The analysis will include the more appropriate flow stress data based on consideration of currently available industry data. The results will be summarized and compared with the results summarized in Tables 4-11 and Table 4-12 of BAW-1847, Rev. 1.

Attachment 2

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

Response to NRC Request for Additional Information
Telecon of January 25, 2001

1. *Duke Power's August 28, 2000 submittal states that the re-analysis of the reactor coolant loop using current analytical approach is similar to the re-analysis that was done for Catawba Unit 1 and McGuire Units 1 and 2 in conjunction with their steam generator replacement projects, which was reviewed and approved by the NRC in an SER dated April 8, 1993. Describe, in detail, the differences in the following aspects between those for Oconee Unit 1, 2, and 3, and those for McGuire/Catawba loop re-analysis methodology, and how those differences are adequately addressed and resolved in the final results:*

   (1) Reactor Coolant System (RCS) structural model development and the computer codes used.
   (2) Development and analysis of the RCS, replacement OTSG (ROTSG), and reactor building hydraulics models including jet-impingement and thrust analysis.
   (3) Structural loading analysis approach and assumptions (Section 5.0 of the submittal) in general, and Seismic Loading (Subsection 5.4) in particular.

Response

See Attachment 3

2. *Section 13.0 (Computer Codes) of the submittal lists the computer codes used for ROTSG analysis. Provide information with respect to the verification, validation, and benchmarking of these computer codes.*

Response

BWSPAN, CRAFT2, COMPAR2, BWHIST are codes developed by Framatome ANP (FRA-ANP). FRA-ANP developed and maintains these codes, which are used for safety related work. These codes are certified for use through an established FRA-ANP corporate procedure. This procedure prescribes how the software is to be tested (benchmarked), what documentation is required, and how the required documentation is to be maintained. Once a code meets all of the certification requirements, it is made accessible for use but its source code is protected. Each run of a certified code will produce a header in the output, which gives the certification status of the code.

<table>
<tr><th>Question</th><th>Topic</th><th>McGuire/Catawba</th><th>Oconee</th><th>Comparison</th><th>Analysis of Differences</th></tr>
<tr><td rowspan="4">1.1.a</td><td rowspan="4">Structural Model Development</td><td colspan="2">Interior Concrete structure modeled with lumped mass, benchmarked against design basis model.</td><td>Same</td><td></td></tr>
<tr><td>Components modeled with distributed mass, internals not explicitly modeled (mass of internals is considered).</td><td>Components modeled with distributed mass, only the ROTSG internals explicitly modeled (mass of RV, RCP and Pressurizer internals is considered).</td><td>Different</td><td>The ROTSG internals were included in the Oconee structural model in order to provide internals loads. Modeling the internals explicitly or distributing their mass does not significantly affect the results calculated for the RCS components, piping and supports.</td></tr>
<tr><td colspan="2">Piping modeled as distributed mass.</td><td>Same</td><td></td></tr>
<tr><td colspan="2">Component support steel made up of AISC shapes or cylinders explicitly modeled, including distributed mass. Other supports represented by springs and lumped mass.</td><td>Same</td><td></td></tr>
</table>

Question	Topic	McGuire/Catawba	Oconee	Comparison	Analysis of Differences
		Surge line/pressurizer excluded consistent with original design basis analysis.	Surge line/pressurizer included consistent with original design basis analysis.	**Different**	Difference is due to different philosophies of the two Original Equipment Manufacturers (OEM). Either method gives accurate results for the RCS components, piping and supports (given the surge lines small size relative to the hot leg).
		Water and insulation weight included in component and piping models as distributed mass.		**Same**	
		CRDM, cooling shroud and seismic platform weight lumped above the RV head.	CRDM and Service Support Structure explicitly modeled.	**Different**	Difference is due to different philosophies of the two OEMs. Either method gives accurate results for the RCS components, piping and supports (given the small mass of the CRDMs and related equipment relative to the RV).
1.1.b	Computer Codes Used	BWSPAN (**FRA-ANP** structural analysis code)		**Same**	

Question	Topic	McGuire/Catawba	Oconee	Comparison	Analysis of Differences
1.2.a	Hydraulic Model Development	RCS, RSG, RB cavities subdivided into appropriate control volumes which are connected by flow paths.		**Same**	
		Controlling guillotine breaks in the piping attached to the primary and secondary sides are considered (large bore primary piping breaks eliminated by LBB).		**Same**	
		Break transient time is 0.5 seconds.		**Same**	
		Time step is 0.00005 seconds for the first 0.0005 seconds and 0.0005 seconds thereafter.	Time step is 0.00001 seconds for the entire transient.	**Different**	Either time step is small enough to yield accurate results.
		Break opening times calculated based on pipe acceleration considerations.		**Same**	
		Leak flow paths are calculated using Zaloudek-Moody with a discharge coefficient of 1.018 and a dimensionless multiplier of 0.9. Maximum quality to use a linear interpolation between Zaloudek and Moody is 0.0001.	Leak flow paths are calculated using Zaloudek-Moody with a discharge coefficient of 1.0 and a dimensionless multiplier of 0.81. Maximum quality to use a linear interpolation between Zaloudek and Moody is 0.02.	**Different**	The parameters selected for the McGuire/Catawba were based on previous analyses for those plants. The parameters selected for Oconee were deemed to be reasonable values based on experience.

<table>
<tr><th>Question</th><th>Topic</th><th>McGuire/Catawba</th><th>Oconee</th><th>Comparison</th><th>Analysis of Differences</th></tr>
<tr><td></td><td></td><td colspan="2">Jet Impingement calculated for credible targets using ANSI 58.2 methodology.</td><td>Same</td><td></td></tr>
<tr><td></td><td></td><td>Operating pressure at time of break multiplied by break area is hand calculated at the break locations and applied in the structural model to represent thrust loading.</td><td>Thrust force is calculated by computer code CRAFT2.</td><td>Different</td><td>Experience has shown that the thrust force calculated by CRAFT2 is approximately equal to the operating pressure at time of break multiplied by break area.</td></tr>
<tr><td>1.2.b</td><td>Computer Codes Used</td><td colspan="2">CRAFT2 (FRA-ANP hydraulics code) used for the RCS and RSG models.
COMPAR2 (FRA-ANP hydraulics code) used for the RB cavity model.
BWHIST (FRA-ANP post processor) converts pressure time histories into force time histories for application in the structural model.</td><td>Same</td><td></td></tr>
</table>

Question	Topic	McGuire/Catawba	Oconee	Comparison	Analysis of Differences
1.3	Structural Loading Analysis and Assumptions	100% power conditions considered: deadweight, thermal expansion, pressure expansion. OBE and SSE considered as well as the bounding pipe breaks (excluding breaks in the large bore primary piping which have been eliminated through the application of LBB).	100% power conditions considered: deadweight, thermal expansion. OBE and SSE considered as well as the bounding pipe breaks (excluding breaks in the large bore primary piping which have been eliminated through the application of LBB).	**Different**	Pressure expansion is not a design basis loading for the B&W designed units. Loads due to pressure expansion for McGuire/ Catawba were small. Pressure is considered in the stress calculations for all units.
		Load combinations are the same as for the design basis analysis.		**Same**	
		Seismic loading is analyzed via the response spectrum method with the basemat spectra being applied at the base of the interior concrete structure model.		**Same**	
		Damping (N-411 and RG 1.61), modal combinations (SRSS, considering closely spaced modes) and earthquake direction combinations (max (X+Y,Y+Z)) are per the original design basis.	Damping (FSAR specified, less than RG 1.61), modal combinations (SRSS) and earthquake direction combinations (X+Y and Y+Z reported) are per the original design basis.	**Different**	The differences in seismic analysis methods arises out of the differences in the original design basis specifications.

Attachment 4

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 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 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 is 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

2. By showing that the stresses, which are present after the ROTSG is introduced into the Reactor Coolant System, 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 Unit 2/3 Reactor Coolant Systems (RCSs) are developed using the Framatome ANP (FRA-ANP) 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 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 are modeled include the RV, RCP assembly (pump, motor stand and motor), ROTSG and pressurizer. Centerline models are used to represent the components. Local flexibilities are included in the model for each component at the RCSpiping connections and for the pressurizer at the support connections to the shell. These are calculated using methods developed by P.P. Bijlaard (see Reference 3, for example).

The RCS piping that is modeled includes the hot legs, lower cold legs, upper cold legs and surge line. Centerline models are used to represent the piping. Nominal dimensions are 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, is decoupled from the RCS model on the basis of the guidance given in Welding Research Council Bulletin 300, "Technical Position on Industry Practice" (Ref. 4).

The RCS component supports that are 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 are represented as centerline models. Each of the SGUSs is represented as a set of five springs, one for each "spur" of the SGUS. The RCP snubbers and spring hangers are represented as springs. The structural steel members which support the RCP hangers and snubbers are modeled as beams with the appropriate cross section properties. The structural steel members which make up the pressurizer support frame are modeled in a similar fashion.

The CRDMs are 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) are also 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 is 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 (see Reference 5). An "isolated" model of the ICS, similar to the model described in Reference 5, is developed in BWSPAN. Frequencies and mode shapes are calculated for this isolated model and these are 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 is inserted into the model to be used in the loading analysis of the RCS.

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

Current modeling techniques are 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 are 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 (Ref. 9) and Standard Review Plan Section 6.2.1.2 of NUREG 0800 (Ref. 10).

The RCS, ROTSG secondary side and reactor building initial conditions are those at 100% power for the HELBA analyses. Temperatures and pressures are taken from the ROTSG Certified Design Specification (Ref. 1) or other documentation provided by BWC.

4.1 <u>RCS Hydraulics Model and Analysis</u>

A thermal hydraulic model consisting of a network of fluid control volumes and flow paths is developed for the complete RCS using the FRA-ANP hydraulic code, CRAFT2. The model, after being initialized to normal steady state pressures and temperatures, is 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 Loss of Coolant Accident (LOCA). The force time histories are oriented to the global coordinate system for input to the structural model using the FRA-ANP computer code, BWHIST.

The hydraulics analyses of the RCS consider the controlling breaks which remain after application of LBB:

- Decay heat line break at the hot leg nozzle
- Terminal end surge line break at the hot leg nozzle
- Terminal end surge line break at the pressurizer nozzle
- Stress-induced intermediate surge line break - The controlling intermediate breaks are established and analyzed.
- Core flood line break at the RV nozzle

Mass and energy release data generated in the analyses for these breaks is 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, are 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 is developed for the ROTSG using the FRA-ANP hydraulic code, CRAFT2. The model, after being initialized to normal steady state pressures and temperatures, is 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 High Energy Line Break Accident (HELBA). The pressure time histories are converted to force time histories by multiplying them by the appropriate area, and are oriented to the global coordinate system for input to the structural model using the FRA-ANP computer code, BWHIST.

The ROTSG hydraulics model accounts 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 also minimizes the pressure drop loading on the generator and its internals.

Hydraulics analyses of the ROTSG consider "Single" and "Double" Main Steam Line Break (SMSLB and DMSLB) and 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 SFWLB is postulated at the juncture of the 14" riser pipe and the 14" tee in the feedwater header. The DFWLB is postulated at the juncture of the two main feedwater lines from a given generator such that both main feedwater lines blowdown. These breaks are the controlling secondary side breaks.

Mass and energy release data generated in the analyses for these breaks are 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, are 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 is not 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 are developed for the reactor vessel and steam generator cavities within the reactor building. Component, equipment and structural steel volumes in the cavities are accounted for, that is, they are subtracted from the control volumes representing the cavities. These models, with mass and energy release data from the RCS and ROTSG hydraulic analyses, are used to determine pressure time histories in the control volumes using the FRA-ANP hydraulic codes, CRAFT2 (reactor cavity) and COMPAR2 (steam generator cavities). These pressure time histories constitute ACP loading on the RCS components. The pressure time histories are converted to force time histories by integration over appropriate surface areas of the components, and oriented to the global coordinate system for input to the structural model using the FRA-ANP computer code, BWHIST.

ACP loading are generated for all of the primary side breaks listed in Section 4.1. ACP loading are also generated for the SMSLB, SFWLB, and DFWLB 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 is 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 are used to determine the JI forces acting on the RCS components which are credible targets. ANSI Standard 58.2 (Ref. 12) methodology is used to calculate the JI forces. JI loading is 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 are also calculated. Thrust load time histories are provided for all of the primary and secondary side breaks listed in Sections 2)a and 2)b except for the DMSLB and the DFWLB: the DMSLB and DFWLB 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

FRA-ANP considers 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 are performed using the BWSPAN model of the RCS and use properties at 100% operating conditions unless otherwise noted.

5.1 Pressure Loading

Design (2500 psi) or operating (2155 psi) pressure is 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 are considered as distributed mass in the deadweight analysis. Mass of the component supports, such as the support skirts and structural steel beams, are considered as distributed or lumped mass, as appropriate. The mass of the ICS is considered as lumped mass in the model in keeping with the original ICS analysis (see Reference 5). Other mass supported by the RCS, such as whip restraints and feedwater headers, is considered as lumped mass if it is deemed significant.

5.3 Thermal Expansion

Four thermal expansion load cases are considered for normal operating conditions: the 0%, 8%, 15% and 100% power conditions. The 0% and 8% power conditions are considered to provide a lower bound for thermal stress ranges in the fatigue calculations and the 15% thermal expansion case is considered because the cold leg temperature is highest at this power level. The RCS temperatures used for each expansion analysis are 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 (Ref. 6) are reviewed to identify the bounding overtemperature transient for the hot and cold leg.

5.4 Seismic Loading

Both the Operating Basis and Safe Shutdown Earthquakes are analyzed. Note that the ONS Design Basis Earthquake is referred to as OBE and the ONS Maximum Hypothetical Earthquake is referred to as SSE.

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

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

Damping is 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 is "2-Dimensional" and the results from both the X+Y and Y+Z combinations are reported, where X and Z are the two horizontal earthquakes. The X (or Z) earthquake results are 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 are applied to the structural model of the RCS. The breaks considered are listed above in the section describing the hydraulics analyses (Section 4.0). The LOCA restraints on the RCP assembly, the hot leg whip restraint, and the cold leg whip restraint are not 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 (Ref. 7) are used: 3% for equipment and piping >12"

OD, 2% for piping <= 12" 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 (see Section 8.0) are per the ROTSG Certified Design Specification. Load combinations for use in the load comparisons for the remaining RCS component nozzles (see Section 10.0) and the RV support skirt (see Section 11.0) are 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) are per the original design calculations. Load combinations for use in the RCS pipe stress calculations (see Section 9.0) 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 all cases where combination of SSE and HELBA loads is called for, the combination is by SRSS.

6.0 Plan for Comparative Analysis of ONS ROTSG/OTSG Dynamic Characteristics

6.1 Background

FRA-ANP developed a math model of the ICS and OTSG using BWSPAN to benchmark against the original Bechtel model from Duke document OSC-7298 (Reference 5). The frequencies, mode shapes, participation factors, and structural amplifications output from the BWSPAN model were compared to the output from the Bechtel model. All matched with excellent accuracy. A spectrum analysis is performed using the hand calculated ground spectrum accelerations from OSC-7298. As expected, the spectrum analysis gives the same joint accelerations as those found by Bechtel.

Based on the above information, it would be expected that FRA-ANP could reproduce the Bechtel ARS for the attachment points on the ICS and the OTSG. However, the methodologies used by Bechtel in the original analysis to generate ARS are not available through FRA-ANP certified computer programs used today. For these reasons, FRA-ANP performs a comparative analysis of the ROTSG/OTSG dynamic characteristics in lieu of generating new attachment point response spectra.

6.2 Comparative Analysis Models

The ROTSG model provided by BWC as input to FRA-ANP for the loop analysis is used in the comparative analysis. This model replaces the OTSG model in the isolated ICS BWSPAN model discussed above. The model is an exact duplicate of the ROTSG model used in the Oconee-1 loop analysis performed in FRA-ANP Document 32-5006604 (Reference 15). The only change is that additional structural joints are required to obtain results at the same elevations as provided by Bechtel.

Although this model is not developed in the same manner as the original model, this is the new model of record for the plant. Therefore, this new model should be used in all comparisons. The differences between the ROTSG model and the Bechtel OTSG model include the following:

BWC/FRA-ANP ROTSG	Bechtel OTSG
Distributed Mass	Lumped Mass
Explicit modeling of ROTSG internals	Shell only model of OTSG

6.3 Comparative Analysis

In the comparative analysis, a dynamic seismic analysis is performed by calculating frequencies, mode shapes, participation factors, structural amplifications and modal damping values for each of the two models. The basemat seismic spectra, provided by Duke document OSC-7298, are applied to the models to determine nodal accelerations. The modal damping for the two models is calculated by BWSPAN using bar strain energy weighted non-proportional methods. This is not the same method used in the original Bechtel analysis. The original Bechtel analysis used mass weighted non-proportional methods and the modal damping for each mode was calculated by hand. The new method does not give the same modal damping values, but for comparative purposes, in evaluating the effect of the ROTSG, this method is acceptable.

Note that the ROTSG model has many more degrees of freedom than the OTSG model. Therefore, this model gives more modes below the cutoff of 33 Hz. These extra modes cannot be compared to the original model except to show that they are the result of the explicit modeling of the ROTSG internals and should not greatly effect shell accelerations. This is shown in a comparison of shell nodal accelerations for the ROTSG and OTSG. Any differences in the nodal accelerations are evaluated and the original shell spectra produced by Bechtel are adjusted if necessary.

6.4 Acceptability

Comparative analyses have been used many times in the past to qualify the replacement of a seismically qualified component. In many cases, nothing more than the comparative analysis is performed to show that the component will not adversely affect the connected piping and components. However, in the case of Oconee, a full deadweight and seismic loading analysis, LOCA loading analysis, thermal transient analysis and loop stress analysis is performed to show that the ROTSG will not adversely affect the remainder of the RCS. Therefore, considering the depth of analytical work to be performed, this is an acceptable substitute for the generation of new attachment point acceleration response spectra.

7.0 RCS Loading Specification

FRA-ANP tabulate displacements and loads at key locations throughout the RCS which result from the analysis of the load cases described in Section 5.0. Displacements are provided at branch nozzle and whip restraint locations. Loads are reported at the primary nozzles (including both of the surge line nozzles) and at the RCS supports. Tabulation is performed using FRA-ANP's specification writer, BWSPEC. Note that the loads being presented here are actual analysis results and not allowables.

8.0 Confirm the ROTSG Design Loads and Force Time Histories

FRA-ANP makes the following comparisons to ensure that the design loads given in the ROTSG CDS (Ref. 1) envelop the actual loads, which result from the loading analysis:

Primary nozzle loads from the FRA-ANP loading analysis and secondary nozzle loads supplied by DPCO are compared to the design loads given in the ROTSG CDS.

Loads acting on the ROTSG from the upper and lower supports are 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 are compared to the design loads in the CDS.

9.0 Primary Piping Stress and Fatigue Analysis

The design code for the RCS piping is 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 are performed using the simplified pipe stress equations as given in Article NB-3600 of the 1983 Code. A code reconciliation is performed to assess the impact of this change. The basic material allowable stresses used are 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) are taken from the 1983 ASME Boiler and Pressure Vessel Code.

Stress analysis of the primary loop piping (including the surge line) is then performed. This is accomplished using FRA-ANP codes BWSPAN and T3PIPE. BWSPAN is used to calculate stresses due to pressure and mechanical loads at all piping locations. T3PIPE is 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 FRA-ANP) and thermal radial gradients/thermal discontinuity stresses (taken from existing analyses, hand calculated or generated using verified computer codes). Appropriate stress intensification factors are used for all geometries.

10.0 Primary Nozzle and Lug Load Comparisons/Stress Analysis

Loads generated by the FRA-ANP loading analysis for the RV and RCP primary nozzles, the CRDM nozzles, and the surge line nozzles are compared to the existing (pre-ROTSG) design loads. FRA-ANP analysis loads are taken from the loading specification described earlier and existing design loads are taken from the original stress reports for the component in question. Where the FRA-ANP analysis loads are higher, stress and fatigue analysis are 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 Unit 2/3 RCP.

Comparison of the FRA-ANP analysis loads on the ROTSG primary nozzles to the design loads contained in the ROTSG CDS is discussed in Section 8.0.

The FRA-ANP analysis loads on the pressurizer support lugs are compared to the design loads given in the FRA-ANP stress report for the pressurizer. If the FRA-ANP analysis loads are higher, stress analysis is performed in accordance with the original stress report and the design code.

11.0 Equipment Support Loads Comparison/Stress Analysis

Loads, taken from the FRA-ANP loading analysis, which act on the supports listed below, are 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 are 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 are calculated by FRA-ANP'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 is 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 are performed as described in Section 5.0.

12.0 Update Stress Report Summaries

FRA-ANP 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 FRA-ANP loading analysis described in Section 5.0 are 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 are added to the summary documents in all cases. Note that actual loads resulting from the FRA-ANP loading analysis of the RCS are contained in the loading specification described in Section 7.0.

13.0 Computer Codes

BWHIST: An FRA-ANP 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. This code also orients force time histories generated by CRAFT2 to the coordinate system of the structural model.

BWSPAN: An FRA-ANP developed code which performs structural analysis of piping and structural systems and B31.7 and Section III (ASME Boiler and Pressure Vessel Code) stress and fatigue calculations for Class 1 and 2 piping. BWSPAN can also calculate stresses for linear type supports according to Subsection NF of Section

III. Deadweight, thermal expansion, response spectrum, time history and thermal stratification loading can be analyzed. Output includes displacements, loads, acceleration and displacement time histories, stresses and fatigue usage factors, as appropriate.

BWSPEC: An FRA-ANP 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 FRA-ANP 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. This program is an FRA-ANP version of COMPARE-MOD1 which has been approved for use by the NRC.

CRAFT2: An FRA-ANP 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 FRA-ANP 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).

T3PIPE: An FRA-ANP 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).

14.0 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. DPCo Document OSC-7298, Revision D1, "Seismic Analysis of the Internal Structure".

6. FRA-ANP 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"