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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Non-Proprietary Response to Request for Additional Information from the Mechanical and Civil Engineering Branch Regarding the ANO-2 Power Uprate License Application

Gentlemen:

On August 23, 2001 (2CAN080104), Entergy Operations, Inc. submitted a response to a request for additional information from the Mechanical and Civil Engineering Branch of the Nuclear Regulatory Commission (NRC) regarding the ANO-2 power uprate license application dated December 19, 2000 (2CAN120001). The August 23, 2001, letter, included responses to thirteen questions from the NRC staff.

In the August 23, 2001, letter, the responses to NRC questions 1, 2, 4, 6 and 7 were categorized as proprietary information of Westinghouse Electric Company, LLC. Subsequent to submittal of the response, Westinghouse informed us that the responses to questions 4 and 6 did not, in fact, contain proprietary information and should not have been treated as such. Therefore, the attachment to this letter contains unedited, non-proprietary responses to these two questions. Responses to questions 1, 2 and 7 are also included with the proprietary information removed. Brackets indicate those areas where proprietary information has been removed. As stated in "a" through "e" of item vi of the affidavit included with the August 23, 2001, letter, the information in the responses to these three questions is considered to be proprietary and should be withheld from public disclosure. For completeness, the attachment also reiterates the responses to the remaining questions.

Correspondence regarding the proprietary aspects of the information contained in this correspondence should be addressed to Mehran Golbabai, Project Manager, ANO-2 Power Uprate, Westinghouse Electric Company, LLC, 2000 Day Hill Road, Windsor, CT 06095.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 13, 2001.

Very truly yours,

Glenn R. Ashley

Glenn R. Ashley
Manager, Licensing

GRA/dwb
Attachment

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Non-Proprietary Responses to the Mechanical and Civil Engineering Branch Request for Additional Information Regarding the ANO-2 Power Uprate

NRC Question 1

In reference to Section 5.3.3.2 of the application, provide the calculated maximum stresses and fatigue usage factors at the critical locations of the control element drive mechanisms for all operating conditions shown in Table 5-3 as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

ANO Response

LEVEL A and B (NORMAL OPERATING AND UPSET CONDITIONS)

LEVEL C (EMERGENCY CONDITIONS)

LEVEL D (FAULTED CONDITIONS)

NRC Question 2

Section 5.4 describes the mechanical and thermal analyses performed to determine the response of the reactor cooling system (RCS) main coolant loop and components, including the reactor vessel (RV), reactor coolant pumps (RCPs), replacement steam generators (RSGs), hot and cold leg piping, and component (RV, RCP, pressurizer and RSG) supports. The piping is discussed separately in Section 5.8. Provide the methodology, assumptions and loading combinations used for evaluating the RV, the pressurizer, the RCPs, the RSGs and their supports. Also provide the calculated maximum stresses and cumulative usage factors at critical locations of each component for the power uprate condition, including the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

ANO Response

Prior to performing the structural integrity analyses of the reactor vessel (RV), pressurizer, reactor coolant pumps (RCPs), and their supports, CENP determined new NOp, seismic, and loss of coolant accident (LOCA) loads on the reactor coolant system for replacement steam generator (RSG) and power uprate conditions (structural analyses performed for RSG included power uprate). New deadweight (DW) and thermal expansion loads (i.e., the load components comprising the NOp loads) were determined by applying conservative factors to the existing DW and thermal loads and calculating new values. New seismic and LOCA loads were determined by analysis. Structural models of the RCS with appropriate boundary conditions were developed, along with the necessary input time history loadings. These models were then analyzed with the ANSYS computer code.

The as-calculated loads form the basis for determining conservative design loads for the various RCS components and their supports. It is concluded that normal operating conditions (pressure and temperature) have not changed enough to affect the allowable stress values used in the analysis of record (AOR).

In a further step, the effects of RSG and power uprate on plant transients were determined. Transients for Heatup/Cooldown, Reactor Trip, Loading/Unloading and Loss of Secondary Pressure were revised due to RSG and power uprate conditions. Except as noted below, the revised transients do not impact the existing stresses or Cumulative Usage Factors (CUFs) and the existing analyses remain valid.

Details of the structural integrity analyses performed for the reactor vessel pressurizer, reactor coolant pumps, replacement steam generators and their supports are presented in the following sections.

REACTOR VESSEL (RV)

The loads and transients impacting the RV are incorporated in a revision to the RV specification. RV components impacted by the revision were reanalyzed with the updated loads using the methodologies employed in the AOR. RV components not impacted by RSG and power uprate include the vessel wall transition, the vessel wall–bottom head junction, core stop lugs, flow skirt incore instrumentation nozzles, and control element drive mechanism (CEDM) nozzles.

The following results were obtained for the critical areas.

Closure Head Flange Region

1. Primary + Secondary stress range for studs is 78.1 ksi which is less than the allowable of 94.0 ksi. The CUF for the RV studs increased from 0.9291 to 0.9374 due to revisions in the plant transients. However, this value remains < 1.0.

2. Faulted loads analysis: prior to RSG and power uprate, faulted conditions were not required to be analyzed in this region. The resulting calculated stress on limiting closure head/flange is 77.0 ksi which is less than the allowable of 99.2 ksi.

Reactor Vessel Inlet and Outlet Nozzles

1. Current inlet nozzle results are bounded by AOR results with the exception of the CUF. The new CUF is 0.167, which remains below the maximum allowable of 1.0.

2. All current outlet nozzle results are bounded by AOR. The CUF is 0.0853, which is less than 1.0.

Reactor Vessel Nozzle Supports

The CUF of 0.0002 from the AOR remains bounding. Table 2-1 summarizes the RSG and power uprate analysis results for all conditions:

Table 2-1

		P_m (ksi)		**P_{m+b} (ksi)**	
		Calculated	**Allowable**	**Calculated**	**Allowable**
NOp	Inlet	3.7	26.7	6.1	40.1
	Outlet	3.9	26.7	6.5	40.1
Upset	Inlet	6.5	26.7	8.3	40.1
	Outlet	9.8	26.7	11.6	40.1
Faulted A(1)	Inlet	9.0	42.6	10.8	63.9
	Outlet	18.4	41.9	19.6	62.9
Faulted B(2)	Inlet	18.3	42.6(3)	17.4	63.9(3)
	Outlet	32.8	41.9(3)	33.9	62.9(3)

where P_m = Primary membrane stress intensity
P_{m+b} = Primary membrane plus bending stress intensity

Notes for Table 2-1:

1. Faulted A is the combination of a) NOp loads, b) control rod scram, lift rig and shroud loads, and c) operating basis earthquake loads.
2. Faulted B is the combination of a) NOp loads, b) design basis earthquake loads, c) LOCA loads, d) control rod scram, lift rig and shroud loads, and e) core drop loads.
3. The allowables for Faulted B are conservatively shown to be the same as for Faulted A.

<u>Reactor Vessel Column Supports</u>

The results of the evaluation are summarized below.

1. RV Columns

   a) The axial and bending load ratios for all loading conditions were found to be less than the maximum allowable value of 1.0.

   b) The fatigue analysis was performed in accordance with AISC Code, Appendix B, and was based on a total number of 1200 cycles due to heatup, cooldown and seismic. The permissible stress range for these conditions is conservatively taken as 1.5 times the applicable value given in Table B3 of the AISC Code for loading condition 1. This produces a limiting value of 1.5 * 40 = 60 ksi.

   The maximum stress range due to NOp + OBE conditions, 14.17 ksi, is less than maximum allowable range of 60 ksi. Therefore, the fatigue usage factor requirements for the columns are satisfied.

2. RV Column Assembly Components

Table 2-2

	Actual (ksi)	**Allowable (ksi)**
Base Plate Bending Stress	35.4	44.9
Base Plate Shear in Ribs	1.3	18.0
Base Plate Anchor Bolts		
Tensile Stress	55.2	75.2
Tensile + Bending Stress	67.8	112.8
Column Upper Flange Bending Stress	39.0	43.2
Upper Flange Shear Key	13.5	23.0
Upper Flange Interface Bolts		
Tensile stress	63.3	75.2
Tensile + Bending Stress	70.7	112.8
Shear Stress	24.7	27.1
Bearing Stress	51.6	51.8

Core Stabilizer Lugs

The faulted loads were revised for RSG and power uprate. The following results were obtained:

Table 2-3

Location	**Condition Evaluated**	**Actual (ksi)**	**Allowable (ksi)**
Lug	Bending	28.2	49.4
Shell	Membrane	25.2	41.4
Shell	Bending	36.6	39.1
Shell	Stress Intensity	51.3	96.8

The Code and Code edition used for the RV evaluations is the 1968 Edition of the ASME B&PV Code, up to and including the Summer 1970 Addenda. This is the Code of Record.

PRESSURIZER

The only pressurizer subcomponent requiring further analysis for RSG and power uprate conditions is the surge nozzle, which is affected by RCS piping load and transient changes. The current methodology for calculating loads due to branch line pipe breaks creates a LOCA load that was not previously defined for the surge nozzle. Therefore, a faulted load case including this new surge nozzle load case was analyzed. Resulting stresses are summarized below:

P_m

Cut B: 12.9 ksi < 1.2*Sm, 22.08 ksi

P_L+P_b

Cut B: 17.6 ksi < 1.8*Sm, 33.12 ksi
Cut J: 22.0 ksi < 1.5*Sy, 60.90 ksi

where P_m = Primary membrane stress intensity
P_L = Primary local membrane stress intensity
P_b = Primary bending stress intensity

The inclusion of LOCA loads does not affect the fatigue evaluations performed in the analysis of record.

The Code and Code edition used for the pressurizer surge nozzle evaluation is the 1968 Edition of the ASME B&PV Code, up to and including the Summer 1970 Addenda. This is the Code of Record.

REACTOR COOLANT PUMP/MOTOR

Reactor Coolant Pump

1. Methodology

The pump vendor, Byron Jackson Corp, performed the AOR for the RCPs. To confirm that the AOR remained valid for the condition of RSG with power uprate, the following approach was taken.

a) Existing information was assessed by:

    1. Reviewing the stress margin survey generated for the RSG to determine those pump locations having low stress margins,

    2. Comparing the new NOp, seismic, and faulted loads to the loads used as input to the AOR.

b) Those pump locations where new stresses might exceed allowables were selected for closer examination. Any location with increases in one or more of the loads that comprised a loading condition was considered. If that location also had a moderate or low stress margin, further examination was required.

c) Further analysis or load reconciliation was performed for the selected locations to show that the increased loads are acceptable.

2) Results

This process identified the following locations:

- Suction nozzle body
- Vanes 1, 8 and 9 (inside of casing)
- Hanger bracket #2
- Discharge nozzle crotch region
- Driver mount (and the closure studs securing the driver mount to the casing)
- Pump casing seal closure

Of these locations, hanger bracket #2 required further analysis in the vicinity of the bracket gusset. A finite element analysis using RSG with power uprate loading conditions demonstrates that the maximum casing membrane stress of 13 ksi, the critical stress in the region, is below the allowable value of 1.0 Sm = 18.7 ksi.

Higher design basis loads at the other locations due to RSG and power uprate are reconciled by performing quantitative and qualitative assessments to determine the impact of changes in load component magnitudes and, in some cases, the impact of changes in the overall direction of the loading.

<u>Reactor Coolant Pump Motor</u>

The effect of RSG and power uprate on the RCP motors was assessed by comparing the RCP accelerations under the revised conditions with the limits defined in the RCP Specification. The maximum accelerations due to OBE, safe shutdown earthquake (SSE) and loss of coolant accident (LOCA) are found to be less than the specified limits.

LOCA accelerations were assessed by making a "square root sum of the squares" (SRSS) combination of the most severe design basis earthquake (DBE) and LOCA accelerations. This faulted load combination was compared to the DBE allowable limits, thus demonstrating in a conservative fashion that the faulted condition g levels are acceptable for the RSG with power uprate conditions.

Table 2-4 summarizes the results of the seismic and faulted condition load comparisons.

Table 2-4

Type of Earthquake (excitation)		Horizontal and Vertical Response (g's)*	Design Basis Limit (g's)
OBE ±X±Y	**h**	0.50	1.5
	v	0.35	1.0
OBE ±Y±Z	**h**	0.81	1.5
	v	0.50	1.0
DBE ±X±Y	**h**	0.88	3.0
	v	0.56	2.0
DBE ±Y±Z	**h**	1.36	3.0
	v	0.80	2.0
Faulted	**h**	2.90	3.0
	v	1.84	2.0

*The horizontal response is the SRSS of the X and Z responses.
Vertical response is the Y direction response.

The Code and Code edition used for the preceding RCP/Motor evaluations is the 1965 Edition of the ASME B&PV Code, up to and including the Winter 1967 Addenda. This is the Code of Record.

REPLACEMENT STEAM GENERATOR

Both the primary and secondary pressure boundaries (including tubes) of the steam generators are designed to satisfy the criteria specified in Section III of the ASME B&PV Code for Class 1 components. Detailed structural analyses have been performed using loadings obtained from the applicable steam generator certified design specification. Stress states due to the applied loads have been calculated using classical analytical methods or finite element analysis. The results of the analyses demonstrate that the maximum stress intensities and cumulative fatigue usage factors are in compliance with the ASME B&PV Code requirements.

The replacement steam generators were originally designed for power uprate conditions. The Code and Code edition used in the evaluation for power uprate is the 1989 Edition, no Addenda of the ASME B&PV Code, Section III, which is the Code of record for the replacement steam generators. Accordingly, no reconciliation was required for power uprate. (Note that a reconciliation was performed between the Code of record for the OSGs and the Code of record for the RSGs).

The loading combinations used in the steam generator evaluations are listed in Table 2-5. A summary of the calculated maximum stresses and cumulative usage factors at critical locations, as well as the allowable Code limits, is provided in Table 2-6.

Table 2-5 Loading Combinations for Steam Generator Structural Evaluations

Condition	**Load Combinations**
Design	Design Pressure + Deadweight + OBE
Level A (Normal)	Deadweight + Normal Transients
Level B (Upset)	Deadweight + Upset Transients (includes OBE)
Level C (Emergency)	Deadweight + Emergency Transients
Level D (Faulted)	Deadweight + Operating Pressure + (combined DBE and Pipe Rupture *)
Test	Deadweight + Test Pressure

* Pipe Rupture is either LOCA, Steam Line Break or Feedwater Line Break

Table 2-6 Summary of Steam Generator Structural Evaluations

Table 2-6 (continued)

Summary of Steam Generator Structural Evaluations

STEAM GENERATOR SUPPORTS

The SG upper supports include the snubber arrangements (i.e., the snubbers, link, lever, lever bracket, and pins) and the upper Z keys. The SG lower support system consists of the sliding base, vertical support pads and anchor bolts, lower Z keys, and lower X stop. The vertical anchor bolts and the lower X stop will not be active for RSG and power uprate conditions.

SG Upper Supports

An assessment of the snubber arrangements was made for the RCS considering the effects of RSG and power uprate. The assessment considered the impact of RSG and power uprate on Accident loadings, which are used to size the SG upper supports. The as-calculated load on a snubber/lever system for RSG and power uprate conditions was determined to be 879 kips, which remains well below the limiting load of 2500 kips. Therefore, the existing snubber/lever system hardware is more than adequate for the new RCS configuration.

A similar accident load assessment was performed for the upper Z keys and keyways. The as-calculated load of 505 kips per key is well below the limiting load of 2500 kips. Therefore, the existing keyway hardware is also more than adequate for the new RCS configuration.

The Code and Code edition used for the OSG evaluation is the 1968 Edition of the ASME B&PV Code, up to and including the Summer 1970 Addenda. This is the Code of Record.

SG Sliding Base and Bolts

Since the sliding base is essentially at ambient temperature, thermal stresses are not impacted by RSG and power uprate. Changes in sliding base support loads were considered relative to the original analysis. Because some components of load increased while others decreased, the stresses in the base were reanalyzed using the methodology defined in the AOR. The results are shown below.

Table 2-7

The SG pedestal/flange unit was redesigned for the RSG effort and, therefore, a reanalysis of the critical stresses at the interface between the RSG pedestal/flange and the sliding base was required. This was done by analyzing an axisymetric finite element model **(**FEM) of the new pedestal/flange geometry and the bolts for the new loads due to RSG with power uprate. Boundary conditions accounted for the effects of interactions at the critical interfaces (i.e., pedestal/flange - sliding base bolt circle and pedestal/flange – bolt - nut). The resulting stresses in the bolts for various loading conditions are summarized below.

Table 2-8

The Code and Code edition used for the preceding SG sliding base and bolt evaluation is the 1971 Edition of the ASME B&PV Code, up to and including the Winter 1973 Addenda. This is the Code of Record for the sliding base and bolts. The pedestal/flange Code of Record is the 1989 Edition of the ASME B&PV Code, no addenda. This same Code of record is used for the RSGs.

NRC Question 3

As a result of the RSGs and the power uprate, the feedwater flow and pressure in the feedwater system have to increase from those required for the RSGs at the current and uprate [sic] power levels. Discuss the potential for flow-induced vibration of the RSG tubes due to various mechanisms, including, in particular, the fluid-elastic instability in the RSGs at the current power level. Provide an evaluation of the flow-induced vibration of the tubes in the RSGs at the power uprate condition regarding the analysis methodology, damping value of the tubes and the computer code used in the analysis, results of the predicted vibration levels during the normal operating condition and the worst case transient condition, and the calculated fluid-elastic instability ratios. Explain whether or not the current analysis considers the potential for a possible degraded RSG condition.

ANO Response

The replacement steam generators were specified to be designed and analyzed for power uprate conditions including consideration of tube vibration and wear degradation. A general discussion of this is included in the response to Question 2.a of our letter dated August 7, 2001, "Response to Request for Additional Information from the Materials and Chemical Engineering Branch Regarding the ANO-2 Power Uprate License Application" (2CAN080101). To address the more detailed question above, the following additional information is provided.

Analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the replacement steam generators when operated at power uprate flows. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

Each replacement steam generator has eight tube support plates and five sets of anti-vibration bars with advanced design features. Alloy 690 thermally treated tube material and 405 stainless steel tube support material were selected to enhance the resistance to corrosion, mechanical wear, and fatigue. Anti-vibration bar widths are wider than in previous conventional designs to reduce wear potential. Accordingly, the increased steam flow rate following power uprate is not expected to result in a change in original design margin to instability or tube wear degradation at the anti-vibration bars. The cross flow in the lower straight leg portions of the tube bundle does not change appreciably since the increased feedwater flowrate is offset by a reduction in recirculation flow. Therefore, the potential for vibration/wear in this region is not significantly affected by uprate. A discussion of the replacement steam generator (RSG) design relative to tube vibration follows.

Potential sources of tube excitation are considered in the design, including primary fluid flow within the U-tubes, mechanically induced vibration, and secondary fluid flow on the outside of the U-tubes. The effects of primary fluid flow and mechanically induced vibration were evaluated and are acceptable. The main source of potential tube degradation due to vibration is the hydrodynamic excitation of the tubes by the secondary fluid. This area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience. RSG thermal hydraulic modeling using ATHOS, a Westinghouse thermal hydraulic analysis code, determines the environmental conditions expected at the uprated power level. These environmental conditions are used as input to the RSG flow induced vibration evaluation.

Three potential tube vibration mechanisms related to hydrodynamic excitation of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluid-elastic vibration mechanisms.

Non-uniform, two-phase turbulent flow exists throughout most of the tube bundle. Therefore, vortex shedding is possible only for the outer few rows of the inlet region. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the inlet region. Bounding calculations consistent with laboratory test parameters confirm that vibration amplitudes are acceptably small, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the RSG.

Flow-induced vibrations due to flow turbulence are also small. Root mean square (RMS) amplitudes are consistent with levels measured in operating steam generators with benign tube wear experience. These vibrations cause stresses that are significantly below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation due to secondary flow turbulence is anticipated.

Fluid elastic tube vibration is potentially more severe than either vortex shedding or turbulence. Fluid-elastic tube vibration is a primary concern for anti vibration bar (AVB) wear. Testing performed by Westinghouse and field experience from previous designs have been utilized to develop analysis techniques to assure significant margin to instability is maintained. Linear dynamic analyses were performed covering a range of support configurations for various tubes using the finite element codes FLOVIB and FASTVIB. These are special purpose finite element codes that were written specifically for flow-induced vibration and fretting wear calculations for multi-span structural members. FLOVIB was written to incorporate the analytical approaches that were largely defined by the work of H. J. Connors at the Westinghouse Research Laboratories (later called Science and Technology Center). Three subprograms, SHAKE, GAMMA, and SUPER, comprise FLOVIB. Natural frequencies and mode shapes are determined in SHAKE. GAMMA uses SHAKE output and specified flow conditions in calculations of flow-induced vibration response of the structural member defined by beam elements. Peak and RMS values of selected GAMMA output parameters (displacements, stresses) are computed in SUPER.

Tube support spacing in the anti-vibration bars in the U-bend region provides tube response frequencies such that the required instability threshold limit is not exceeded for power uprate secondary fluid flow conditions. This approach provides large margins against initiation of fluid-elastic vibration for tubes, which are effectively supported by the tube support system. The largest stability ratio[1] for the Delta 109 steam generator tube bundle is 0.555 versus the conservatively specified 0.75 limit. The stability ratio is evaluated at power uprate loading conditions up to ten-percent tube plugging.

For the straight leg portion of the tubing, the worst case is where deposits are postulated to build between tubes and supports to the point where tube motion within the clearance is restricted or eliminated. For this case, frequency increases and damping reduces, so a separate evaluation was performed. When postulating limiting supports with reduced damping as a result of buildup, the largest stability ratio is 0.681.

Based on operating condition transient definitions, there are no Level A or B transients with higher than full-power steam flow, and normal operation analyses bound these transient conditions. Level C and D transients are short duration events and are evaluated for bending stresses due to vibration in accordance with ASME B&PV Code requirements.

The Regulatory Guide 1.121 analysis for technical specification tube plugging limits addresses degraded tubes during normal and accident conditions at the uprated condition, including flow induced vibration loadings. A summary of this analysis has been previously submitted to the NRC in a letter dated July 19, 2000, "Regulatory Guide 1.121 Analysis for Arkansas Nuclear One Unit 2 Replacement Steam Generators" (2CAN070007). A discussion

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[1] Stability ratio defined as Fluidelastic Stability Ratio (FSR) = effective velocity/critical velocity

of the effects of power uprate on RSG degradation was included in the response to Question 2.b of our letter dated August 7, 2001, "Response to Request for Additional Information from the Materials and Chemical Engineering Branch Regarding the ANO-2 Power Uprate License Application" (2CAN080101).

As outlined, analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the RSGs when operated at power uprate flows. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

Table 3-1 summarizes the results.

Table 3-1 Summary Vibration Analysis Results for Expected Uprate Conditions Maximum for U-Bend Region and Straight Leg Region

Location	**Fluidelastic**			**Turbulence**		**Vortex**
	Maximum Stability Ratio, (Max FSR)	**Damping (%)**	**Frequency @ Max FSR (Hz)**	**Amplitude (RMS) Peak (10^{-3} in)**	**Stress (RMS) Peak (psi)**	**Amplitude (10^{-3} in)**
U-Bend, Tube R135C90	**0.555**	0.35	115.0	(0.44) 1.5	(71) 315	*
Peripheral Tubes-Straight Leg, R34C3-TSP Clean	0.423	1.5	41.4	(0.14)0.5	(37)164	<3.86
R1C4-TSP1-Plugged	**0.681**	0.9	84.6	(0.49) 1.7	(111) 494	-

*Shedding effects are bounded by turbulence correlations in the remainder of the bundle.

NRC Question 4

In regard to Section 5.2.2, you stated that for the holddown ring evaluation, rocking and sliding margins were calculated using the revised hydraulic input loads and moments, in combination with holddown ring loads derived from recent field ring deflection measurement data. Confirm whether and how the holddown ring is acceptable to provide adequate reactor vessel internal (RVI) hold down force and provide technical basis that the margin factors of 2 and 1.5 are considered acceptable as stated in Section 5.2.2. Also, in regard to Section 5.2.2, provide an assessment of flow-induced vibration of the RVI components due to the power uprate.

ANO Response

Reactor vessel internal (RVI) hold down results and margins

The derivation of allowable rocking and sliding margins is not based on any regulatory document or requirement. The allowable margin values used by Westinghouse in its rocking and sliding evaluations represent applications of engineering judgement, based on operating experience with numerous plants. Basically, rocking or sliding margin is defined as hold down load or moment divided by applied load or moment. Any margin greater than 1.0 will prevent rocking or sliding and is therefore adequate. To account for the uncertainties present in any evaluation, Westinghouse uses an allowable margin of 2.0 for four-pump operation (the normal operating configuration). This allowable margin represents an increase of 100% over the threshold margin of 1.0, and is thus very conservative. The other operating configurations are in effect only during plant transient conditions, and thus account for a much lower percentage of the plant operating life. For these less commonly occurring configurations, a less stringent allowable margin of 1.5 is employed, which still represents an increase of 50% over the threshold margin of 1.0. In addition, the calculation of hydraulic loads used in the rocking and sliding evaluation is itself based on conservative assumptions.

Holddown ring rocking and sliding margins for ANO-2 satisfy the above criteria. These margins demonstrate that the holddown ring provides sufficient hold down force to prevent rocking and sliding of the core support barrel and upper guide structure assemblies.

Flow-induced vibration of RVI components

Flow-induced vibration is caused by the application of dynamic hydraulic loads. As applied to the RVI components, these dynamic hydraulic loads are associated with pump pulsation and random turbulence. Hydraulic load input to the RVI structural evaluation included both static and dynamic components, and the resulting RVI structural margins (summarized in Table 5-1 of the Power Uprate Licensing Report) therefore reflect the application of flow-induced vibration loads. Because of their high cyclical rate of application (related to component natural frequency), these flow-induced vibration loads also affect high-cycle (i.e., $> 10^6$ cycles) fatigue usage. At the time of the Analyses of Record (AOR), high-cycle fatigue curves were not included in the ASME B&PV Code, and high-cycle fatigue evaluations were not, and are not, required by the ANO-2 FSAR. For RSG/Power Uprate, high-cycle fatigue of RVI components was addressed via a scoping evaluation, which used the high-cycle fatigue curves from current ASME B&PV Code editions to demonstrate that high-cycle fatigue usage would be generally acceptable for the ANO-2 RVI components. Low-cycle (i.e., $< 10^6$ cycles) fatigue was addressed by demonstrating that the peak alternating stress required to achieve maximum allowable fatigue usage was greater than that calculated for any of the RVI components.

NRC Question 5

In reference to Section 5.7-1, you stated that following the application of leak-before-break (LBB), the remaining pipe breaks in the mechanical design basis of the RCS are all primary and secondary side branch line pipe breaks (BLPBs) interfacing with the RCS. Of these, the limiting breaks with respect to RCS structural considerations are breaks in the largest tributary pipes such as main steam line, feedwater line, surge line, safety injection line and shutdown cooling line. Clarify whether the thermal transient effects due to large-bore RCS pipe-break loss-of-coolant accidents (LOCAs) were considered in current licensing basis for the design of the ANO-2 RSGs. If not, explain why they were not considered (note that the approved LBB condition applies only to dynamic effects). Also, provide the stress analysis results for the primary side components of the RSGs including the RSG tubes to demonstrate the adequacy of the ANO-2 RSGs for the effects of thermal transients arising from postulated large-bore RCS pipe-break LOCAs during the power uprate.

ANO Response

Leak-before-break methodology was applied in the replacement steam generator design only for consideration of dynamic effects. The thermal effects during a loss of coolant accident (LOCA) were analyzed considering a large-bore reactor coolant system pipe-break. Since resulting thermal stresses are either secondary or peak stresses per Section III of the ASME B&PV Code, such stresses need not be considered in the Level D primary stress evaluation. In addition, since Level D events need not be included in the Section III fatigue assessment, and consistent with the requirements of Section III of the ASME B&PV Code, the thermal effects for a Level D (faulted) condition need only be considered in the Section III Appendix G nonductile fracture evaluation. A summary of the large break LOCA nonductile fracture evaluation results for the primary side components of the replacement steam generators is provided in Table 5-1. Consistent with the guidelines of Appendix G, the calculated critical flaw sizes are large (readily detectable) and thus are acceptable. It should be noted that the methods of evaluating nonductile behavior outlined in Appendix G are applicable only to ferritic materials. Non-ferrous materials such as alloy 690 used for the tubes exhibit ductile behavior even at relatively low operating temperatures, and thus brittle fracture of the tubes is not a concern.

Table 5-1 Summary of Large Break LOCA Nonductile Fracture Evaluation Results for Replacement Steam Generator Primary Side Components

Component	**Calculated Critical Flaw Size (inches)**[1]
Tubesheet (at channel head junction)	1.47
Primary Nozzle	> T / 4
Primary Manway	> T / 4

Notes:

[1]Reported flaw size is depth in the through wall direction. Consistent with Section III methods, the flaw length is six times the depth.

T = wall thickness

NRC Question 6

In reference to Section 5.7.2 [RCS Pipe Break Analysis Methodology], you stated that for the RCS with the RSGs, non-linear response time history analyses were performed to calculate the RCS response to the limiting BLPBs following the application of LBB technology. You also stated that a more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro mass and coupling terms, as well as additional nodes for RV blowdown input loadings. Confirm whether the analyses of the RV blowdown forcing functions and the non-linear structural responses due to the RSGs and the power uprate were performed by computer codes that were approved by the NRC or used in the analysis of record at ANO-2. Identify the computer codes that were used for the analyses of pipe breaks, seismic and transient events, that are different from those used in the original design basis analysis, and provide a justification that the new code was bench-marked for this application.

ANO Response

The blowdown induced forces in the reactor vessel were calculated using the CEFLASH-4B computer code in accordance with the assumptions and methods described in the NRC approved topical report CENPD-252-P-A.

The computer code ANSYS was used to compute the dynamic structural response of the ANO-2 reactor coolant system (with replacement steam generators and power uprate) to loads associated with branch line pipe breaks. This code was used for the seismic and branch line pipe break analyses of the reactor coolant system with replacement steam generators and power uprate.

ANSYS was not used in the analysis of record for the ANO-2 reactor coolant system with the original steam generators. In order to utilize the ANSYS code for these reactor coolant system analyses, the STRUDL/DYNAL model of the ANO-2 reactor coolant system with original steam generators was translated into ANSYS, and the ANSYS model was benchmarked against the STRUDL/DYNAL model. Following the benchmarking of this model in ANSYS language, the reactor coolant system model was revised to include the replacement steam generators and other necessary details for performing seismic, secondary side and primary side branch line pipe break analyses.

ANSYS is a detailed finite element analysis computer code and is a standard industry code used for these types of applications. ANSYS, as a code in the public domain, is not described in a topical report which the NRC reviews and approves via a safety evaluation report. It is, however, controlled under a Quality Assurance program and is benchmarked and validated for changes that occur in the program. The code version utilized is maintained by Swanson Analysis Systems, Inc.

NRC Question 7

In reference to Section 5.7.2, you indicated that for the pipe break analysis of the RCS with RSGs, two three-dimensional ANSYS models of the entire RCS were developed from the RCS seismic model, one for secondary side breaks and one for primary side breaks. For the secondary side pipe break model, the representation of the RVI remained essentially the same as that for the seismic model, because secondary side breaks do not cause RV blowdown. A more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro-mass and coupling terms, as well as additional nodes for RV blowdown input loadings. The response of the entire RCS to pipe breaks was calculated using non-linear response time history analysis. The ANSYS computer code was used to perform the time history analyses due to BLPBs, using the modal superposition method and constant 3% modal damping. Clarify whether the ANSYS computer code was used to perform the non-linear time history analysis, using the modal superposition method. Describe the nonlinear parameters used in analysis. Also, provide a summary of analysis with a detailed model of the reactor internals to account for the depressurization blowdown loading in the BLPB analysis.

ANO Response

The ANSYS computer code was used to perform the non-linear time history analyses due to branch line piping breaks using the modal superposition method, using a constant 3% modal damping.

The results of the detailed reactor vessel internals (RVI) loads analysis are given in Section 5.2 of the Power Uprate Licensing Report.

To provide depressurization blowdown loadings for the RCS branch line pipe break (BLPB) analysis, a simpler RVI model is built. The basis for this simpler model is the detailed RVI model that is later used in the RVI loads analysis. The simplified RVI model contains fewer nodes and fewer dynamic degrees of freedom than the detailed RVI model, but it maintains the total mass, center of gravity, structural properties, natural frequencies, and hydrodynamic mass couplings of the detailed model. Since the simpler model has fewer loading points for the blowdown loads than the detailed model, some of the blowdown loadings are combined for the RCS analysis. This RVI model and sets of blowdown load time histories are provided for the RCS primary side BLPB analyses.

This methodology is the same as previously used for RCS asymmetric loads analysis for main coolant loop breaks.

NRC Question 8

In reference to Section 5.8, provide, for the most critical RCS pipe systems evaluated, the calculated maximum stresses and fatigue usage factor, and code allowable limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification. Were the analytical computer codes used in the stress analysis different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

ANO Response

Section 5.8, "RCS Tributary Line Reconciliation Analysis" of the Power Uprate Licensing Report discussed the evaluation of changes resulting from the RSGs at power uprate conditions and reconciliation of the resultant loads against applicable code allowables. This discussion included safety injection, shutdown cooling, pressurizer spray, main steam, and main feedwater lines. The steam generator replacement necessitated the reconciliation analyses because the RSGs weigh more than the OSGs. When these analyses were performed for the RSGs, power uprate conditions were conservatively included. The impact of power uprate itself on the analyses was insignificant. In anticipation of a license extension for ANO-2, the piping was qualified for 60 years (CUF are calculated based on 60-year life).

Analytical Computer Codes

As discussed in Section 5.8.2.2 of the PULR, the ME101 pipe stress analysis program was used to create mathematical models of the designated lines. Any piping stress reanalysis performed for the piping due to power uprate changes was performed using the Bechtel

ME101 analysis software, which is a later version of the same analytical computer code (ME632) used for the original design basis analysis.

The ME101 program is an industry standard program that has been used on every facility designed by Bechtel. It performs the piping analysis in accordance with the ASME B&PV Section III or B31.1 Code formulae and rules. The Bechtel suite of programs containing ME101 is controlled under a Quality Assurance program that has been benchmarked and validated for changes that have occurred in the program since the analysis code version used during the initial design of ANO.

Code of Record

Main Steam and Main Feedwater

For the main steam and main feedwater lines inside containment, the Code and Code edition used in the analysis was the Code of record, which is ASME Boiler and Pressure Vessel (B&PV) Code Section III, NC/ND (Class 2/3), 1971 Edition through Summer 1971 Addenda.

Safety Injection, Shutdown Cooling, and Pressurizer Spray

Analyses for Class 1 piping for safety injection, shutdown cooling, and pressurizer spray were made per the 1980 ASME B&PV Code, Section III, NB-3600. Since the Code of record is the 1971 ASME B&PV Code through the Summer 1972 Addenda, a reconciliation was required. The following reconciliation applies to the stress analysis on the piping only.

The changes made to indices and stress equations from the Code of record to the 1980 Code are consistent with better understanding of piping stress. This understanding is derived from test and detailed finite element analysis. Using a more recent Code edition is not a problem since the analytical methods are not connected to evolving fabrication practices. The indices are established for standard piping components and weld types and changes in the specification of the components or welds are only allowed if it can be shown the indices are unaffected. Since the more recent Code edition has more joint types (definition of geometry for group or type of joint) than the older Code edition, some review is required to determine into which joint type in the more recent Code edition a joint fits. Since mixing codes is not recommended, all the piping reanalysis is done in the 1980 Code.

Allowables from the Code of record were used because the materials were tested and certified to meet the Code of record. 1980 Code allowables were permissible if the yield and ultimate are equal to those of the 1971 Code through Summer 1972 Addenda.

Generally, the ASME concurs with the use of more recent Code editions for stress analysis as long as the analyst is consistent and logical (ASME B&PV Section III 1986-NCA 1140). Therefore, the reanalysis was performed using the 1980 Code edition and the joint types were classified per the geometry limits in the 1980 Code. Allowables were taken either from the

Code of record or from the 1980 Code edition where yield and ultimate equal the Code of record.

Calculated Maximum Stresses, Fatigue Usage Factors, Code Allowable Limits

This information is contained in the following tables. The bracketed numbers in the location descriptions refer to nodes used in the ME101 analysis.

As main steam and main feedwater are ASME B&PV Section III Class 2 piping, no fatigue usage factors are required for this piping.

Table 8-1
Safety Injection Line to B RCP
Class 1 Stress Summary Piping (2CCA-21 and 2CCA-51)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to 1" vent line between core flood tank check valve and isolation valve [96]	21,441	27,450	0.78
Faulted <3.0 Sm (Level D)	Elbow nearest cold leg [15]	41,680	54,900	0.76
Primary plus Secondary Ranges (Equation 10)	Elbow nearest shield wall penetration [70]	57,701	54,900	1.05
Primary plus Secondary Ranges (Equation 12)	Elbow nearest shield wall penetration [70]	35,457	54,900	0.65
Primary plus Secondary Ranges (Equation 13)	Branch from 8" to 12" safety injection line [45]	27,422	54,900	0.50
Cumulative Usage Factor	Branch from 8" to 12" safety injection line [45]	NA	1.0	0.22

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-2
Safety Injection Line to A RCP
Class 1 Stress Summary Piping (2CCA-22 and 2CCA-52)

Description	**Location of Max. Stress**	**Maximum Stress (psi)**	**Allowable (psi)**	**Ratio**
Design < 1.5 Sm (Level A)	Connection to pressure transmitter [95]	15,888	27,450	0.58
Faulted <3.0 Sm (Level D)	Elbow nearest cold leg [10]	26,430	54,900	0.48
Primary plus Secondary Ranges (Equation 10)	Change from 6" to 8" line [240]	45,108	54,900	0.82
Primary plus Secondary Ranges (Equation 12)	Not required since Equation 10 is met.			
Primary plus Secondary Ranges (Equation 13)				
Cumulative Usage Factor	Connection to 3" HPSI header [235]	NA	1.0	0.13

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-3
Safety Injection Line to D RCP
Class 1 Stress Summary Piping (2CCA-23 and 2CCA-53)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to 3" HPSI header [400]	21,046	27,450	0.77
Faulted <3.0 Sm (Level D)	Branch from 8" to 12" safety injection line [125]	31,870	54,900	0.58
Primary plus Secondary Ranges (Equation 10)	Connection to 3" HPSI header [400]	58,746	54,900	1.07
Primary plus Secondary Ranges (Equation 12)	U-bend between shield wall open penetration and connection to 8" line[100]	34,780	54,900	0.63
Primary plus Secondary Ranges (Equation 13)	Connection to 3" HPSI header [400]	34,205	54,900	0.62
Cumulative Usage Factor	Branch from 8" to 12" safety injection line [125]	NA	1.0	0.18

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-4
Safety Injection Line to C RCP
Class 1 Stress Summary (Piping 2CCA-24 and 2CCA-54)

Description	**Location of Max. Stress**	**Maximum Stress (psi)**	**Allowable (psi)**	**Ratio**
Design < 1.5 Sm (Level A)	Connection to pressure transmitter [201]	17,036	27,450	0.62
Faulted <3.0 Sm (Level D)	Connection to pressure transmitter [201]	23,910	54,900	0.44
Primary plus Secondary Ranges (Equation 10)	Branch from 8" to 12" safety injection line [35]	53,334	54,900	0.97
Primary plus Secondary Ranges (Equation 12)	Not required since Equation 10 is met.			
Primary plus Secondary Ranges (Equation 13)				
Cumulative Usage Factor	Branch from 8" to 12" safety injection line [35]	NA	1.0	0.20

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-5
Shutdown Cooling Line
Class 1 Stress Summary Piping (2CCA-25 and 2CCA-57)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	Connection to pressure point between the two inside containment isolation valves [416]	23,042	27,450	0.84
Faulted <3.0 Sm (Level D)	Connection to pressure point between the two inside containment isolation valves [416]	29,320	54,900	0.53
Primary plus Secondary Ranges (Equation 10)	Elbow in line to pressure relief valve 2PSV 5085 [180]	58,508	54,900	1.07
Primary plus Secondary Ranges (Equation 12)	Elbow in line to pressure relief valve 2PSV 5085 [180]	46,144	54,900	0.84
Primary plus Secondary Ranges (Equation 13)	Connection to 3" line from HPSI header [12]	19,696	54,900	0.36
Cumulative Usage Factor	Connection to 3" line from HPSI header [12]	NA	1.0	0.64

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

Table 8-6
Pressurizer Spray Line
Class 1 Stress Summary Piping (2CCA-13, -14, -15, -16)

Description	Location of Max. Stress (see table)	Maximum Stress (psi)	Allowable (psi)	Ratio
Design < 1.5 Sm (Level A)	[231]	13,920	27,450	0.51
Faulted <3.0 Sm (Level D)	[86]	36,650	54,900	0.67
Primary plus Secondary Ranges (Equation 10)	[108] [V1] [400] [10]	108,710 69,220 67,790 60,920	59,500 54,900 54,900 54,900	1.83 1.26 1.23 1.11
Thermal Expansion Moments (Equation 12)	[108] [V1] [400] [10]	24,310 6,800 43,910 48,840	59,500 54,900 54,900 54,900	0.41 0.12 0.80 0.89
Primary plus Secondary, without the Thermal Expansion Moments (Equation 13)	[108] [V1] [400] [10]	54,500 42,240 22,480 13,160	59,500 54,900 54,900 54,900	0.92 0.77 0.41 0.24
Cumulative Usage Factor	[V1],[V1A](1) [108] [140] [V20] [128] [106] [4]	NA		0.92 0.99(2) 0.96 0.89 0.88 0.86 0.82

Node	Description	Node	Description	Node	Description
[4]	Connection to main spray vent line	[108]	Elbow in auxiliary spray line	[400]	Connection to 1" line to reactor drain tank
[10]	Elbow upstream of main spray vent line	[128]	Bend in auxiliary spray line	V1	Connection on main spray vent line
[86]	Connection to 1" line to reactor drain tank	[140]	Elbow in auxiliary spray line	V1A	Elbow in main spray vent line
[106]	Elbow in auxiliary spray line	[231]	Reducer upstream of main spray valve 2CV-4656	V20	Reducer in auxiliary spray line to pressure point

(1) Node points V1 and V1A, are socket-welded connections on the ¾" Schedule 160 main Spray vent line which have been shown have acceptable CUF of 0.92. In addition, these locations have been shown to have acceptable stress levels when considered as Class 2 piping.

(2) Mostly due to conservative thermal stratification reanalysis.

Table 8-7
Main Steam Piping Inside Containment
(Piping 2EBB-1 and 2EBB-2)

Description	**Location of Max. Stress**	**Maximum Stress (psi)**	**Allowable (psi)**	**% of Allow.**
A Steam Generator				
Deadweight	Elbow at 47° H run [135B]	8,938	17,500	51.1
Maximum of Deadweight + OBE Seismic or Deadweight + Dynamic Steam Hammer Time History	Penetration [180]	18,973	21,000	90.3
Deadweight + (DBE Seismic + LOCA)[1]	Penetration [180]	24,265	42,000	57.8
Thermal Expansion + SAM-OBE Building Displacements	Elbow before penetration [175M]	13,592	26,250	51.8
B Steam Generator				
Deadweight	Elbow on 47° H run [135B]	9,115	17,500	52.1
Maximum of Deadweight + OBE Seismic or Deadweight + Dynamic Steam Hammer Time History	Containment penetration [170]	11,619	21,000	55.3
Deadweight + (DBE Seismic + LOCA)[1]	Steam generator connection [10]	20,835	42,000	49.6
Thermal Expansion + SAM-OBE Building Displacements	Containment penetration [170]	14,921	26,250	56.8

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

(1) DBE Seismic and LOCA are combined by square root sum of the squares (SRSS)

Table 8-8
Main Feedwater/Emergency Feedwater Piping Inside Containment
(Piping 2-DBB-1 and 2-DBB-2)

Description	Location of Max. Stress	Maximum Stress (psi)	Allowable (psi)	% of Allow.
A Steam Generator				
Deadweight	Support upstream of check valve 2FW-5A [58]	7,578	17,500	43.3
Deadweight + OBE Seismic	Connection to pressure point line downstream of check valve 2EFW-9A [900]	12,961	18,000	72.0
Deadweight + (DBE Seismic + LOCA)[1]	Connection to steam generator [5]	26,414	42,000	62.9
Thermal Expansion + SAM-OBE Building Displacements	Connection to steam generator [5]	19,496	26,250	74.3
B Steam Generator				
Deadweight	Support downstream of check valve 2FW-5B [48]	7,459	17,500	42.6
Deadweight + OBE Seismic	Connection to 4" EFW line [32]	14,723	21,000	70.1
Deadweight + (DBE Seismic + LOCA)[1]	Shield wall penetration [5]	26,500	42,000	63.1
Thermal Expansion + SAM-OBE Building Displacements	Shield wall penetration [5]	19,514	26,250	74.3

(Note: Bracketed numbers refer to nodes in the ME101 analysis)

(1) DBE Seismic and LOCA are combined by square root sum of the squares (SRSS)

NRC Question 9

In reference to Section 2, you stated that the balance-of-plant (BOP) structures, systems and components have been evaluated for the impact of the 7.5 percent power uprate and in general found acceptable. Those requiring modifications due to power uprate consideration are provided in Table 2-2. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

ANO Response

BOP piping, pipe supports, nozzles, penetrations, guides, and anchorage

The methodology used for evaluating BOP piping for the effects of power uprate was either the standard ANO engineering process for modifying systems or the ANO engineering methodology to evaluate systems not requiring physical modifications. If a piping configuration is modified to support the power uprate, the modified piping configuration is analyzed and qualified for the appropriate loading conditions as part of the normal modification process. The modification process qualification considered the post-power uprate parameters such as pressure and temperature and qualifies the piping stress, pipe supports, nozzles, penetrations, and anchors for the new configuration and loading.

There are very few modifications to piping as a result of power uprate, so the ANO engineering evaluation process was used for systems not being modified.

Piping systems that do not require modifications resulting from power uprate can still be affected by the uprate. Therefore, the following process was used to evaluate the impact of uprate on those systems. The primary design input parameters for piping analysis are the piping configuration and the loading. Since the physical configuration is not changing, the primary effect that power uprate might have on piping systems is driven by changes that affect pipe loads. The parameters that could affect loading if changed by power uprate are primarily pressure and temperature. For ANO-2, pressure and temperature parameters are determined and documented in “Pressure and Temperature” (PT) calculations. These PT calculations specify the maximum pressure and temperature values for each line class for the applicable plant operating modes (normal, upset, emergency, and faulted). Based on those maximum bounding design values, a temperature and pressure for each line class is established in the PT calculation. PT calculations affected by power uprate were revised to reflect the resulting changes. Revised PT values were evaluated for effect on piping stress, pipe support, nozzle and anchor qualification. The qualification of the piping systems for the revised PT values is documented in ANO calculations. In many of the systems, the changes in maximum pressure and temperature values due to power uprate were already bounded by design pressures and temperatures which therefore remain in effect. For those systems that did have a change that affected the design analysis of the piping system, the qualification of record for that piping was reviewed to identify the maximum stress and load values. Typically, a scaling factor was used to increase the stress or load calculated in the qualification of record by the ratio of the parameter increase. The new stress or load was then compared and documented to be within allowable limits. Typically, the increase in the input parameters was only a few psi or degrees, and the limits were not significantly challenged. The effects of the changes were specifically evaluated for the following aspects:

- thermal expansion stress
- pipe support loads
- nozzle qualifications
- flange and pipe fitting qualification
- pressure design (hoop stress) of piping systems
- creation of new high energy piping systems for HELB/MELB effects
- creation of new missile hazards from pressurized piping systems
- Flow Accelerated Corrosion acceptance criteria for minimum wall thickness limits
- piping thermal movement limits in fire barrier penetrations
- past flaw evaluations
- expansion joints
- dynamic loading due to fast valve closure transients.

The piping stress analysis does not use flow rate as a direct loading input. Changes in flow rate were considered from a structural perspective only with regard to the effect that changes in the flow rates would have on flow-induced vibration of the piping or on the dynamic loading due to fast valve closure transients. The response to Question 11 discusses the impact on flow-induced vibration.

Because the pressure and the mass flow rate will increase with power uprate, the main turbine stop valve fast closure transient analysis was updated, creating new dynamic forcing functions for the main steam headers and all hydraulically attached branch piping greater than four (4) inches. Bechtel evaluated the new forcing functions against the original dynamic analysis. The original analysis is bounding for the new forcing functions with the exception of the branch lines for the main steam supply to the main feed water pump driver turbines and the main steam supply to the second stage moisture separator-reheater tube bundles. A reanalysis of those four lines is being performed to qualify the piping and supports for the revised dynamic loads.

Piping stress reanalyses performed for the BOP piping due to power uprate changes were performed using the Bechtel ME101 analysis software, which is a later version of the same analytical computer code (ME632) used for the original analysis. See the response to NRC Question 8 for additional information regarding the Bechtel ME101 program.

Other BOP Components

PT calculations for power uprate were also used to evaluate other mechanical components such as manual valves, motor-operated valves, air-operated valves, solenoid-operated valves, bleeder trip valves, check valves, relief valves, tanks, heat exchangers, and pumps. Evaluations began by identifying changes in the PT calculations caused by the RSGs and power uprate. Line classes identified as having increased values were reviewed against the applicable piping and instrument drawings containing that line class. Mechanical components and valves within these line classes were then identified as requiring further evaluation due to increases in either design or maximum pressure and/or temperatures. The line classes identified were also used to evaluate impacts on insulation and room heat loads.

Components identified as requiring further evaluation were reviewed against the design conditions for the components. This included a comparison with such items as Code ratings, manufacturer's information, material used, etc. These were very detailed reviews which ensure that all components are acceptable for operation under power uprate conditions. Additionally, feedwater heaters were evaluated by the vendors for thermodynamic performance under power uprate conditions.

The results of these reviews were presented in Section 2 of the PULR, particularly in Section 2.4.2, "Main Steam Supply System," and Section 2.4.5, "Condensate and Feedwater."

Various computer codes were used to assist in these analyses. Bechtel FLASH TE605 was used for control valve sizing. PIPEFLOW was used for modeling the condensate and feedwater system, including the heater drain portion. The ASME B&PV Code, Section VIII, 1992 edition was used for relief valve sizing. The Heat Exchange Institute standard for closed feedwater heaters and a standard Napier formula for orifices were used to analyze feedwater vents to the condenser.

NRC Question 10

Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

ANO Response

Critical BOP piping systems consist of the main steam (MS) and main feedwater (MFW) headers both inside and outside the containment building.

Stresses for main steam and feedwater piping inside containment are described in the response to Question 8.

The main steam and main feedwater header piping located outside of the containment building remains qualified for the power uprate parameters because the changes in the pressure and temperature due to power uprate are bounded by existing analyses of record. This conclusion was documented in the piping reconciliation calculations for the main steam and main feedwater PT calculations. A review of the pressure and temperature values in the PT calculations for these line classes confirmed that there were no changes in applicable values as a result of power uprate. Because the analyses for the main steam and main feedwater header piping outside containment were not revised, calculated stresses, allowable limits, and the analysis code and code edition remain unchanged.

NRC Question 11

In reference to Section 2.4.5.3, you stated that the feedwater heaters have been evaluated for the power uprate condition for extractions, design pressures, pressure drops, and drain, tube and nozzle velocities. You also stated that feedwater heater vibration characteristics and shell-side relief valve capacities have been evaluated. The main steam and feedwater flow rates increase about 10 percent for the power uprate as shown in Table 3-1. Discuss the potential for flow-induced vibration in the main steam and feedwater pipe and the BOP heaters and heat exchangers following the power uprate.

ANO Response

Based on studies discussed in Section 2.4.5 of Enclosure 5 to the Power Uprate Licensing Report, the original condensate, main feed water (MFW), extraction and drain system piping is generously sized, and will have new flow velocities that are well within acceptable and recommended ranges. Because of this, it was concluded that the MFW header piping is not expected to experience unacceptable flow-induced vibration as a result of changes from power uprate. The feedwater heaters were reviewed by the vendor and found to be acceptable. The feedwater trains are sized to carry a substantial load with a single train. This generous sizing renders the feedwater heaters less sensitive to flow-induced vibration. Any flow induced vibration problems caused by power uprate are expected to be confined to small vents and drains.

Historically, the main steam piping has been the system that has displayed the most sensitivity to flow-induced vibration. Because of this history ANO, along with a second party review by Southwest Research Associates, studied the potential changes in the main steam piping vibration due to changes in the pressure and mass flow rate for original design, (Cycle 14 - the last cycle with the OSGs), and the first cycle for power uprate (Cycle 16). This study evaluated the physical geometry of the piping, vibration data collected on the main steam piping, and the effects that pressure, temperature, and mass flow rate changes would have on the kinetic energy available to drive flow-induced vibration of the piping. A summary of the comparison between the original (Cycle 14) and post-power uprate conditions is provided in Table 11-1 below. From this table, it can be seen that although the mass flow rate is increasing, steam velocity and kinetic energy levels will be less after power uprate than during Cycle 14. Based on this and the measured vibration levels in the piping during Cycle 14, it is not anticipated that power uprate operating conditions will cause unacceptable increases in the vibration of the major main steam piping. Again, for the main steam system the only significant vibration problems are expected to be associated with small vent and drain piping. Modifications have already been installed to reduce the vibration levels on the level controllers on the 2E-1 (high pressure) feedwater heaters, and, as other unacceptable vibration levels are identified, appropriate measures will be taken.

Table 11-1

	Mass Flow Rate (lbm/hr)	**Pressure (psia)**	**Temp (°F)**	**Relative Flow Velocity**	**Relative Kinetic Energy**
Original Conditions	12697495	878.2	529.0	1.00 X Original	1.00 X Original
Cycle 14 Conditions	12720000	769.0	513.7	1.18 X Original	1.18 X Original
Power Uprate Conditions	13660920	900.0	532.0	1.05 X Original; 0.90 X Cycle 14	1.13 X Original; 0.96 X Cycle 14

However, it was recognized that there may be secondary effects that may not be predicted; therefore, ANO has implemented a piping walkdown and vibration testing program to identify and resolve any such problems. The pre-2R14 and post-2R14 walkdowns and testing identified the equipment and systems with potential vibration concerns. The start-up testing program included the installation of vibration monitoring instrumentation on the main steam piping inside containment, hand-held collection of vibration data on main steam piping outside containment, and structured walkdowns of virtually all of the piping outside containment for visual identification of piping vibration. For piping identified by visual walkdowns, vibration data collection was performed. As recommended by ASME/ANSI OM-3, "Operation and Maintenance Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems," any vibration exceeding the 0.5 inch per second screening criterion was evaluated by the Design Engineering Structural Group. This same testing approach is planned again for start-up after power uprate.

NRC Question 12

Discuss the functionality of safety-related mechanical components (i.e., all safety related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at ANO-2 will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the uprated power level was not evaluated. Also, discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

ANO Response

For the safety-related pumps, minimum performance requirements did not increase due to power uprate. Therefore, power uprate has no effect on the functionality of the pumps. The safety analyses confirmed that these minimum requirements are sufficient for the pumps to perform their intended function. (Section 7 of the Power Uprate Licensing Report describes the safety analyses for power uprate.) Power uprate does not affect the ability of the pumps to meet their technical specification requirements. As stated in Section 2.4.6 of the Power Uprate Licensing Report, the emergency feedwater pumps are adequate for power uprate. Although decay heat will increase, engineering evaluations for power uprate determined that no change to the EFW pump flow rate is needed. Calculations demonstrate that the EFW pumps can provide the minimum flow rate necessary to support the safety analysis flow rate assumptions. Similarly, no changes are necessary for the high pressure injection pumps or the low pressure injection pumps.

As discussed in Section 5.9 of the PULR, the specific overpressure protection requirements of the ASME B&PV Code were evaluated for power uprate. All general requirements and component requirements for pressurizer safety valves and main steam safety valves were found to be in compliance with the code and the original design requirements. As discussed in Enclosure 4, Section 1.0.1 of our letter dated November 29, 1999 (2CAN119901), the pressurizer code safety valves' (PSVs) capacity ratings were revised based on the use of the Napier Factor which was adopted by later versions of the ASME B&PV Code. The revision supported rerating the PSVs. Analysis of bounding reactor and steam plant transients causing pressure excursions have been conducted. These transients were evaluated to ensure both peak primary and secondary pressure did not exceed 110% of design pressure. ANO-2 has no power-operated relief valves.

The safety-related motor-operated valves (MOVs) and air-operated valves (AOVs) were evaluated for the pressures and temperatures expected for power uprate. For systems not affected by power uprate (no pressure or temperature increase, no increase in differential pressure), the valves were considered acceptable without further evaluation.

The safety-related AOVs were evaluated and found to be acceptable for power uprate conditions. AOVs were evaluated for the proper pressure/temperature rating for the expected conditions. This included an evaluation of the air actuator for the same conditions as well as for the expected differential pressure which will be experienced by the valve under power uprate conditions. Of the safety-related AOVs for ANO-2, only the main steam isolation valves are in a system or application impacted by power uprate. These were determined to be capable of performing their intended function under power uprate conditions. Those AOVs performing a containment isolation function had been previously evaluated for the containment upgrade, which included power uprate conditions, and found acceptable.

The safety-related MOVs were determined to be acceptable for power uprate conditions. The MOVs, including the GL 89-10 MOVs, were evaluated for the pressures and temperatures expected for power uprate. The pressures and temperatures were reviewed with respect to each MOV's design function to stroke. As part of this review, applicable setpoint, maximum expected differential pressure (MEDP), seismic, and weak link calculations were examined for potential impact from power uprate. Based on this evaluation, no physical changes to MOVs are required for power uprate conditions.

In regard to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," the required relief valves have already been installed. These relief valves have been evaluated for power uprate conditions and found to be acceptable.

Evaluations for Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," are scheduled to be completed by September 30, 2001. No significant impacts due to power uprate have been identified to date and none are expected.

NRC Question 13

Confirm whether the steam generator replacement and the proposed power uprate will increase the accident temperature, pressure and sub-compartment pressurization that affect the design basis analyses for steel and concrete in the containment, steam tunnel and the spent fuel pool. If the structural steel and concrete will be affected, provide the design basis margin and margins after considering increased accident loading due to the steam generator replacement/power uprate.

ANO Response

Before replacing the steam generators, ANO-2 evaluated the effect on containment of the replacement steam generators and a 7.5% power uprate. The revised loss of coolant accident (LOCA) and main steam line break (MSLB) analyses necessitated an increase to the containment design pressure to 59 psig. This was documented in correspondence dated November 3, 1999 (2CAN119903), "Proposed Technical Specification Change Request Supporting Containment Building Design Pressure Increase to 59 Psig." The effect of the replacement steam generators (RSGs) and power uprate on the accident temperature, pressure and compartment pressurization was included in that submittal. Enclosure 3 of the November 3, 1999, letter describes the LOCA and MSLB analyses which included the 7.5% power uprate. Enclosure 4 describes the structural reanalysis performed for 59 psig including the design basis analysis for the steel and concrete in the building. Enclosure 5 summarizes the review of structures, systems and components inside containment. This review included the compartment pressurization.

Section 9.2 of the Power Uprate Licensing Report discusses high-energy line break analyses evaluated for RSG and power uprate. Changes to these analyses have been incorporated into Amendment 16 of the ANO-2 Safety Analysis Report (SAR). SAR Section 3.6 discusses the ANO-2 main steam tunnel. Section 3.6.4.1.1.2 explains that only one break location inside the steam tunnel is postulated. This postulated break was reanalyzed for the environmental effects for a power level of 3026 MWt with credit for the flow limiting device located in each steam generator outlet nozzle. The peak pressure remains bounded by the previous evaluation. The peak temperatures were increased due to higher steam enthalpy conservatively predicted from superheating as steam passes over uncovered tubes. A new peak temperature of 424 °F near the end of blowdown at 190 seconds was calculated. However, the reinforced concrete wall of concern that separates the turbine building from the auxiliary building is a 3-hour fire rated barrier. A 3-hour fire rated barrier is designed to withstand temperatures well in excess of that postulated from a high-energy line break.

Cooling for the spent fuel pool was discussed in detail in our letter dated May 30, 2001 (2CAN050105). Pool temperatures will be maintained as they are currently. The cooling system is adequate for power uprate conditions. If spent fuel pool cooling is lost, the pool is allowed to boil and makeup is provided by the service water system; therefore, power uprate causes no increase in pool temperature under a loss of cooling condition. Since pool temperatures will not increase for normal operation or loss of cooling conditions, power uprate does not affect the design basis analysis for the steel and concrete in the pool.