Entergy

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November 6 , 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket 50-293
License No. DPR-35

Response to NRC Request for Additional Information
Appendix K Measurement Uncertainty Recovery – Power Uprate Request

REFERENCE:

1. NRC Fax, "Request for Additional Information," dated September 20, 2002
2. Entergy Letter 2.02.048, dated July 5, 2002, Appendix K Measurement Uncertainty Recovery – Power Uprate Request
3. Entergy Letter 2.02.080, dated August 29, 2002, Appendix K Measurement Uncertainty Recovery – Power Uprate Request Submittal of Non-Proprietary Version of TSAR
4. Entergy Letter 2.02.087, dated September 27, 2002, "Response to NRC Request for Additional Information, Appendix K Measurement Uncertainty Recovery – Power Uprate Request

LETTER NUMBER: 2.02.096

Dear Sir or Madam:

Entergy has reviewed the subject NRC request for additional information (RAI) dated September 20, 2002 and the requested information is enclosed. Attachments 1,2, 3, and 4 contain the responses to the questions. Attachment 5 of this document contains General Electric proprietary report NEDC-33050P, Revision 1. An affidavit signed by an authorized representative of GE is provided in the front of the document, pursuant to 10 CFR 2.790. It is requested that this proprietary information be withheld from public disclosure. Revision 1 replaces Revision 0 which was submitted as part of Entergy Letter 2.02.048, dated July 5, 2002. The non-proprietary version, NEDO-33050, Revision 1, is submitted as Attachment 6.

Should you have any question or comments concerning this submittal, please contact Bryan Ford at (508) 830-8403.

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I declare under penalty of perjury that the foregoing is true and correct. Executed on the __6__th day of November 2002.

Sincerely,

C.M. Dugger

JRH/dd

Attachments:
1. Response to NRC Request for Additional Information (10 pages)
2. Response to RAI 9A (91 pages)
3. Response to RAI 9B (15 pages)
4. Response to RAI 9C (9 pages)
5. General Electric Proprietary Document NEDC-33050P, Revision 1 (83 pages)
6. General Electric Non-proprietary Document NEDO-33050, Revision 1 (79 pages)

cc:

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<u>ATTACHMENT 1</u>

LETTER NUMBER 2.02.096

Response to NRC Request for Additional Information

Appendix K Measurement Uncertainty Recovery-Power Uprate Request

Request for Additional Information Responses

1. NRC Request: Since the effects of flow-accelerated corrosion (FAC) on degradation of carbon steel components are plant specific, the values of the parameters affecting FAC; i.e. velocity and temperature changes, must be included. In addition, the corresponding changes in components wear rates due to FAC before and after the power uprate must be provided.

   A. Please provide the name and version of the predictive code used to project the need for maintenance/replacement of balance-of-plant (BOP) piping components prior to reaching minimum wall thickness requirements.

   B. Please provide the predicted change of wear rates calculated by the revised predictive code for the components most susceptible to flow-accelerated corrosion. Specifically, provide a detailed table with this information as illustrated below in a sample table.

System	Description	% Change in Predicted Wear Rate		Change in Predicted Wear Rate, mils/year	
		Average	Max	Average	Max
FW	Feedwater (FW) to FW Pump to High Pressure FW Heater	+0.46%	+0.19%	+0.008	+0.02

Response:

1-A The predictive code used at PNPS is "Flow Acceleration Corrosion (FAC) Version 1.0F (Build 52)".

1-B The chart below is provided as requested.

System	Description	% Change in Predicted Wear Rate		Change in Predicted Wear Rate, mils/year	
		Average	Max	Average	Max
Extraction Steam	EX STM: 1st STAGE TO E-105	0.23	0.47	0.094	0.506
Extraction Steam	EX STM: 8th STAGE TO E-104	6.45	8.00	0.727	2.046
Extraction Steam	EX STM: 9th STAGE TO E-103	0.005	0.005	0.12	0.12
Extraction Steam	EX STM: 13th STAGE TO E-101	20.77	21.166	0.709	0.935
Heater Drains	E-105 Drains	0.53	1.72	0.0002	0.002
Heater Drains	E-104 Drains	4.690	5.520	0.008	0.020

System	Description	% Change in Predicted Wear Rate		Change in Predicted Wear Rate, mils/year	
		Average	Max	Average	Max
Heater Drains	E-103 Drains	4.893	5.137	0.080	0.255
Heater Drains	E-102 Drains	4.872	5.700	0.172	0.478
Feedwater	FW Pump P-103 to E-104	3.231	3.337	1.0415	2.649

Notes: The FAC comparison was done for the same time period using Valves Wide Open (VWO) data from the current heat balance and the Appendix K VWO heat balance. Only results where the predicted wear rate increased are shown in the above table. Some lines were not included in the evaluation because the piping was of a non-susceptible material (1 ¼% Cr ½% Mo).

Details of the analyses are maintained at PNPS.

2. NRC Request: Attachment 2 to the amendment request (TSAR), page S-1, second paragraph from the top, states,

"This report follows the Nuclear Regulatory Commission (NRC)-approved format and content for Boiling Water Reactor (BWR) Thermal Power Optimization (TPO) licensing reports documented in NEDC-32398 P, "Generic Guidelines and Evaluations for Boiling Water Reactor Thermal Power Optimization," called "TLTR."

The above referenced document, NEDC-32938, has not been approved by the staff. Therefore, the above reference statement and all related references to NRC approval of this document should be removed from the submittal.

Response:

Revision 1 of the Pilgrim TPO Safety Analysis Report (TSAR), NEDC-33050P, has eliminated references to the TLTR as approved. Specifically, the Executive Summary and Section 1.1, Overview were revised to delete " Nuclear Regulatory Commission (NRC)-approved" from the sentences. Revision 1 of the TSAR is included as Attachment 5.

3. NRC Request: Please provide information relating to contingencies for an inoperable Crossflow UFM and effect of inoperable Crossflow UFM on thermal power measurement and plant operation. In this regard, also provide the following information:

A. A proposed allowed outage time (AOT) for the feedwater flow instrument, along with the technical basis for the time selected.

B. Proposed actions to reduce power level if the AOT is exceeded, including a discussion of the technical basis for the proposed reduction in the power level.

Response:

During the past 4 years Pilgrim found the AMAG UFM to be extremely reliable, with no periods where the instrument was inoperable. However, as a contingency Pilgrim is installing two (2) new redundant Crossflow UFMs on its feedwater piping. Both units are independent and operating at all times. Each is capable of accurately measuring flow and performing the required correction factor calculations. In the event one of the two AMAG units fail, there is an alarm function that prompts the operator to switch control to the standby UFM. This switchover is controlled by procedure, (and also allows the operator to take one UFM out-of-service (OOS) for maintenance). With one unit out-of-service, and the other in operation, there is no change in plant operation.

In the unlikely event that both of the independent AMAG Crossflow UFMs are out-of-service at the same time, Pilgrim would continue to operate with the last good correction factor applied until the allowed out-of-service time (AOT) for both UFMs is reached. Once the AOT is reached, PNPS will procedurally limit power to an alternate value that accounts for the uncertainty associated with the instrumentation being used to measure power at that time.

A. Pilgrim proposes to use an AOT of 14 days. This is based on a review of correction factor drift over the past year. This review demonstrated a standard deviation of less than 0.09%.

B. In the event the AOT is reached, PNPS will be required, by procedure, to reduce its power level to an alternate value that accounts for the uncertainty associated with the instrumentation then being used to measure power. This value has not yet been determined. With both AMAG UFMs OOS, the feedwater flow nozzles will be used to measure feedwater flows without the benefit of the UFM. The feedwater flow nozzles are presently being recalibrated based on in-situ data developed using a UFM. The uncertainty associated with this calibration will be used along with the accuracy of other power measurement instruments used in the development of the alternate power level.

4. <u>NRC Request:</u> For all instruments that affect the power calorimetric, provide information to specifically address the following aspects of the calibration and maintenance procedures. The amendment request provides the required information only for the Crossflow UFM. Please provide it for the remaining instruments.

a. Maintaining Calibration
b. Controlling software and hardware configuration
c. Performing corrective actions
d. Reporting deficiencies to the manufacturer
e. Receiving and addressing manufacturer deficiency reports

Response:

a. Pilgrim Procedure 1.8, "Master Surveillance Tracking Program" provides a mechanism for maintaining the calibration of all plant instrumentation. A list of all instruments affecting the power calorimetric is shown below.

<u>Power Calibration Instrumentation</u>

Listed below are the instruments used to perform the power calorimetric and their Calibration Procedure, if applicable. (Note: The computer points are not calibrated because the computer performs a self-check of the analog input modules. This test replaces the manual calibration. There are no provisions on the instruments for field calibrating flow and temperature elements.)

Feedwater Flow
FE641A & B (Calibration Procedure N/A)
PTD644A & B (Calibration Procedure 8.E.6)
Computer Point FWR114 & 116 (Calibration Procedure N/A)

Feedwater Temperature
TT261-25A & B (Calibration Procedure 8.E.6)
TT261-26A & B (Calibration Procedure 8.E.6)
TE261-33A & B (Calibration Procedure N/A)
Computer Point FWR002, 004, 006, 008 (Calibration Procedure N/A)

Reactor Pressure
PT647A & B (Calibration Procedure 8.M.2-6.1.1)
Computer Point RXX001 & 004 (Calibration Procedure N/A)

Barometric Pressure
PT6201(Calibration Procedure (Calibration Procedure 8.F.1)
Computer Point MTR002 (Calibration Procedure N/A)

Control Rod Drive Flow
FE302-53 (Calibration Procedure N/A)
FT302-54 (Calibration Procedure 8.E.3-1)
SQRT340-17 (Calibration Procedure 8.E.3-1)
Computer Point CRD002 (Calibration Procedure N/A)

Condenser Hotwell
TE6223B (Calibration Procedure N/A)
Computer Point CON024 (Calibration Procedure N/A)

Reactor Water Cleanup Flow
FE1279-74A & B (Calibration Procedure N/A)
FT1279-75A & B (Calibration Procedure 8.E.12)
P/E1279-77A & B (Calibration Procedure 8.E.12)
Computer Point RWC010 & 012 (Calibration Procedure N/A)

Reactor Water Cleanup Temperature
TE1279-10 (Calibration Procedure N/A)
TE1279-45 (Calibration Procedure N/A)
TE1279-36 (Calibration Procedure N/A)
Computer Point RWC002, 004 & 008 (Calibration Procedure N/A)

Recirc Pump Power
202-60-780A & B (Calibration Procedure N/A)
Computer Point REC130 &132 (Calibration Procedure N/A)

b. Software and Hardware configuration is maintained by Procedure 1.5.14 "Process Computer Maintenance and Updating" and by the Pilgrim Software Quality Assurance program.

c. The Entergy Nuclear Northeast Corrective Action Process ensures that when issues requiring action are identified, appropriate corrective actions are identified and tracked to completion.

d. Pilgrim has several programs for reporting deficiencies to manufacturers. The first is the Part 21 program for Q equipment. For non-Q equipment, the Corrective Action Program often initiates a notice to the vendor if the issue is considered significant. Vendors and other plants are also notified of equipment issues through the Operating Event (OE) report system and by Pilgrim's participation in Owner's Groups and other industry forums.

e. Manufacturer deficiency reports, are handled by the OE and Corrective Action processes discussed above. This includes NRC notifications, various vendor information reports and material identified on the OE database.

5. NRC Request: In reference to Section 2.5 of Attachment 2 (TSAR) to the amendment request, provide a summary describing the effect of the proposed power uprate on the structural integrity of the control rod drive mechanisms (CRDMs). Confirm that the existing design basis analysis for stress and fatigue cumulative usage of the CRDMs remains unchanged for the proposed 1.5 percent power uprate.

Response:

The components of the CRDM, which form part of the primary pressure boundary, have been designed in accordance with the applicable ASME B&PV Code, Section III. The CRDM structural and functional integrity is acceptable for a bottom head pressure of at least 1250 psig. The CRD mechanism also has been evaluated for higher postulated abnormal operating pressures and conditions that that subsequently apply the maximum CRD pump discharge pressure to the CRD mechanism internal components.

The CRD mechanism has been evaluated for the proposed 1.5% power uprate operating conditions and found to be acceptable. The CRDM qualification is based on the temperature and internal reactor differential pressure changes caused by 1.5% power uprate operating conditions relative to the CRDM structural design margins. Therefore, the existing design basis analysis for stress and fatigue cumulative usage of the CRDMs remains unchanged for the proposed 1.5% power uprate for Pilgrim.

6. NRC Request: In Section 3.2.2 of Attachment 2 (TSAR) to the amendment request, you indicated that the effect of Thermal Power Optimization (TPO) was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. For the components under consideration, the 1965 Edition of the Code with addenda to and including Summer 1966, which is the construction code of record, was used as the governing Code. You also indicated that if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Provide a summary of the components that were modified and the code editions/code cases (if applicable) other than the code of record that were used for the power uprate evaluation.

Response:

Table 3-2 of the Pilgrim TPO Safety Analysis Report (TSAR), NEDC-33050P, presents the results of the fatigue analysis of the limiting components. The feedwater nozzle thermal sleeve was modified in 1979 and is the only identified limiting component that was modified. The fatigue curves from the 1989 edition of the ASME Code were used, although, these curves have not changed since the 1980 edition of the ASME Code.

7. NRC Request: In Table 3-2 of Section 3.1, you indicated that the cumulative fatigue usage factor (CUF) for the feedwater nozzle is less than 0.8 for the current rated condition and less than 1.0 for the power uprate condition. Provide the actual calculated CUFs. Also, provide the allowable stress limits for reactor vessel components listed in Table 3-2. In reference to Section 3.3.2, provide a summary describing the effect of the proposed power uprate on the existing stress and fatigue analysis of the reactor internals. Also, provide comparison of calculated stresses and CUFs (similar to Table 3-2) for the limiting reactor internal components including allowable stress limits.

Response:

A CUF for the feedwater nozzle was not recalculated at TPO conditions. However, an existing calculation using a dome pressure of 1,000 psig and dome temperature of 546 degree F established a feedwater fatigue usage factor of U=0.600 (system cycling only) and U<0.8 (system + rapid cycling). The feedwater CUF value for current conditions presented in Table 3-2 of the Pilgrim TPO Safety Analysis Report (TSAR), NEDC-33050P, reflects the results of the existing calculation since the conditions approach the current licensed conditions.

At TPO conditions, the system cycling usage has been recalculated and is insignificantly affected (U increased to U=0.604).

The feedwater nozzle experiences rapid cycling during steady state operation. During steady state, the differential temperature between the feedwater nozzle and the RPV increases slightly due to TPO conditions. This will cause the fatigue usage to increase slightly. The changes in fatigue at TPO operating conditions due to rapid cycling were therefore not recalculated.

The feedwater CUF value at TPO conditions presented in Table 3-2 indicates that the acceptance criterion is satisfied although an exact value was not calculated.

The allowable stress limits for the components listed in Table 3-2 are:
Recirculation Outlet Nozzle: 51.80 ksi
Feedwater Nozzle: 53.10 ksi
CRD Nozzle: 60.0 ksi

Regarding the reactor internals, the loads due to pressure, temperature, weight, seismic and flow were either bounded by the design basis values or the changes due to TPO were insignificant. For components falling in the first category, no additional analyses were performed. For the components of the second category, where the changes in loads were insignificant, the evaluation of the TPO effect is qualitatively done consistent with TLTR and design basis. Therefore, recalculation of the stresses and CUFs was not required.

8. NRC Request: In reference to Section 3.5.3, you state that the original code of record for BOP piping and also for most safety-related systems was ANSI B31.1. However, in your presentation on July 24, 2002, you indicated that design evaluation for SRV discharge line piping for the power uprate is in accordance with the requirements of ASME B&PC Code Section III 1977 Edition through Summer 1977 addenda. Provide the codes, code editions that were used for the RCPB piping and BOP piping for the power uprate.

Response:

The original piping Code used for the reactor coolant pressure boundary (RCPB) at Pilgrim was the USAS B31.0.0, 1967 Edition. The USAS B31.7, 1969 Edition was used for main steam fatigue evaluation. The main steam lines were later reanalyzed and the current analysis of record is per the ASME/USAS B31.1.0, 1973 Edition and does not include a fatigue evaluation. The original fatigue evaluation performed for the main steam lines demonstrated that the transients were minor and thus, usage factors were not significant. When the recirculation piping was replaced, the piping was analyzed using ASME III, 1980 Edition, through the winter 1981 Addenda. As Part of the MK 1 Containment Program, ASME III, 1977 Edition, through summer 1977 Addenda was used for the current safety relief valve (SRV) discharge piping analysis (per NUREG-0661). Power uprate reanalysis of the main steam and SRV discharge piping to address the increased loads from the larger throat SRVs is being performed to the ASME III, 1977 Edition, through Summer 1977 Addenda, Subsection NC-Class 2. A fatigue evaluation was not performed as a part of the design basis SRV discharge line analysis. The discharge lines were originally considered balance of plant piping. The MK 1 Containment Program did tabulate SRV line cumulative usage factors for the BWR fleet and determined that fatigue usage factors were small enough to obviate the need for a plant unique analysis. The current ISI classification is ASME ISI Class 3 or lower. Therefore, no fatigue evaluation is required by Code.

The BOP piping Code used for the power uprate piping changes on non-safety related piping was "Power Piping ASME B31.1-1989 Edition."

9. NRC Request: In reference to Section 3.5.1, you indicated that the Response Spectrum Independent Support Motion (ISM) piping analysis methodology was applied for the SRVDL. At the July 24, 2002 meeting with the staff in Rockville, MD, you indicated the need to increase the SRV capacity as a result of the power uprate, and therefore, the current design margins of this piping system are reduced, due to the higher piping loads induced by the increased SRV flow. The analysis for the SRVDL was therefore performed using the ADLPIPE computer code, the ISM methodology and Regulatory Guide (RG) 1.61 damping values. The ISM analysis used the square root of the sum of squares (SRSS) approach for combining Group responses. This approach does not correspond to the staff position as stated in NUREG 1061, Volume 4. Likewise, the use of damping values in the analysis is not in accordance with the licensing basis for Pilgrim.

A. Provide the user manual, including the theoretical basis and benchmarking verification problems, for the ISM option in the ADLPIPE computer code.

B. Provide the justification for using the RG 1.61 damping values instead of the licensing basis damping values, as shown in the FSAR, for application with the ISM approach.

C. Provide the maximum stresses and CUFs for the SRVDL at the critical locations, subjected to the following loading conditions:

(1) Current operating conditions.

(2) Power uprate operating conditions, using the uniform response spectrum piping analysis approach, and licensing basis damping values.

(3) Power uprate operating conditions, using the ISM piping analysis approach based on the Absolute Sum of Group responses, and licensing basis damping values.

(4) Power uprate operating conditions, using the ISM piping analysis approach based on the SRSS combination of Group responses, and licensing basis damping values.

(5) Power uprate operating conditions, using the ISM piping analysis approach based on Absolute Sum of Group responses, and RG 1.61 damping values.

D. Provide the licensing basis stress allowables.

Response:

Response to RAI 9A:

Attachment 2 provides the user manual, including the theoretical basis and benchmarking verification problems, for the ISM option in the ADLPIPE computer code. The calculation documenting the ISM option is identified as PNPS Calculation No. 1214, Revision 0, ADLPIPE Bench Marking and Verification of ISM Piping Analysis and can be audited at Pilgrim Station.

Response to RAI 9B:

Attachment 3 provides the justification for using the RG 1.61 damping values instead of the licensing basis damping values, as shown in the FSAR, for application with the ISM approach.

Response to RAI 9C:

A fatigue evaluation was not performed as a part of the design basis SRV discharge line analysis because a MK 1 Containment Program study determined that fatigue usage factors were small [Reference: Technical Report TR-5310-1, Rev. 2, "Mark I Containment Program, PUAR Pilgrim, September 14, 1984, Appendix A4.2-10, Item 7]. The response to RAI 8 contains some additional discussion.

Attachment 4 provides the response to RAIs 9C1 through 9C5.

(1) The response is provided by viewing Case 1.

(2) The response is provided by viewing Case 42A2.

(3) The response is provided by viewing Case 74 (for 2D results) and Case 74R (for 3D results).

(4) The response is provided by viewing Case 71 (for 2D results) and Case 71R (for 3D results).

(5) The response is provided by viewing Case 77 (for 2D results) and Case 77R (for 3D results).

Response to RAI 9D:

For the piping in question, the main steam and safety relief valve discharge piping inside the drywell, all piping material is A106, Grade B carbon steel. The licensing basis (ANSI B31.1 1967) allowable for this material is 15,000 psi for both Sc and Sh. The analysis currently being performed is using the 15,000 psi allowable since that is the allowable from the code under which the material was procured. The code of record proposed for the new analysis, ASME Section III 1977 Summer 1977 Addenda, specifies the same value of 15,000 psi for both Sc and Sh.

ATTACHMENT 2

LETTER NUMBER 2.02.096

Response to RAI 9A

Double-click Icon to view Attachment

ATTACH2.PDF

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Bench Marking ISM Analysis with NUREG/CR-1677

The U.S. Regulatory Commission Piping Review Committee Report NUREG-1061 has recommended that the independent support motion (ISM) response spectrum method should be allowed as an option in calculating the response of multiple supported piping with independent inputs.

NUREG/CR-1677, Volume II presents benchmark problems and solutions designed to assess the adequacy of computer programs used to determine the inertial component response of linear elastic piping subjected to seismic induced, independent support excitations in three directions, evaluated using ISM response spectrum method of analysis.

PNPS has used Problem No. 2 to benchmark ADLPIPE computer program for the ISM analysis. Attached PNPS Report No. M1214, Rev. 0 presents results of the benchmark analysis. The results derived by ADLPIPE program are essentially identical to the results presented in NUREG/CR-1677. The small differences that were observed can be considered reasonable and not significant considering two different piping analysis codes were used. Therefore, the ISM method as presented in ADLPIPE Release 10 is considered benchmarked with NUREG CR-1677.

Documents Attached:

1. Copies of the input requirements from the ADLPIPE Release 10 - User Manual specific to ISM analysis.

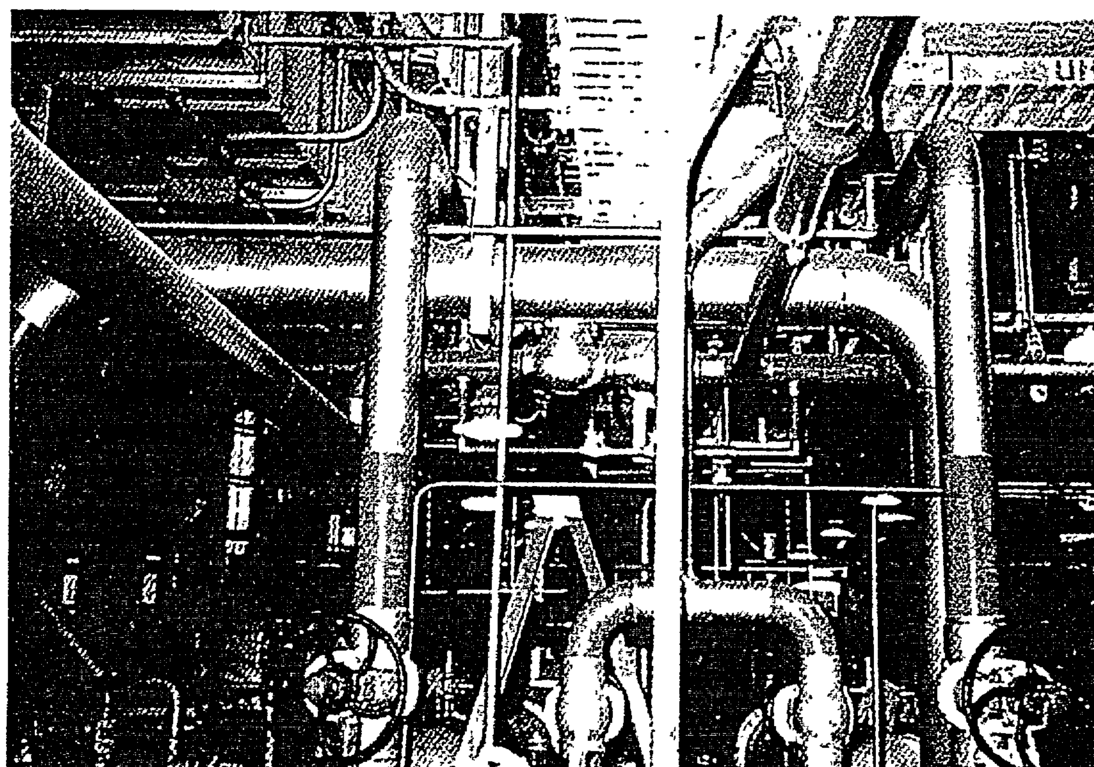
2. PNPS Calculation No. 1214, Rev. 0 ADLPIPE Bench Marking and Verification of ISM Piping Analysis

ADLPIPE

Release 10

Windows95/NT

Complete
Piping
Analysis
and
Design
Solution



User Manual

3.2.10 Multiple Response Spectra Analysis

Purpose:
This selection allows the user to define a load set for multiple response spectra analysis

Interactive Input:

◊ Select *Yes* to modify the pressure distribution: Select *No* to use the pressure distribution defined during the routing See Section 3.2.1, *Deadweight* on how to modify pressure distribution

◊ Select one of the lumped mass options·
  - ⇒ Lump all points
  - ⇒ Lump all point except supports
  - ⇒ User defined lumped points - select the node number from a list of nodes

◊ Define a cut-off frequency or mode If both frequency and mode are defined. the cut-off frequency is determined by the lesser of the two criterions If the cut-off frequency or mode is not defined. all modes are considered in the response spectrum analysis

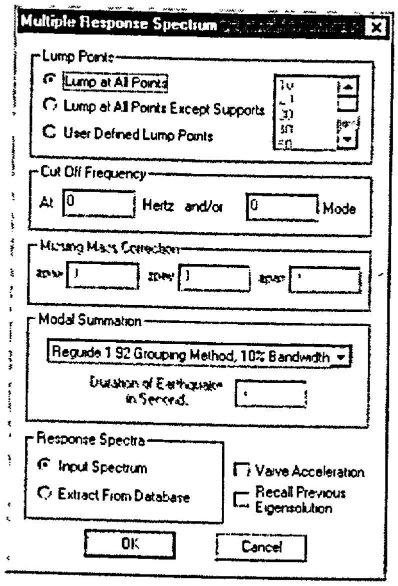
◊ Missing mass correction will be included if the ZPA are defined.

◊ Define a modal summation criterion.

◊ Check valve acceleration if the acceleration of the mass points are to be calculated.

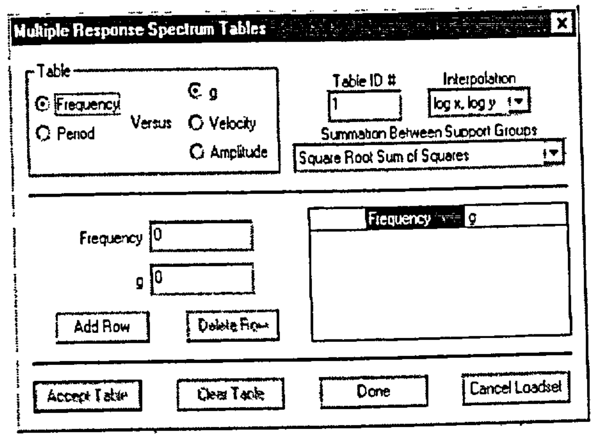
◊ Check *Recall Previous Eigensolution* to use the eigensolution of the previous load set.

◊ Select a method to input the seismic response spectra: input manually or extract from the response spectra database. Please see Section 1.7.4 *Database* on how to set up the database.



◊ The following input menu is used to input a response spectrum manually.

⇒ Select the type of a response spectrum table.
⇒ Define a unique ID for the response spectrum table
⇒ Define the method of interpolation used for the response spectrum
⇒ Define the criteria used for the summation between supports
⇒ Enter data points, click *Add Row* to include the data point in the response spectrum Click *Delete Row* to remove a data point from the response spectrum Click *Accept Table* to accept a response spectrum, click *Clear Table* to discard a response spectrum Click *Done* after all the response spectra are entered. Click *Cancel Loadset* to cancel the creation of the loadset.

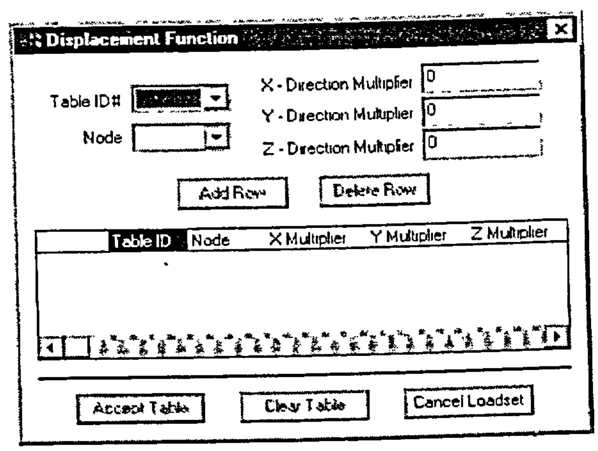


◊ After all the response spectra are defined, the next input menu is used to define how the response spectra are applied to each support.

⇒ Enter or select the ID of the response spectrum.

⇒ Enter or select the node of the support.

⇒ Enter the multiplier in the direction(s) that the response spectrum is applied The spectral data will be multiplied by the multiplier. Normally the multiplier is equal to 1 0.



Notes:

1) Please refer to Section 3.4 2.8 Seismic multiple Response Spectrum for the explanation of the text input instructions.

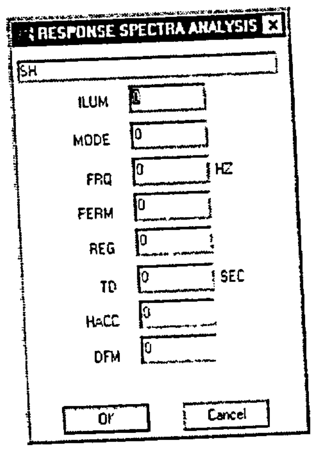
Also See:

SHOCK	Define parameters for analysis	Section 3.4.2.7.1
TBLE	Define response spectra	Section 3.4.2.8 2
X	Define response spectra	Section 3.4.2.8.3
Y	Define response spectra	Section 3 4.2.8.3
DF	Apply response spectrum	Section 3.4 2.8 4
INERTIA	Define mass point	Section 3.4.6.3

3.4.2.7.1 Shock

Purpose:
This selection allows the user to define parameters for a seismic single or multiple response spectrum analysis

Interactive Input:



Text Input:

General Format:

SHOCK.ILUMP.MODE,FRQ.PERMOD,REGUIDE ,TD,HACC,DFM

Description:

The SHOCK instruction is the load instruction that instructs ADLPIPE to compute the natural frequencies and modal response of the piping system The forces. moments and stresses computed by the SHOCK analysis are for dynamic motion only.

ILUMP	=	0.	Consider only the mass entered on the INERTIA instruction. If this option is chosen, the lumped mass should be calculated manually and entered on the INERTIA instructions.
ILUMP	=	1,	The distributed and concentrated weight of the piping system will be added to the masses on the INERTIA instruction
ILUMP	=	7,	Same as ILUMP = 1 except that this option requires all junctions to be defined as lumped mass points. A junction is defined as a node with three or more pipe elements. i.e. the center of a tee.
ILUMP	=	8.	Same as ILUMP = 7 except that ADLPIPE will create extra INERTIA instructions at the network points if there are no INERTIA instructions input at these points
MODE	=	0.	All modes are included in the response spectrum analysis.
MODE	=	N.	Only the first N modes are included in the response spectrum analysis
MODE	=	-N.	ADLPIPE will compute the first N frequencies then terminate the analysis This option should be used for checking the natural frequencies
FRQ	=	0,	All modes are included in the response spectrum analysis.
FRQ	=	F.	Only the modes with natural frequencies up to F hertz are included in the response spectrum analysis.
FRQ	=	-F,	ADLPIPE will compute the frequencies up to F hertz then terminate the analysis. This option should be used for checking the natural frequencies.
PERMOD	=		The bandwidth in percent which is used to define whether a cluster of modes are closely spaced. This input parameter is only valid if REGUIDE parameter is 1.60 or 0 The default value for REGUIDE= 1.60 or 0 is 10%.

For REGUIDE equal to 1.92. 1.921, or 1.922, the bandwidth is ten percent and is not allowed to modify.

For REGUIDE equal to 1.70 (SRSS), the bandwidth is zero and is not allowed to be modified.

REGUIDE	=	1.92 ,	NRC Regulatory Guide 1.92. modal summation technique. the grouping method, ten percent bandwidth (paragraph 1.2.1, Regulatory Guide 1.92)
REGUIDE	=	1.921,	Same as REGUIDE =1.92
REGUIDE	=	1.922,	NRC Regulatory Guide 1.92, modal summation technique, the ten percent method, ten percent bandwidth (paragraph 1.2.2. Regulatory Guide 1.92).
REGUIDE	=	1.923.	NRC Regulatory Guide 1 92. modal summation technique. the double sum method, (paragraph 1.2.3. Regulatory Guide 1.92).
REGUIDE	=	1 70.	The modal amplitudes are computed by the SRSS of each earthquake direction The modal response is summed in accordance with SRSS method The parameter 1.70 does not refer to an existing NRC Regulatory Guide.
REGUIDE	=	1.60.	The modal amplitudes are computed by the absolute sum of each earthquake direction. The modal responses are summed in accordance with SRSS method with PERMOD considered. The parameter 1.60 does not refer to an existing NRC Regulatory Guide.
REGUIDE	=	0.	For a unidirectional earthquake analysis in which the response directional components represent a single earthquake skew to the axis of the structure being analyzed. Do not use this technique for the analysis of nuclear piping in accordance with Regulatory Guide 1.92 as it is not acceptable to the current NRC guidelines.
TD	=		Duration, in seconds, of the earthquake for the double sum method
HACC	=		Acceleration threshold value in "g" that is

used as the criterion by "MASS POINT AND ANCHOR ACCELERATION REPORT". This report includes all mass nodes with acceleration exceeding the threshold value. The default of the threshold value is 0.01g

DFM = Dynamic displacements are calculated on a modal basis and those modes that have all displacements less than DFM will be ignored.

It is not required to input a DFM unless the analysis is used to benchmark against an ADLPIPE output using version D42 or earlier versions In the versions prior to D42, the DFM was set to 0 01 inch in ADLPIPE.

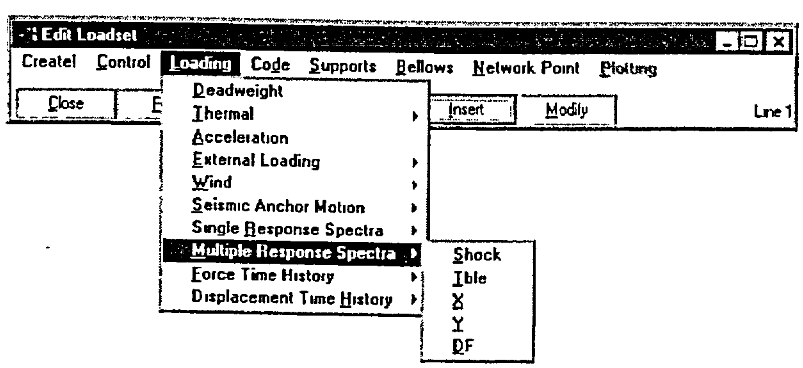
Notes:

1) If MODE and FRQ are both entered, the cut-off mode depends on the fewest specified by the MODE or FRQ parameter.

3.4.2.8 Seismic Multiple Response Spectra

A load set for a multiple response spectra analysis requires the following instructions

EXECUTE	The title information for load set. Section 3.4.1.1, *Control-Execution*
SAVE	Optional, only required for recalling the eigensolution from the preceding multiple response spectra analysis Section 3.4.1.7, *Control-Save*
CODE selector	Please refer to Section 3.4.3.1
MATERIAL	Please refer to Section 3.4.3.2. This input is not required if SAVE is included in the load set
SBSP	Optional, request subspace iteration for eigensolution Section 3.4.1.6, *Control-Subspace*
SHOCK	Load instruction, define parameters for the analysis
TBLE	Define a response spectrum
X	Define a response spectrum
Y	Define a response spectrum
DF	Apply a response spectrum at a support node
DAMPING	Only required for NRC REGUIDE 1.92 double sum method
INERTIA	Define lumped mass points This input is not required if SAVE is included in the load set. Section 3.4.6.3, *NetworkPoint - Inertia*
END	End of load set Section 3.4.1.8, *Control-End*



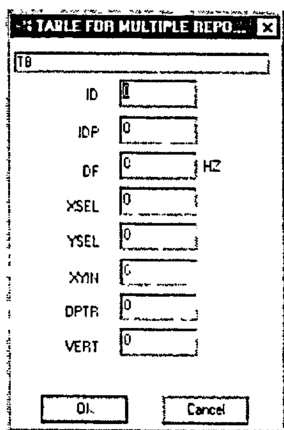
3.4.2.8.1 Shock

This instruction is also used by the seismic single response spectrum analysis Please see Section 3.4.2.7.1 for the explanation

3.4.2.8.2 Tble - Multiple Response Spectra

Purpose:
This selection allows the user to define a unique identification for a response spectrum table that is referenced in the seismic multiple response spectra analysis. It also allows the user to define the properties of the response spectrum and the summation method between the support groups. Use of the TBLE instruction and the SHOCK instruction automatically activate the multiple response spectra analysis

Interactive Input:



Text Input:

General Format:

TBLE , ID , IDP . DF , XSEL . YSEL , XYINT . OPTR , VERT

Description:

ID	=	The identification of the response spectrum table, an integer between 1 and 99999.
IDP	=	A previously defined table number. If IDP is entered, the set of frequencies for response spectrum IDP will be used by this table with the frequency shift parameter, DF.
DF	=	Frequency shift parameter to be used if IDP is entered This shift is incorporated into the set of frequencies defined by the table IDP and this table will use the new set of frequencies.
		If DF = -1, the set of frequencies for the table IDP will be used without shifting.
XSEL	=	Frequency/period selector
	=	0 or blank, X instruction values are frequency (Hertz or CPS)
	=	1., X instruction values are period (second)
YSEL	=	g/velocity/amplitude selector
	=	0. or blank, Y instruction values are g
	=	1., Y instruction values are velocity (in/sec, mm/sec, mm/sec)
	=	2., Y instruction values are amplitude (in, mm, mm)
XYINT	=	Interpolation selector
	=	0. or blank, log X, log Y
	=	1 , log X, linear Y
	=	2 , linear X, log Y
	=	3., linear X, linear Y
OPTR	=	Support group combination option, this parameter should be entered once on the first TBLE instruction
	=	3., cumulative intramode motion is calculated by the algebraic sum of each foundation contribution within a support group having the same response spectrum. The group combination is by absolute summation in each mode.
	=	4., cumulative intramode motion is calculated by the algebraic sum of each foundation contribution within a support group having the same response spectrum. The group combination is by square root sum of squares (SRSS) in each mode.
	=	0. or blank, cumulative intramode motion is calculated by adding the absolute value of each foundation contribution in each mode. This is not a grouping method.

= 1., cumulative intramode motion is calculated by square root sum of square of each foundation contribution in each mode This is not a grouping method.

= 2., cumulative intramode motion is calculated by the algebraic sum of each foundation contribution in each mode. This method is used by the single response spectrum analysis. This is not a grouping method

VERT = Vertical axis
= 1, X axis
= 2, Y axis (default)
= 3, Z axis

Notes:

1) If IDP and DF are entered. the X instructions (frequency/period) should be omitted.
2) The support group combination option, OPTR = 3 or 4, is consistent with the methods used in the Volume II of the NUREG/CR-1677, Piping Benchmark Problems, Dynamic Independent Support Motion Response Spectrum Method

3.4.2.8.3 X/Y-Response Spectrum Table

Purpose:
These selections allow the user to define a response spectrum for a multiple response spectra analysis in the tabular format.

Text Input:

General Format:

X, L, K, X(L), X(L+1), X(L+2), X(L+3), X(L+4), X(L+5)
Y, L, K, Y(L), Y(L+1), Y(L+2), Y(L+3), Y(L+4), Y(L+5)

Description:

Value of frequency (XSEL=0), or period (XSEL=1) may be input on the X instruction in ascending order. The corresponding value of g (YSEL=0), velocity (YSEL=1), or amplitude (YSEL=2) may be input on the Y instruction.

L	=	Subscript of the first entry on the X/Y instruction
K	=	Subscript of the last positive entry on the X/Y instruction
X(L ~ L+5)	=	Frequency associated with L~L+5 subscript for XSEL=0 (Hertz or cps)
	=	Period associated with L~L+5 subscript for XSEL=1 (sec)
Y(L ~ L+5)	=	Acceleration associated with L~L+5 subscript for YSEL=0 (g)
	=	Velocity associated with L subscript for YSEL=1 (in/sec, mm/sec, mm/sec)
	=	Amplitude associated with L subscript for YSEL=2 (in, mm, mm)

Notes:

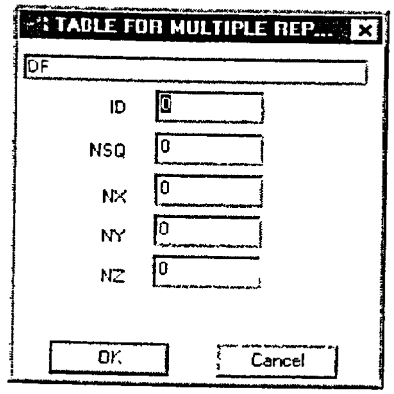
1) All the X instructions follow the TBLE instruction for each response spectrum table and give the frequencies/periods at which spectral data are known. The X instruction is used to input the frequency/period data and all X instructions precede the spectral data which are input

by the Y instructions. Below the lowest frequency/period, the spectral data of the lowest frequency/period is used. Above the highest frequency/period, the spectral data is set to zero.

3.4.2.8.4 DF-Apply a Response Spectrum

Purpose:
This selection allows the user to apply a response spectrum that is defined by TBLE/X/Y to a restrained support. This selection is valid for a multiple response spectra analysis.

Interactive Input:



Text Input:

General Format:

DF, ID , NP , NX , NY , NZ

Description:

ID	=	The identification of a pre-defined response spectrum table.
NP	=	The node number of the support point where the response spectrum is applied
NX	=	A positive factor. If this factor is entered, the response spectrum is multiplied by the factor and applied to the node NP in the X direction
NY	=	A positive factor. If this factor is entered, the response spectrum is multiplied by the factor and applied to the node NP in the Y direction.

NZ = A positive factor. If this factor is entered, the response spectrum is multiplied by the factor and applied to the node NP in the Z direction.

Notes:

1) If multiple seismic directions are included in a load set and the response spectra at a support are varied in each direction, more than one DF instruction may be needed to define the spectrum at a node

CALCULATION COVER PAGE

PILGRIM NUCLEAR POWER STATION		
Calculation No. M1214	**Revision** 0	**Sheet** 1 **of** 8 (plus 63 pages of attachments)
Title: ADLPIPE Bench Marking and Verification of ISM Piping Analysis		**Status:** ☐ Preliminary ☐ Pending ☐ As-Built ☐ Void ☐ Superceded
☒ QR ☐ NQR		
Discipline: Mechanical/Civil/Structural Department	**Design Basis Calculation?** ☐ Yes ☒ No	
This calculation supercedes calculation: N/A		
Modification No./Task No/ER No: N/A		
Software Used? ☒ Yes ☐ No (If Yes, include Computer Run Summary Sheet)		
System No./Name: N/A **Component No./Name:** N/A		
Print / Sign		
Preparer: P. K. Shah P.K. Shah	**Date:** 10/9/2002	
Reviewer/Design Verifier: Gerry Bechen Gerald J Bechen	**Date:** 10-9-02	
Other Reviewer/Design Verifier:	**Date:**	
Approver Ray Pace [signature]	**Date:** 10/10/02	

RECORD OF REVISIONS

Revision No.	Description of Change	Reason For Analysis
0	Initial Issue	M1187Rev. 0 did not contain verification problems for Independent Support Motion (ISM) methods for seismic analysis. This analysis incorporates verification problems for Independent Support Motion (ISM) methods for seismic analysis in ADLPIPE Release 10.

CALCULATION SUMMARY

CALCULATION OBJECTIVE:
This calculation presents verification problems specific to Independent Support Motion (ISM) method for seismic analysis using ADLPIPE Release 10. These problems were not included in M1187, Rev. 0.

These ISM problems have been also benchmarked with the problems in NUREG/CR-1677.

The purpose of M1187, Rev. 0 calculation was to document the qualifications performed to establish the computer program ADLPIPE Release 10 - Version 4F10.1 and ADLPOST Version F10.0 (PC version) as a "Q" program in accordance with the requirements of NOP95A2. Calculation M1187, Rev. 0 remains valid in its entity.

CONCLUSIONS:
ADLPIPE Release 10 is considered to be a "Q" program that can be fully utilized for seismic analysis in the evaluation and design of safety related piping systems using ISM method.

The results are in very good agreement with NUREG/CR-1677 problems. Therefore, ISM method as presented in ADLPIPE Release 10 can be considered verified and benchmarked.

ASSUMPTIONS:
No Assumptions were necessary. This calculation presents solutions of the verification problems as provided by the Software Vendor. The results are compared with the results provided by the vendor for verification.

DESIGN INPUT DOCUMENTS:
ISM Problems were provided by Email by the software vendor which are documented in Attachment A. No additional design input was necessary.

AFFECTED DOCUMENTS:
This analysis (M1214) supplements M1187, Rev. 0.
Calculation M1187, Rev. 0 remains valid in its entity.

METHODOLOGY:
- Input files for ISM method verification were received from the ADLPIPE vendor.
- These files were used to generate the output at PNPS. These output were compared with the vendor supplied out, and the installation of ADLPIPE Release 10 for ISM analysis was thus verified.
- The output generated at PNPS was also compared with the NUREG/CR-1677 output, and thus the ISM analyses results were benchmarked.

TABLE OF CONTENTS

LIST OF EFFECTIVE PAGES

Page	Revision
All pages 1 through 8	0
Attachment A	0
Attachment B	0
Attachment C	0
Attachment D	0
Attachment E	0

CALCULATION SECTIONS

1. Background

The present analysis M1214 verifies ISM method for performing seismic analysis on a stand alone PC at PNPC. The stand alone PC used for this analysis is the same PC that was used for M1187, Rev. 0 (Ref. 6.1) analysis for qualifying ADLPIPE Release 10 (Ref. 2) as a "Q" program.

The purpose of M1187 calculation was to document the qualifications performed to establish the computer program ADLPIPE Release 10 - Version 4F10.1 and ADLPOST Version F10.0 (PC version) as a "Q" program in accordance with the requirements of NOP95A2. ADLPIPE Release 10 may interchangeably be identified as ADLPIPE Version 4F10.1 in this analytical report.

The set of verification problems in M1187 did not contain problems for performing seismic analysis using Independent Support Motion (ISM) method. Research Engineering Inc. (supplier of ADLPIPE Release 10) provided input and output of two problems that applied ISM method for seismic analysis. One verification problem applied "square root of sum of squares" (SRSS) group combination and the other problem applied absolute group combination. These results were bench marked against the results of the same problems in NUREG CR-1677 (Ref. 6.3).

As such, M1187, Rev. 0 remains valid in its entity. The present calculation M1214 only supplements M1187; it does not revise any part of M1187.

2. Purpose

Refer to "Objective" in the Calculation Summary (Page 3) and the above Section for Background.

3. Method of Analysis

Refer to "Methodology" in the Calculation Summary (Page 3)

4. Assumptions

Refer to "Assumptions" in the Calculation Summary (Page 3)

5. Input and Design Criteria

Refer to "Design Input Documents" in the Calculation Summary (Page 3)

6. References

a) M1187, Rev.0 ADLPIPE 10 Verification
b) ADLPIPE Release 10 (also called as ADLPIPE Release 10 – Version 4F10.1)
c) NUREG/CR-1677 Vol. II, Piping Benchmark Problems – Dynamic Analysis Independent Support Motion Response Spectrum Method, August 1985.

7.0 Calculations / Analysis

Research Engineers, Inc. provided two verification problems with input listings that were used to benchmark ISM method of seismic analysis using ADLPIPE Release 10 with NUREG-1677 (Ref. 3). The input listing is included in Attachment A.

7.1 Problem 1 - Verification for ISM Analysis Using SRSS Group Combination

The input file is NRC2SRSS.adi. The output file that was generated using ADLPIPE Release 10 is NRC2SRS_output.adi. Both of these files are stored on the stand-alone PC (Serial No. 6848BXH2A427 described in Ref. 1) in a directory c:\ADLPIPE10\ISM_Method. Also, both of these files are stored on hp-cd-recordable (CD-R). The CD-R is in the custody of the System Administrator for ADLPIPE.

This problem is a benchmark problem for Problem 2b presented in NUREG 1677 (Ref. 3). The piping model used is shown on page 78 of Ref. 3 (See Page 7 of Attachment A). The solution is presented on pages 101 through 126 of Ref. 3 (See Pages 2 through 27 of Attachment E). The piping analysis code that was used in Ref. 3 is PSAFE2. The following table presents a comparison between the NUREG-1677 results and ADLPIPE Release 10 results.

SRSS Group Combination
Benchmarking Maximum Displacement (inches)

Node No.	NUREG CR-1677, Vol 2			ADLPIPE Release 10			Difference
	DX	DY	DZ	DX	DY	DZ	
5	0.029	0.0109	0.0398	0.029	0.011	0.040	Negligible

Benchmarking Maximum Resultant Moment(lb-inches)

Node No.	NUREG CR-1677, Vol 2	ADLPIPE	% Deviation
1	11621.7116	11625.801	0.035

Summary:

- SRSS Group Combination Verification: The results of ADLPIPE Release 10 generated on the stand-alone PC are identical with the results provided by Research Engineers Inc. which are presented in Attachment D. Therefore, the ADLPIPE Release 10 as installed at PNPS is considered verified.

- SRSS Group Combination Benchmarking: The results of ADLPIPE Release 10 generated on the stand alone PC matches the results provided in NUREG -1677, Vol. 2 within reasonable accuracy (0.035%). Considering the piping analysis codes used are different, 0.035% deviation is considered acceptable. Therefore, the ADLPIPE Release 10 as installed at PNPS is considered benchmarked with NUREG-1677.

7.2 Problem 2 - Verification Problem for ISM Analysis Using Absolute Group Combination

The input file is NRC2abs.adi. The output file that was generated using ADLPIPE Release 10 is NRC2abs_output.adi. Both of these files are stored in a directory c:\ADLPIPE10\ISM_Method on the stand-alone PC described in Ref. 1. Also, both of these files are also stored on hp-cd-recordable (CD-R). The CD-R is in the custody of the System Administrator for ADLPIPE.

This problem is a benchmark problem for Problem 2c presented in Ref. 3. The piping model used is shown on page 78 of Ref. 3. The solution is presented on pages 127 through 149 of Ref. 3.

The piping analysis code that was used in Ref. 3 analysis is PSAFE2. The following table presents a comparison between the NUREG-1677 results and ADLPIPE Release 10.

Absolute Group Combination
Benchmarking Maximum Displacement (inches)

Node No.	NUREG CR-1677, Vol. 2 (inch)				ADLPIPE Release 10 (inch)				% Difference
	DX	DY	DZ	Resultant	DX	DY	DZ	Resultant	
5	0.0411	0.0156	0.0566	0.0717	0.0406	0.0149	0.0548	0.070	2.6%

Benchmarking Maximum Resultant Moment (lb-inches)

Node No.	NUREG CR-1677, Vol 2 (FT-LB)	ADLPIPE Release 10 (PNPS Results)			ADLPIPE Release 10 Resultant (lb-Inches)	% Difference
		MX (FT-LB)	MY (FT-LB)	MZ (FT-LB)		
1	16202.65	1182	368	494	15994.63	1.28%

Summary:

- Absolute Group Combination Verification: The results of ADLPIPE Release 10 generated on the stand-alone PC are identical with the results provided by Research Engineers Inc. which are in file a nrcabs_results.adi. This file is stored in a directory c:\ADLPIPE10\ISM_Method on the stand-alone PC. Also, this file is also stored on hp-cd-recordable (CD-R). The CD-R is in the custody of the System Administrator for ADLPIPE. Therefore, the ADLPIPE Release 10 as installed at PNPS is considered verified.

- Absolute Group Combination Benchmarking: The results of ADLPIPE Release 10 generated on the stand alone PC matches with the results provided in NUREG -1677, Vol. 2 within reasonable accuracy (1.28% for moment and 2.6% for displacement). The piping analysis codes used are different. The differences in the bit definition of variables results in small round off deviations. (Refer to Appendix D for explanation of platform effects.) These deviations are considered acceptable. Therefore, the ADLPIPE Release 10 as installed at PNPS is considered benchmarked with NUREG-1677.

Conclusion:

ISM Method for performing seismic analysis was evaluated using ADLPIPE Release 10 piping analysis program. The stand-alone PC used was same as that was used for M1187 analysis. SRSS and absolute group combination approaches for evaluating ISM results were compared with the results provided in NUREG CR-1677. The small differences that were observed can be considered reasonable and not significant considering two different piping analysis codes were used. Therefore, the ISM method as presented in ADLPIPE Release 10 is considered benchmarked with NUREG CR-1677.

The results generated at PNPS were also compared with the results provided by the software vendor Research Engineers, Inc. All the results were in agreements with only very minor differences. Therefore, the ISM method presented in ADLPIPE Release 10 as used on the stand-alone PC at PNPS is considered verified.

File "NRC2SRSS.adi" from Research Engineers Inc.
SRSS Group combination

```
GE,NUREG/1677 MRS BENCHMARK 2
GE,DECK 406C
RE,,1,,,,1.,1.,1.
AN,,1,0,0,0
RE,,21,,,,1.,1.,1.
AN,,21,18.7167,12.1,1.6667
RE,,22,1.,1.,1.,1.,1.,1.
RE,,23,1.,1.,1.,1.,1.,1.
RE,,24,1.,1.,1.,1.,1.,1.
RE,,25,1.,1.,1.,1.,1.,1.
RE,,26,1.,1.,1.,1.,1.,1.
RE,,27,1.,1.,1.,1.,1.,1.
RE,,28,1.,1.,1.,1.,1.,1.
RE,,29,1.,1.,1.,1.,1.,1.
RE,,30,1.,1.,1.,1.,1.,1.
RE,,31,1.,1.,1.,1.,1.,1.
RE,,32,1.,1.,1.,1.,1.,1.
RE,,33,1.,1.,1.,1.,1.,1.
RE,,34,1.,1.,1.,1.,1.,1.
RE,,35,1.,1.,1.,1.,1.,1.
AN,,17,9.025,24.1667,18.7583
SE
PI,1,6,7.288,.2410,.240E2,,,2.179
RU,1,2,,4.5375
RU,2,3,,7.5625
EL,3,4,,,,36.3,45
EL,4,5,,,,36.3
RU,5,6,4.5125
SE
RU,6,7,4.5125
EL,7,8,,,,36.3,45
EL,8,9,,,,36.3
RU,9,10,,,4.7333
RU,10,11,,,1.7084
TE,11,12,,,1.7083
SE
TE,12,18,3.3333
RU,18,19,6.3584
EL,19,20,,,,36.3
RU,20,21,,,-6.4833
SE
TE,12,13,,,1.7083
RU,13,14,,,8.9
EL,14,15,,,,36.3
RU,15,16,,7.5667
RU,16,17,,4.5
SE
RU,1,22,.01
2S,1,22,.1E11
SE
RU,1,23,,.01
2S,1,23,.1E11
SE
RU,1,24,,,.01
2S,1,24,.1E11
SE
```

```
RU,7,25,,,-.01
2S,7,25,.1E9
SE
RU,9,26,.01
2S,9,26,.1E9
SE
RU,11,27,,.01
2S,11,27,.1E5
SE
RU,13,28,,.01
2S,13,28,.1E5
SE
RU,15,29,-.08333,.08333,.11667,-.11667,,-.08333
2S,15,29,.1E9
SE
RU,17,30,.01
2S,17,30,.1E11
SE
RU,17,31,,.01
2S,17,31,.1E11
SE
RU,17,32,,,.01
2S,17,32,.1E11
SE
RU,21,33,.01
2S,21,33,.1E11
SE
RU,21,34,,.01
2S,21,34,.1E11
SE
RU,21,35,,,.01
2S,21,35,.1E11
END
EXEC,SRSS GROUP ,SRSS MODAL
xp,-2,-27,20
SHOCK,1,25,,0.00001,1.70
SB
CL,,,1.,1977.
CO,1,1,10,350
TBL,1,0,0,1.,0.,3.,4
X,1,6,.0256,.0286,.0303,.0909,.1166,.1515
Y,1,6,.22,.22,.25,.42,.85,1.290
TBL,2,0,0,1.,0.,3,4
X,1,2,.0256,.0313
X,3,8,.0351,.0625,.0649,.1069,.1149,.1515
Y,1,2,.19,.19
Y,3,8,.205,.205,.22,.42,.68,.9
TBL,3,0,0,1,0,3,4
X,1,6,.0256,.0385,.0455,.0699,.1242,.1818
Y,1,6,.17,.17,.2,.24,.47,.63
TBL,4,0,0,1.,0.,3,4
X,1,4,.0256,.0455,.0568,.303
Y,1,4,.23,.23,.03,.38
TBL,5,0,0,1.,0.,3,4
X,1,4,.0256,.0645,.125,.1818
Y,1,4,.37,.37,.65,.85
TBL,6,0,0,1.,0.,3,4
X,1,5,.0256,.0571,.0769,.1254,.1538
Y,1,5,.55,.55,.65,1.,1.3
TBL,7,0,0,1.,0.,3,4
X,1,5,.0256,.0606,.0661,.1010,.1515
Y,1,5,.65,.65,.77,.9,1.75
TBL,8,0,0,1.,0.,3,4
```

```
X,1,5,.0256,.0364,.0404,.0909,.1136
Y,1,5,.23,.23,.25,.42,.73
TBL,9,0,0,1.,0.,3,4
X,1,6,.0256,.0313,.0349,.0588,.0713,.1136
Y,1,6,.112,.112,.125,.135,.18,.346
TBL,10,0,0,1.,0.,3,4
X,1,4,.0256,.0555,.0673,.1136
Y,1,4,.126,.126,.14,.215
TBL,11,0,0,1.,0.,3,4
X,1,6,.0256,.0357,.0541,.0673,.1033,.1212
Y,1,6,.21,.21,.26,.36,.42,.73
DF,1,22,1.
DF,2,23,,1.
DF,3,24,,,1.
DF,6,25,,,1.
DF,4,26,1.
DF,5,27,,1.
DF,5,28,,1.
DF,4,29,1.
DF,5,29,,1.
DF,6,29,,,1.
DF,7,30,1.
DF,8,31,,1.
DF,9,32,,,1.
DF,10,33,1.
DF,11,34,,1.
DF,11,35,,,1.
IN,,1,.01,.01,.01
IN,,2,.01,.01,.01
IN,,3,.01,.01,.01
IN,,4,.01,.01,.01
IN,,5,.01,.01,.01
IN,,6,.01,.01,.01
IN,,7,.01,.01,.01
IN,,8,.01,.01,.01
IN,,9,.01,.01,.01
IN,,10,.01,.01,.01
IN,,11,.01,.01,.01
IN,,12,.01,.01,.01
IN,,13,.01,.01,.01
IN,,14,.01,.01,.01
IN,,15,.01,.01,.01
IN,,16,.01,.01,.01
IN,,17,.01,.01,.01
IN,,18,585.94,585.94,585.94
IN,,19,.01,.01,.01
IN,,20,.01,.01,.01
IN,,21,.01,.01,.01
END
```

File "NRC2abs.adi" from Research Engineers Inc.
Absolute Group combination

```
GE,NUREG/1677 MRS BENCHMARK 2
GE,DECK 406C
RE,,1,,,,1.,1.,1.,
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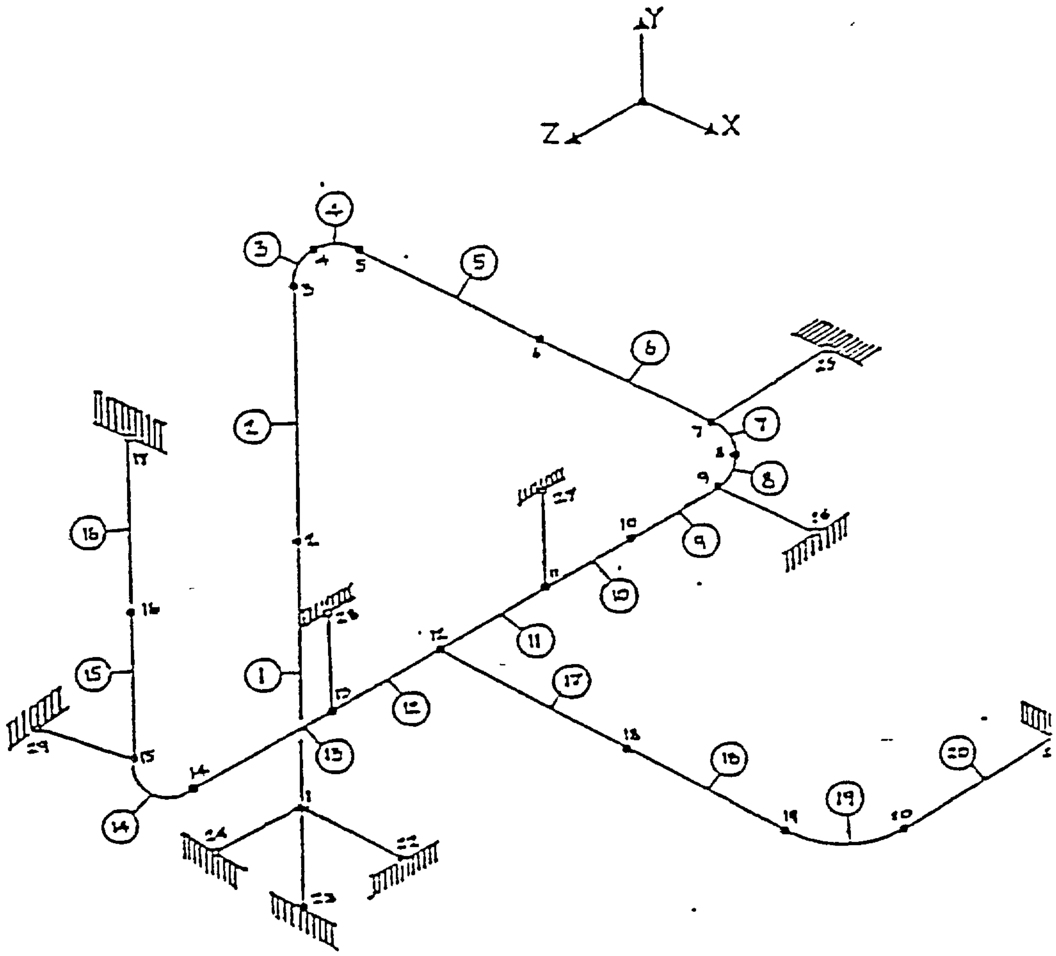

BENCHMARK PROBLEM № 2

Problem No. 2 Model
(From Page 77 of NUREG/CR-1677)

50.59 REVIEW FORM

Facility: PNPS

Document Reviewed: Calculation M-1214 Rev 0, "ISM Piping Analysis Verification of ADLPIPE Release 10"

System Designator(s): 1 – Main Steam

Check the applicable review(s):

			RTYPE
☐	EXCLUSION	Sections I and II required	A9.02
☒	SCREENING	Sections I and II required	A9.02
☐	50.59 EVALUATION EXEMPTION	Sections I, II, and III required	A9.02
☐	50.59 EVALUATION Evaluation #: ______	Sections I, II, and IV required	A9.03

NOTE: Only the sections required as indicated above must be included in the Review.

I. SIGNATURES / OVERVIEW

Preparer: P.K. Shah P.K. Shah / 10-9-02
Signature / Name (print) / Company / Work Group / Date

Reviewer: Gerald Bechen Gerald Bechen /ENN/ NESG-Programs & Components 10-9-2002
Signature / Name (print) / Company / Work Group / Date

Approver: ENN / MCS / 10/10/02
Signature / Name (print) / Company / Work Group / Date

S&SA Supv/Mgr: ______
Signature / Date (N/A for 50.59 Screenings, Exemptions, and Exclusions)

ORC: ______
Chairman's Signature / Date (N/A for Screenings and 50.59 Evaluation Exemptions)

List of Assisting/Contributing Personnel:

Name: **Scope of Assistance:**

Description of Proposed Change

Calculation M1214 rev 0, provides verification of the Independent Support Motion (ISM) method as incorporated by Reasearch Engineers, Inc in its ADLPIPE software product. Results calculated by ADLPIPE were verified and benchmarked against results presented in NRC NUREG/CR-1677.

50.59 Review Form

II. SCREENING

A. Licensing Basis Document Review

Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒		
TS	☐	☒		
NRC Orders	☐	☒	☐	
If any are "YES", obtain NRC approval prior to implementing the change. (See base document Step 5.2.1[13] for exceptions.)				

LBDs controlled under 50.59	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
UFSAR	☐	☒		
TS Bases	☐	☒		
Core Operating Limits Report	☐	☒		
Fire Hazard Analysis	☐	☒	☐	
Fire Protection Program	☐	☒	☐	
Offsite Dose Calculations Manual	☐	☒	☐	
Process Control Program (PNPS 1.15.3)	☐	☒	☐	
NRC Safety Evaluation Reports[1]	☐	☒		
If any are "YES", perform an Exemption Review in accordance with Section III OR perform a 50.59 Evaluation in accordance with Section IV.				

LBDs controlled under other regulations	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual [2]	☐	☒		
Emergency Plan [2]	☐	☒		
Security Plan [2,3]	☐	☒		
Inservice Inspection Program [4]	☐	☒	☐	
Inservice Testing Program [4]	☐	☒	☐	
If any are "YES", evaluate/process any changes in accordance with the appropriate regulation.				

Notes:
1 If "YES", see base document Step 5.2.1[5]

2 If "YES", notify the responsible work group and ensure a 50.54 Evaluation is performed.

3 The Security Plan is classified as Safeguards and can only be reviewed by personnel with the appropriate security clearance. The Preparer should notify Security of potential changes to the Security Plan.

4 If "YES", process the change in accordance with the 10CFR50.55a control program.

B. **Does the proposed activity involve a test or experiment not described in the FSAR?**

☐ Yes
☒ No

If "yes," perform an Exemption Review in accordance with Section III OR perform a 50.59 Evaluation in accordance with Section IV.

C. Basis

(Provide a basis for the "no" items checked in Sections II.A and II.B above. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. If a 50.59 Evaluation is required, this section may be marked "N/A".)

Section II.A boxes were checked NO for the following reasons.

Calculation M1214 rev 0, provides verification of the Independent Support Motion (ISM) method as incorporated by Reasearch Engineers, Inc in its ADLPIPE software product. Results calculated by ADLPIPE were verified and benchmarked against results presented in NRC NUREG/CR-1677.

Operating License, Technical Specifications, and NRC Orders:
A review of the Operating License, Technical Specifications, and NRC Orders was performed. This calculation does not impact the Operating License, Technical Specifications, or NRC Orders. Revision of the Operating License, Technical Specifications, or NRC Orders is not required to support the use of this calculation.

UFSAR:
A search of the USFAR resulted in no section being impacted by the issuance and use of this calculation. Seismic analysis is discussed in the USFAR. However, UFSAR revision is not required to support this use of this calculation.

TS Bases:
The Technical Specification Bases were reviewed. The Technical Specification Bases provide a summary of the reasons behind the associated Technical Specification. The issuance and use of this calculation does not have any impact on or require revision of the TS Bases.

Core Operating Limits Report:
The Core Operating Limits Report (COLR), Rev. 14A for Cycle 14 was reviewed. in its entirety. The Core Operating Limits Report specifies cycle specific core operating limits. The core operating limits are APRM and RBM trip settings, APLHGR, LHGR, MCPR, and core power/flow relationship limits. The core design for the specific fuel cycle is also described. The parameters used to determine the core limits are not used in calculation M1214 rev 0. Therefore, the issuance and use of calculation M1214 does not have any impact on the Core Operating Limits Report.

Fire Hazards Analysis:
PNPS Updated Fire Hazards Analysis (UFHA), Report No. 89XM-1-ER-Q, Rev. E5 was reviewed. The PNPS Updated Fire Hazards Analysis provides the methodology and analysis results delineating how the fire protection and safe shutdown requirements of BTP 9.5-1 Appendix A and 10CFR50.48 including Appendix R are met. The UFHA divides the plant into zones that have been analyzed in the Updated Fire Hazards Analysis Report. Calculation M1214 has nothing to do with Fires or Fire Hazards The issuance and use of Calculation M1214 does not impact the Updated Fire Hazards Analysis Report nor do they relate in any way to any fire hazard.

Fire Protection Program:
PNPS NOP83FP1, Fire Protection Plan, Rev. 6 was reviewed. The purpose of the Fire Protection Plan is to describe the PNPS fire protection program as required by 10CFR50.48 and to protect Entergy's investment in PNPS. PNPS NOP83FP1 is basically an administrative procedure that describes the fire protection organization, responsibilities and philosophy at PNPS. Calculation M1214 has no impact on the fire protection program. The issuance and use of Calculation M1214 does not affect any of the fire protection requirements specified in the Fire Protection Plan.

Offsite Dose Calculation Manual:
The Offsite Dose Calculation Manual (ODCM) was reviewed. The purpose of the ODCM is to present the methodology, parameters, data and information used to calculate offsite doses due to radioactive gaseous and liquid effluents, to calculate gaseous and liquid effluent monitor alarm/trip setpoints, and to administer the conduct of the radiological environmental monitoring program. Radioactive gaseous effluent monitoring instrumentation requirements, setpoints, surveillance, and release limits are specified. In addition, the methodology for calculating offsite dose from radioactive gaseous releases is specified. The issuance and use of Calculation M1214 does not impact any of the methodology, parameters, data or information in the ODCM.

Process Control Program:
PNPS 1.15.3, Process Control Program, Rev. 5 was reviewed. The purpose of the process Control Program is to provide administrative and operational controls for the processing, solidification, dewatering and packing of the applicable radwaste forms for ultimate disposal. None of the issues addressed in calculation M1214 impact the process control program. The issuance and use of Calculation M1214 does not affect the administrative and operational controls for the processing, solidification, dewatering and packing of the applicable radwaste forms for ultimate disposal as specified in PNPS 1.15.3.

NRC Safety Evaluation Reports:
The issuance and use of Calculation M1214 does not impact NRC Safety Evaluation Reports that were issued to Pilgrim. The plant modifications were previously evaluated under separate Safety Evaluations. Thus, the issuance and use of Calculation M1214 does not impact NRC Safety Evaluation Reports issued to Pilgrim.

Quality Assurance Program Manual:
The Entergy Quality Assurance Program Manual (EQAPM) was reviewed in its entirety. EQAPM defines the quality assurance program at PNPS. There are no administrative and operational controls contained in Calculation M1214 that have an impact on the requirements specified in the EQAPM. The issuance and use of Calculation M1214 does not impact on the Entergy Quality Assurance Program Manual.

Emergency Plan:
The Emergency Plan was reviewed in its entirety. The Emergency Plan describes the emergency preparedness program at PNPS. The plan outlines the basis for response actions that would be implemented in an emergency and documents the methods by which PNPS meets the criteria set forth in 10CFR50 Section 47(b) and Appendix E. There are no administrative and operational controls contained in Calculation M1214 that have an impact on the administrative controls or implementing procedures used at PNPS to deal with emergency situations. Therefore, the issuance and use of Calculation M1214 does not impact the Emergency Plan.

Security Plan:

It is logically determined that Calculation M1214 has no impact on the requirements specified in the Security Plan. The issuance and use of Calculation M1214 has no impact on the Security Plan.

Inservice Inspection Program:

The Inservice Inspection Program at PNPS is documented in the "Pilgrim Station 3rd Interval Inservice Inspection Plan", Rev. 2. The ISI Plan outlines the requirements for the inspection of Class 1, 2, and 3 pressure retaining components and their supports at PNPS. Calculation M1214 has no impact on the requirements specified in the ISI Program. The issuance and use of Calculation M1214 has no impact on the ISI Program.

Inservice Testing Program:

PNPS 8.I.1.1, Inservice Pump and Valve Testing Program, Rev. 14, was reviewed in its entirety. The purpose of PNPS 8.I.1.1 is to identify the pumps and valves included in the inservice testing program and to specify the testing requirements for compliance with 10CFR50.55a(f), Inservice Testing Requirements. The details contained in calculation M1214 have no impact on the requirements specified in the IST Program. The issuance and use of Calculation M1214 has no impact on the IST Program.

Section II.B box was checked NO for the following reason:

Calculation M1214 rev 0 provides verification of the Independent Support Motion (ISM) method as incorporated by Reasearch Engineers, Inc in its ADLPIPE software product. Results calculated by ADLPIPE were verified and benchmarked against results presented in NRC NUREG/CR-1677.
It has no affect on the performance of leak rate testing of valves, penetrations, and seals. Leak rate testing is discussed in UFSAR Section 5.2. However, Calculation M1214 rev 0 has no affect on the leak rate testing as described in the UFSAR.

D. Is the validity of this Review dependent on any other change?

☐ Yes
☒ No

If "Yes," list the required changes.

E. References

[Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., keywords) or the general extent of manual searches in accordance with base document Step 5.2.2[2](d).]

Documents:

1. PNPS USFAR
2. PNPS Technical Specifications
3. PNPS Procedures
4. PNPS Procedure 8.I.1.1, Inservice Pump And Valve Testing Program, Rev. 14
5. PNPS NOP83FP1, Fire Protection Plan, Rev. 6
6. PNPS Emergency Plan, Rev. 24
7. PNPS Offsite Dose Calculation Manual, Rev. 8
8. PNPS Core Operating Limits Report, Rev. 14A
9. Updated Fire Hazards Analysis, Report No. 89XM-1-ER-Q, Rev. E5
10. Entergy Quality Assurance Program Manual, Rev. 7
11. Pilgrim Station 3rd Interval Inservice Inspection Plan, Rev. 2
12. PNPS Procedure 1.15.3, Process Control Program, Rev.5

Keywords:
Electronic search of Technical Specifications and UFSAR using the following keywords:

ADLPIPE, Pipe analysis, modal combination, group combination, independent support, absolute sum, NUREG/CR-1677, support group, stress analysis, analysis software.

FSAR Sections Reviewed:

1. 12.2.3.5.4
2. 12.2.3.5.4
3. 12.2.3.5.4

FSAR Figures/Tables Reviewed:

The UFSAR and Technical Specifications were searched electronically using the keywords listed above. Each hit was reviewed for impact by calculation M1214 rev 0.

III. 50.59 EVALUATION EXEMPTION

Enter this section only if a "Yes" box was checked in either Section II.A or II.B above.

A. **Check the applicable box (es) below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section IV. Provide supporting documentation or references as appropriate.**

- ☐ The proposed activity is editorial/typographical as defined in base document Step 5.2.4[1].
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Step 5.2.4[2]. (Insert sub step letter from base document Step 5.2.4[2].)
- ☐ The proposed activity impacts design function as described in base document Step 5.2.4[3] as follows:

  The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; **AND**

  The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; **AND**

  The proposed activity does not adversely affect an evaluation that demonstrates intended functions of an SSC described in the FSAR will be accomplished.
- ☐ The proposed activity, or portions thereof, is controlled by another regulation instead of 50.59 in accordance with base document Step 5.2.4[4]. (Portions of the change not controlled under the other program must be evaluated under 50.59.)
- ☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed change already exists in accordance with base document Step 5.2.4[5]. Refer to 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.
- ☐ The proposed activity, in its entirety, has been approved by the NRC in accordance with base document Step 5.2.4[6].
  Reference: ____________________

B. **Basis**

(Provide an adequate basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions.)

IV. 50.59 EVALUATION

A. **Executive Summary** (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to Regulatory and Industry Affairs after ORC approval, if available.)

Brief description of change, test, or experiment:

Reason for proposed Change:

50.59 Evaluation summary and conclusions

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 - 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☐ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☐ No

   BASIS:

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☐ No

   BASIS:

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☐ No

   BASIS:

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☐ No

   BASIS:

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☐ No

   BASIS:

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☐ No

   BASIS:

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☐ No

   BASIS:

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☐ No

   BASIS:

DESIGN VERIFICATION COVER PAGE

☐ IP-2 ☐ IP-3 ☐ JAF ☒ PNPS

Document No. M1214	Revision 0	Page 1 of 6

Title: ISM Piping Analysis Verification of ADLPIPE Release 10

☒ Quality Related ☐ Non Quality Related

DV Method: ☒ Design Review ☐ Alternate Calculation ☐ Qualification Testing

VERIFICATION REQUIRED	DISCIPLINE	VERIFICATION COMPLETE AND COMMENTS RESOLVED (DV print, sign, and date)
☐	Electrical	
☒	Mechanical	Gerald J. Bechen 10-9-02 Gerald J Bechen
☐	Instrument and Control	
☐	Civil/Structural	
☐		
☐		

Print/Sign After Comments Have Been Resolved

Originator: Gerald Bechen Gerald J Bechen	Date: 10-9-2002

IDENTIFICATION:			DISCIPLINE:
Document Title: ISM Piping Analysis Verification of ADLPIPE Release 10			☐ Civil/Structural
Doc. No.: M1214	Rev. 0	QA Cat. "Q"	☐ Electrical
Verifier: Gerald Bechen (Print)	______ (Sign)	10-9-2002 (Date)	☐ I & C
Manager authorization for supervisor performing verification ☐ N/A			☒ Mechanical
			☐ Other

METHOD OF VERIFICATION:

Design Review ☒ Alternate Calculations ☐ Qualification Test ☐

1. **Design Inputs** – Were the inputs correctly selected and incorporated into the design?	Reference
Design inputs include design bases, plant operational conditions, performance requirements, regulatory requirements and commitments, codes, standards, field data, etc. All information used as design inputs should have been reviewed and approved by the responsible design organization, as applicable. All inputs need to be retrievable or excerpts of documents used should be attached. See site specific design input procedures for guidance in identifying inputs.	Page No. ______ OR Paragraph No. ______ Completion of the Reference Boxes is optional for all questions.
Yes ☒ No ☐ N/A ☐	

Verifier Comments: Inputs are from REI (ADLPIPE) and NUREG and are included:

A. ADLPIPE input decks for ISM.

B. Excerpts from NUREG/CR-1677

Resolution: None required

2. **Assumptions** – Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are assumptions identified for subsequent re-verification when the detailed activities are completed?

Yes ☐ No ☐ N/A ☒

Verifier Comments: N/R

Resolution: N/R

Reference
Page No. ____
OR
Paragraph No. ____

3. **Quality Assurance** – Are the appropriate quality and quality assurance requirements specified?

Yes ☒ No ☐ N/A ☐

Verifier Comments: Document is "Q"

Resolution: None Required

Reference
Page No. ____
OR
Paragraph No. ____

4. **Codes, Standards and Regulatory Requirements** – Are the applicable codes, standards and regulatory requirements, including issue and addenda properly identified and are their requirements for design met?

Yes ☒ No ☐ N/A ☐

Verifier Comments: Agrees with NUREG/CR-1677

Resolution: None Required

Reference
Page No. ____
OR
Paragraph No. ____

5. **Construction and Operating Experience** – Have applicable construction and operating experience been considered?

Reference
Page No. ____

Yes ☒ No ☐ N/A ☐	OR Paragraph No. ______

Verifier Comments: None Required

Resolution: None Required

6. **Interfaces** – Have the design interface requirements been satisfied and documented? Yes ☐ No ☐ N/A ☒	**Reference** Page No. ______ **OR** Paragraph No. ______

Verifier Comments: None Required

Resolution: None Required

7. **Methods** – Was an appropriate analytical method used? Yes ☒ No ☐ N/A ☐	**Reference** Page No. ______ **OR** Paragraph No. ______

Verifier Comments: The method used is a simple comparison of two set of similar data.

Resolution: None Required

8. **Design Outputs** – Is the output reasonable compared to the inputs? Yes ☒ No ☐ N/A ☐	**Reference** Page No. ______ **OR** Paragraph No. ______

Verifier Comments: The results were verified and Benchmarked using an appropriate source.

Resolution: None Required

9. **Acceptance Criteria** – Are the acceptance criteria incorporated in the calculation sufficient to allow verification that design requirements have been satisfactorily accomplished? Yes ☒ No ☐ N/A ☐	**Reference** Page No. ______ **OR** Paragraph No. ______

Verifier Comments: The calculated ADLPIPE results agreed with NRC results.

Resolution: None Required

10. **Records and Documentation** – Are requirements for record preparation, review, approval, retention, etc., adequately specified? Are all documents prepared in a clear legible manner suitable for microfilming and/or other documentation storage method? Have all impacted documents been identified for update? Yes ☒ No ☐ N/A ☐	**Reference** Page No. ______ **OR** Paragraph No ______

Verifier Comments: None Required

Resolution: None Required

11.	**Software Quality Assurance-** For a calculation that utilized software applications (e.g., GOTHIC, SYMCORD), was it properly verified and validated in accordance with ENN IT-104 or previous site SQA Program? Yes ☒ No ☐ N/A ☐ **Verifier Comments:** ADLPIPE software is properly SQA'd **Resolution:** None Required	**Reference** Page No. ______ **OR** Paragraph No. ______

OTHER COMMENTS

RESOLUTIONS

THERE WERE NO "NO" answers

All comments for "NO" answers have been resolved satisfactorily.

(Note: The following response from Research Engineers explains the reason for small deviation in results for Absolute Group Combination method when the operating systems are different; Window'98 vs. Window 2000. The results are identical for SRSS group combination method even though the operating systems are different. Along with the following response, Research Engineers provided a output result file for Absolute Group Combination, nrc2abs_Results.adi, using Window '98, These results were identical with the results generated using the stand-alone PC at PNPS which are in a file nrc2abs_output.adi. Both of these files are stored in a directory c:\ADLPIPE10\ISM_Method on the stand-alone PC. Also, these files are also stored on hp-cd-recordable (CD-R). The CD-R is in the custody of the System Administrator for ADLPIPE.)

From: REMUMBAI [remumbai@bom5.vsnl.net.in]
Sent: Wednesday, August 28, 2002 11:42 AM
To: Shah, Pankaj
Subject: ADLPIPE

Sir,
The results you have got is most probably from Windows 98.We have ran the file in Windows 98 and got identical results. The slight difference of this result from the results sent to you earlier is due to the difference in the operating system.The bit-definition of type of variables may not be the same for all operating systems. Please find enclosed the output file received by us in Windows 98.

Regards
SantanuC
from
RESEARCH ENGINEERS (A Division of NetGuru Inc.)
241, Hill Road, Hill View No: 2,
Opposite Mehboob Studio,
Bandra (West)
Mumbai - 400 050.

Tel : 022 - 6426479 / 6552756
Fax : 022 - 6552766

Shah, Pankaj

From: Santanu C. [csantanu@ca.reiusa.com]
Sent: Thursday, August 15, 2002 7:21 PM
To: Shah, Pankaj
Subject: Re: ISM Analysis Info

Comparison doc   Nrc2abs.adl   NRC2SRSS adl

Dear Sir,

Please find enclosed the verification problem nos 2 of NUREG CR-1677 vol 2.The frequencies in the ADLPIPE output match with those in the standard reference(NUREG CR-1677).

Please also find a word document showing the comparison of maximum displacement and maximum resultant moment.

Regards
Support Desk

===

----- Original Message -----
From: "Shah, Pankaj" <PShah90@entergy.com>
To: <csantanu@ca.reiusa.com>
Sent: Monday, August 12, 2002 11:04 AM
Subject: ISM Analysis Info

> Dear Santanu Chakrabarti;
>
> We are in process of presenting our piping analyses that use ISM seismic
> analysis to NRC. The following information would be of great help to us.
>
> * Has NRC reviewed ISM analysis which was performed using ADLPIPE by
> other Nuclear utilities?
> * Does your verification problems set include ISM analysis? Which one?
> If it does not, would you provide a verification problem to qualify
ADLPIPE
> for ISM seismic analysis?
> * You said that one of the NUREG includes ISM analysis using ADLPIPE
> for benchmarking results. Which NUREG? More information on this item
will
> be very useful.
>
> In summary, any information that you can provide to defend ISM seismic
> analysis using ADLPIPE for ASME Class I analysis will be greatly
> appreciated.
>
>
> P.K. Shah, PE
>

Comparison of results of Piping Benchmark Problem No.2 from NUREG –CR 1677 vol 2
With those from ADLPIPE engine version 10.1

A] SRSS Summation:-

a) Maxm. Displacement(inches)

Node No.	NUREG CR-1677, Vol 2			ADLPIPE			Difference
	DX	DY	DZ	DX	DY	DZ	
5	0.029	0.0109	0.0398	0.029	0.011	0.040	Negligible

b) Maximum Resultant Moment(lb-inches)

Node No.	NUREG CR-1677, Vol 2	ADLPIPE	% Deviation
1	11621.7116	11625.801	0.035

B] Absolute Summation:-

a) Maxm. Displacement(inches)

Node No.	NUREG CR-1677, Vol 2			ADLPIPE			Difference
	DX	DY	DZ	DX	DY	DZ	
5	0.0411	0.0156	0.0566	0.041	0.016	0.056	Negligible

b) Maximum Resultant Moment(lb-inches)

Node No.	NUREG CR-1677, Vol 2	ADLPIPE	% Deviation
1	16202.6567	16197.088	0.034

NOTE: These results are using Window 2000 operating system which are slightly different from Window '98

Attachment E

Copy of Pages 101 through 137 From NUREG/CR-1677

- Pages 2 through 27
  Problem 2b Independent Support Motion Solution SRSS Group Combination
- Pages 28 through 38
  Problem 2c Independent Support Motion Solution Absolute Group Combination

Problem 2b

Independent Support Motion Solution
SRSS Group Combination

SUPPORT GROUP 1

DIRECTION FACTORS

X = 386.4000 Y = 386.4000 Z = 386.4000

INDICATOR FOR DISPLACEMENT OR ACCELERATION SPECTRUM = 1

EQ 0 DISPLACEMENT
EQ.1 ACCELERATION IN IN./SEC.2
EQ.2 ACCELERATION IN GS

3 SPECTRA ARE ENTERED FOR CASE 1 KIND= 2

CLUSTER FACTOR, CF = .00010

MODAL PARTICIPATION FACTORS

MODE	FREQ(CPS)	X-DIRECTION	Y-DIRECTION	Z-DIRECTION
1	9.360	-.1122E+00	.6439E-01	.3110E+00
2	12.706	.8941E-01	.1132E+00	.9867E-03
3	15.377	-.7981E-01	-.1836E-01	.3034E-01
4	17.797	.2474E+00	.3770E+00	-.1160E+00
5	21.603	.7131E-01	.6494E-01	.2769E+00
6	25.098	.3160E-01	-.3618E-01	.1765E+00
7	32.035	.8215E-01	.1021E-01	-.1523E-01
8	38.069	-.2948E-01	-.7938E-01	.4375E-01
9	40.293	.1151E+00	-.3142E-01	.1790E-01
10	48.898	.2728E+00	[illegible]	[illegible]
11	57.515	[illegible]	[illegible]	[illegible]
12	61.500	-.5360E-01	-.1953E-01	-.2121E+00
13	62.541	-.7594E-01	-.6786E-01	-.2305E+00
14	69.348	.1285E+00	-.1031E+00	.3435E-02
15	77.444	.1115E+00	-.1895E+00	.4605E-01
16	78.081	-.1609E+00	.7236E-02	.1247E+00
17	101.715	.8130E-02	-.2527E-01	.1866E-01
18	103.583	-.3154E-01	.4775E-01	.4883E-02
19	107.966	.8817E-01	-.1353E+00	.3535E-01
20	115.098	-.1552E-03	-.3572E-01	-.1125E+00
21	135.244	-.2822E-01	.3784E-01	.1955E-01
22	155.220	-.2700E-01	-.5084E-01	-.1833E-03
23	160.601	-.3018E-01	-.2437E-01	.4319E-03
24	203.789	.1278E-02	.4051E-01	.1400E-02
25	209.025	.1078E-01	.2488E+00	-.1109E-02

SPECTRUM TABLE (X SPECTRUM GROUP 1   )

NUMBER OF POINTS = 17
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.2200E+00
2	.2860E-01	.2200E+00
3	.3030E-01	.2500E+00
4	.9090E-01	.4200E+00
5	.1166E+00	.8500E+00
6	1515E+00	.1290E+01
7	.1748E+00	.1420E+01
8	.2137E+00	.1420E+01
9	.2849E+00	.1150E+01
10	.3704E+00	.7500E+00
11	.4545E+00	.6300E+00
12	.5555E+00	.6300E+00
13	.7407E+00	.3800E+00
14	.9091E+00	.5700E+00
15	.1111E+01	.5700E+00
16	.1389E+01	.2200E+00
17	.2500E+01	.1300E+00

SPECTRUM TABLE (Y SPECTRUM GROUP 1   )

NUMBER OF POINTS = 18
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.1900E+00
2	.3130E-01	.1900E+00
3	.3510E-01	.2050E+00
4	.6250E-01	.2050E+00
5	.8490E-01	.2200E+00
6	.1069E+00	.4200E+00
7	.1149E+00	.6800E+00
8	.1515E+00	.9000E+00
9	.1818E+00	.9980E+00
10	.2222E+00	.0980E+00
11	.2857E+00	.7700E+00
12	.5555E+00	.4200E+00
13	.6061E+00	.3500E+00
14	.6993E+00	.7150E+00
15	.0091E+00	.9050E+00
16	.1111E+01	.9950E+00
17	.1563E+01	.1800E+00
18	.2500E+01	.1800E+00

NUMBER OF POINTS = 15
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.1700E+00
2	.3850E-01	.1700E+00
3	.4550E-01	.2000E+00
4	.6990E-01	.2400E+00
5	.1242E+00	.4700E+00
6	.1818E+00	.6300E+00
7	.2222E+00	.6300E+00
8	.2778E+00	.5100E+00
9	.4000E+00	.3000E+00
10	.4545E+00	.3000E+00
11	.6494E+00	.7800E+00
12	.7273E+00	.1600E+01
13	.8889E+00	.1600E+01
14	.1667E+01	.1700E+00
15	.2500E+01	.1700E+00

CLUSTERING. +1 MEANS MODE I CLOSE TO I+1. -1, NOT.
-1. -1. -1 -1 -1. -1. -1 -1. -1. -1. -1. -1. -1. -1.. -1. -1. -1. -1. -1. -1. -1. -1 -1 -1 0.

SUPPORT GROUP 2
****

DIRECTION FACTORS

X = 386.4000 Y = 386.4000 Z = 386.4000

INDICATOR FOR DISPLACEMENT OR ACCELERATION SPECTRUM = 1

EQ.0 DISPLACEMENT
EQ 1 ACCELERATION IN IN./SEC.2
EQ.2 ACCELERATION IN GS

3 SPECTRA ARE ENTERED FOR CASE 2 KIND= 2

CLUSTER FACTOR, CF = .00010

MODAL PARTICIPATION FACTORS

MODE	FREQ(CPS)	X-DIRECTION	Y-DIRECTION	Z-DIRECTION
1	9.360	-.9420E+00	.3275E+00	-.8688E+00
2	12.706	.2870E-01	.1015E+01	.6397E-02
3	15.377	.6903E+00	-.2656E+00	.1391E+00
4	17.797	-.3614E-01	.2207E+00	.1607E+00
5	21.603	.3470E+00	-.1951E+00	.4168E+00
6	25.098	.2763E+00	.1041E+00	.9865E+00
7	32.035	.1785E+00	-.3384E+00	-.1913E+00
8	38.069	.2183E-01	.2849E+00	- 1593E-01
9	40.293	.4088E+00	.9595E-01	-.1530E+00
10	48.898	.2799E+00	.8265E-01	-.1721E+00
11	57.515	-.1037E+00	.2301E+00	.2639E+00
12	61.500	.1597E+00	- 1320E+00	.3315E-01
13	62.541	.2225E+00	.9838E-01	.1036E+00
14	69.348	.1566E+00	-.3275E-01	-.2162E+00
15	77.444	.3348E+00	-.5642E-01	-.3167E+00
16	78.881	4796E+00	.3210E-01	-.4258E+00
17	101.715	.5668E-01	.1807E-01	.8629E-02
18	103.583	.6029E-01	-.4751E-01	-.6563E-01
19	107 066	-.8069E-01	.5838E-01	.1688E+00
20	115.098	.4180E-01	.6366E-01	-.2616E+00
21	135.244	-.2822E+00	.1588E+00	3962E+00
22	155.220	.1612E+00	-.2725E-01	-.5932E-01
23	160 601	-.7781E-01	-.4070E-01	- 3619E-01
24	203 780	-.9466E-01	.7401E-03	.1625E-01
25	200.025	- 3881E-02	-.1188E-02	.1188E-01

Calc. No. 1214, Rev. 0
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SPECTRUM TABLE (X SPECTRUM GROUP 2

NUMBER OF POINTS = 11
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.2300E+00
2	.4550E-01	.2300E+00
3	.5680E-01	.3000E-01
4	.3030E+00	.3800E+00
5	.5000E+00	.7500E+00
6	.6061E+00	.1070E+01
7	.6993E+00	.2250E+01
8	.8547E+00	.2250E+01
9	.1852E+01	.2300E+00
10	.2083E+01	.1700E+00
11	.2500E+01	.1700E+00

SPECTRUM TABLE (Y SPECTRUM GROUP 2

NUMBER OF POINTS = 11
SCALE FACTOR = 10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	3700E+00
2	6450E-01	3700E+00
3	.1250E+00	.6500E+00
4	.1818E+00	.8500E+00
5	.2222E+00	.8500E+00
6	.4041E+00	.9000E+00
7	.5682E+00	1500E+01
8	.7273E+00	.2870E+01
9	.8089E+00	.2870E+01
10	.1639E+01	.3000E+00
11	.2500E+01	.1700E+00

SPECTRUM TABLE (Z SPECTRUM GROUP 2

NUMBER OF POINTS = 14
SCALE FACTOR = 10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2580E-01	.5500E+00
2	.5710E-01	.5500E+00
3	.7690E-01	.6500E+00
4	.1254E+00	.1000E+01
5	.1538E+00	.1300E+01
6	.1818E+00	.1380E+01
7	.2222E+00	.1380E+01

8	.3704E+00	.1080E+01
9	.5682E+00	.1950E+01
10	.7273E+00	.3550E+01
11	.8889E+00	.3550E+01
12	.1667E+01	.3620E+00
13	.2000E+01	.2000E+00
14	.2500E+01	.2000E+00

CLUSTERING. +1 MEANS MODE I CLOSE TO I+1. -1, NOT.
-1, -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1 -1. -1. 0.

SUPPORT GROUP 3

DIRECTION FACTORS

X = 386.4000 Y = 386.4000 Z = 386.4000

INDICATOR FOR DISPLACEMENT OR ACCELERATION SPECTRUM = 1

EQ.0 DISPLACEMENT
EQ.1 ACCELERATION IN IN./SEC.2
EQ.2 ACCELERATION IN GS

3 SPECTRA ARE ENTERED FOR CASE 3 KIND= 2

CLUSTER FACTOR, CF = 00010

MODAL PARTICIPATION FACTORS

MODE	FREQ(CPS)	X-DIRECTION	Y-DIRECTION	Z-DIRECTION
1	9.360	-.7529E-01	-.2967E+00	-.1691E+00
2	12.706	-.3327E-01	.1407E+00	.5527E-01
3	15.377	-.5500E-01	.1436E+00	.2765E-01
4	17.797	.3986E-02	-.2167E+00	-.5547E-01
5	21.603	.3835E-01	.2543E+00	.1519E+00
6	25.098	-.1933E-01	-.2516E+00	-.1332E+00
7	32 035	-.1382E+00	-.6691E-01	-.6119E-01
8	38.069	-.1114E+00	.1437E-02	-.9437E-01
9	40.293	.1130E+00	-.5580E-01	-.3728E-01
10	48.898	-.4419E-01	.8519E-01	.5065E-02
11	57.515	-.1190E+00	-.3555E+00	.1655E+00
12	61.500	.5730E-01	-.9955E-04	.2957E-03
13	62.541	-.3506E-01	.2078E-01	-.2214E-02
14	69.348	.9052E-01	-.1314E+00	.6276E-01
15	77.444	-.3383E-01	.1582E+00	-.4416E-01
16	78.881	.2271E-01	.3494E-01	.2113E-01
17	101.715	.6760E-02	.2779E-01	.6204E-02
18	103.583	-.4437E-02	-.9379E-03	-.8561E-02
19	107.966	.6415E-02	.9544E-02	.1289E-01
20	115.008	.6132E-02	.7141E-01	.1155E-01
21	135.244	8480E-02	.1035E+00	.2234E-01
22	155.220	.3247E-02	- 3714E-01	-.1026E-02
23	180.601	-.4322E-02	-.1200E+00	- 6413E-02
24	203.789	.2379E-03	-.9738E-03	.1883E-03
25	209.025	.1331E-03	-.1059E+00	- 1293E-02

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SPECTRUM TABLE (X SPECTRUM GROUP 3

NUMBER OF POINTS = 15
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.6500E+00
2	.6060E-01	.6500E+00
3	.6610E-01	.7700E+00
4	.1010E+00	.9000E+00
5	.1515E+00	.1750E+01
6	.1748E+00	.2050E+01
7	.2137E+00	.2050E+01
8	.2924E+00	.1600E+01
9	.3636E+00	.1460E+01
10	4132E+00	.1700E+01
11	.5865E+00	.2400E+01
12	.7273E+00	.4150E+01
13	.8889E+00	.4150E+01
14	.1661E+01	.4200E+00
15	.2500E+01	1800E+00

SPECTRUM TABLE (Y SPECTRUM GROUP 3

NUMBER OF POINTS = 14
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	2560E-01	.2300E+00
2	.3640E-01	.2300E+00
3	.4040E-01	.2500E+00
4	.9090E-01	.4200E+00
5	.1136E+00	.7300E+00
6	.1515E+00	1320E+01
7	.1852E+00	.1320E+01
8	.3831E+00	5600E+00
9	.5555E+00	.5300E+00
10	.9259E+00	.2800E+00
11	.1111E+01	1700E+00
12	1420E+01	.2800E+00
13	.1736E+01	.2800E+00
14	.2500E+01	.1400E+00

SPECTRUM TABLE (Z SPECTRUM GROUP 3

NUMBER OF POINTS = 18
SCALE FACTOR = .10000E+01

JT SPECTRUM

1	.2560E-01	.1120E+00
2	.3130E-01	.1120E+00
3	.3490E-01	.1250E+00
4	.5880E-01	.1350E+00
5	.7130E-01	.1800E+00
6	.1136E+00	.3460E+00
7	.1515E+00	.5500E+00
8	.1852E+00	.5500E+00
9	.2778E+00	.3300E+00
10	.3759E+00	.2140E+00
11	.6173E+00	.1700E+00
12	.7017E+00	.1250E+00
13	.7273E+00	.1950E+00
14	.9091E+00	.2500E+00
15	.1070E+01	.4980E+00
16	.1357E+01	.5350E+00
17	.1658E+01	.5350E+00
18	.2500E+01	.2240E+00

CLUSTERING. +1 MEANS MODE I CLOSE TO I+1. -1, NOT
-1 -1. -1. -1 -1 -1. -1 -1 -1 -1 -1 -1 -1 -1 -1 -1. -1. -1 -1. -1. -1. -1 -1 -1 0.

SUPPORT GROUP 4

DIRECTION FACTORS

X = 386.4000 Y = 386.4000 Z = 386.4000

INDICATOR FOR DISPLACEMENT OR ACCELERATION SPECTRUM = 1

EQ.0 DISPLACEMENT
EQ 1 ACCELERATION IN IN /SEC.2
EQ.2 ACCELERATION IN GS

2 SPECTRA ARE ENTERED FOR CASE 4. KIND= 2

CLUSTER FACTOR, CF = .00010

MODAL PARTICIPATION FACTORS

MODE	FREQ(CPS)	X-DIRECTION	Y-DIRECTION	Z-DIRECTION
1	9.360	.1256E+00	-.2492E-01	-.2471E+00
2	12.706	.1676E+00	.5631E+00	-.1069E+00
3	15.377	.8715E+00	-.6408E-01	-.4225E+00
4	17.797	-.1401E-01	-.1551E+00	.6173E-01
5	21.603	.8529E-02	-.9633E-01	-.6805E-01
6	25.098	-.1571E+00	.9049E-01	.3075E+00
7	32.035	.1034E+00	.1588E+00	-.2016E+00
8	38.069	.1295E-01	-.2087E+00	-.4261E-01
9	40.293	.1855E+00	-.2674E-01	-.2483E+00
10	48.898	-.5327E-01	-.4115E-01	.8967E-01
11	57.515	.2302E-01	-.6021E-01	-.2156E-01
12	61.500	-.1309E-01	-.2067E+00	.1830E-01
13	62.541	.1020E-02	.1684E+00	.6823E-02
14	69.348	-.3686E-01	.7659E-01	.4170E-01
15	77.444	-.1819E-01	-.3966E-01	.2977E-01
16	78.881	-.4225E-01	-.1408E-01	.5378E-01
17	101.715	.2351E+00	-.2577E-01	-.1365E+00
18	103.583	.2999E-01	.2357E+00	-.1471E-01
19	107.966	-.1885E-01	.7470E-01	.6679E-02
20	115.098	.8568E-02	.1936E-01	-.2065E-02
21	135.244	-.1203E-01	.6157E-02	-.8529E-02
22	155.220	-.8902E-02	-.1286E-02	.4626E-02
23	160.601	.8302E-02	.1443E-02	-.2239E-02
24	203.780	.1083E+00	.1302E-03	.2001E+00
25	209.925	-.1092E-01	.1313E-02	-.3002E-01

SPECTRUM TABLE (X SPECTRUM GROUP 4   )

NUMBER OF POINTS = 13
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.1260E+00
2	.5550E-01	.1260E+00
3	.6730E-01	.1400E+00
4	.1136E+00	.2150E+00
5	.1515E+00	.3100E+00
6	.1852E+00	.3100E+00
7	.3704E+00	.2650E+00
8	.4914E+00	.3350E+00
9	.7576E+00	.5360E+00
10	.1070E+01	8500E+00
11	.1307E+01	.8500E+00
12	.1587E+01	.8140E+00
13	.2500E+01	.2700E+00

SPECTRUM TABLE (Y AND Z SPECTRUM GROUP 4   )

NUMBER OF POINTS = 15
SCALE FACTOR = .10000E+01

INPUT POINT	PERIOD	SPECTRUM VALUE
1	.2560E-01	.2100E+00
2	.3570E-01	.2100E+00
3	.5410E-01	.2600E+00
4	.6730E-01	.3600E+00
5	.1033E+00	.4200E+00
6	.1212E+00	.7300E+00
7	.1515E+00	.1020E+01
8	.1852E+00	.1020E+01
9	.2315E+00	.8500E+00
10	.4914E+00	6800E+00
11	.6006E+00	.6800E+00
12	7273E+00	.8500E+00
13	.1070E+01	.1200E+01
14	.1307E+01	.1200E+01
15	.2500E+01	3400E+00

CLUSTERING. +1 MEANS MODE I CLOSE TO I+1. -1, NOT.
-1. -1 -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1. -1 -1 -1 -1 0.

COMBINATION OF INDIVIDUAL SUPPORT GROUP RESPONSES

KIND=1 SUPPORT COMBINATION AFTER SAP SUMMATION

KIND≠1 SUPPORT COMBINATION DONE FIRST

GROUP COMBINATION TYPE= 3

ALGEBRAIC METHOD = 1
ABSOLUTE METHOD = 2
SRSS METHOD = 3

NODE DISPLACEMENTS/ROTATIONS

NODE NUMBER	NODE NUMBER	X-TRANSLATION	Y-TRANSLATION	Z-TRANSLATION	X-ROTATION	Y-ROTATION	Z-ROTATION	ACCELERATIONS IN G'S X-DIRECTION	Y-DIRECTION	Z-DIRECTION
1	TOTAL..	5.30472E-09	4.61580E-09	1.12862E-08	0.	0.	0.			
2	TOTAL..	6.36066E-03	1.96320E-05	1.58008E-02	5.04243E-04	2.87333E-04	2.02770E-04			
3	TOTAL..	1.97235E-02	3.90762E-05	4.87824E-02	6.49739E-04	5.74667E-04	2.73621E-04			
4	TOTAL..	2.66822E-02	3.12475E-03	5.73824E-02	5.69370E-04	7.59410E-04	2.99381E-04			
5	TOTAL..	2.94735E-02	1.09077E-02	3.98934E-02	4.68800E-04	1.03441E-03	3.07665E-04			
6	TOTAL..	2.94712E-02	1.62432E-02	2.06823E-02	4.31734E-04	1.11796E-03	2.99254E-04			
7	TOTAL.	2.94673E-02	2.13453E-02	4.40790E-06	3.95583E-04	1.19820E-03	2.82800E-04			
8	TOTAL..	1.69767E-02	2.41779E-02	3.10970E-02	3.47226E-04	1.04460E-03	2.30415E-04			
9	TOTAL..	2.56882E-06	1.94097E-02	3.98157E-02	2.74651E-04	2.55930E-04	2.11065E-04			
10	TOTAL..	2.92126E-03	1.51914E-02	3.98760E-02	2.34744E-04	1.31652E-04	2.09375E-04			
11	TOTAL...	5.13698E-03	1.23203E-02	3.99338E-02	1.79170E-04	2.63864E-04	2.17231E-04			
12	TOTAL...	1.03063E-02	1.10035E-02	3.99890E-02	1.23606E-04	3.56425E-04	2.33672E-04			
13	TOTAL..	1.77740E-02	9.82730E-03	4.00241E-02	1.09503E-04	4.43116E-04	2.32375E-04			
14	TOTAL..	5.30455E-02	5.30834E-03	4.01248E-02	1.08168E-04	5.35917E-04	3.20913E-04			
15	TOTAL..	5.65540E-02	1.04505E-04	4.03324E-02	1.73797E-04	4.82379E-04	4.55409E-04			
16	TOTAL..	3.01157E-02	5.21802E-05	2.54754E-02	3.07215E-04	4.82379E-04	5.26124E-04			
17	TOTAL..	3.20487E-09	1.23706E-08	8.84878E-09	5.08406E-04	4.82379E-04	5.72329E-04			

18 TOTAL..	1.03394E-02	1.87419E-02	2.67095E-02	1.21591E-04	3.63864E-04	1 37770E-04
19 TOTAL..	1.03416E-02	1.95108E-02	1.11438E-02	2.05499E-04	3.97830E-04	7.84569E-05
20 TOTAL..	3.28690E-03	5.66850E-03	3.74595E-05	2.23175E-04	1.30004E-04	9.22975E-05
21 TOTAL..	1.03230E-08	1.13989E-08	1.15548E-08	0.	0.	0
22 TOTAL..	0.	0.	0.	0.	0.	0.
23 TOTAL..	0.	0.	0.	0.	0.	0.
24 TOTAL..	0.	0.	0.	0.	0.	0.
25 TOTAL..	0.	0.	0.	0.	0	0
26 TOTAL..	0.	0.	0.	0.	0	0.
27 TOTAL.	0.	0.	0.	0.	0.	0
28 TOTAL..	0.	0.	0.	0.	0.	0
29 TOTAL..	0.	0.	0	0.	0.	0
30 TOTAL .	0	0.	0	0.	0.	0
31 TOTAL..	0	0.	0.	0.	0.	0
32 TOTAL..	0.	0	0.	0.	0.	0
33 TOTAL .	0	0.	0.	0.	0	0
34 TOTAL..	0.	0.	0	0.	0.	0
35 TOTAL	0	0	0	0.	0	0

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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (1)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	4	1	10	1	1	1	4	1	10	1	1	1
X MAXIMUM	1.164E+01	1.755E+01	9.784E+00	5.539E+02	6.780E+02	1.745E+03	1.164E+01	1.755E+01	9.784E+00	5.539E+02	3.332E+02	7.802E+02
Y	2	1	2	1	1	1	2	1	2	1	1	1
Y MAXIMUM	2.165E+01	3.202E+01	1.710E+01	1.011E+03	1.237E+03	3.184E+03	2.165E+01	3.202E+01	1.710E+01	1.011E+03	6.081E+02	1.440E+03
Z	1	1	1	1	1	1	1	1	1	1	1	1
Z MAXIMUM	1.925E+01	9.297E+01	3.355E+01	2.934E+03	-3.592E+03	-9.243E+03	1.925E+01	9.297E+01	3.355E+01	2.934E+03	1.765E+03	4.181E+03
GRAND TOTAL	4.616E+01	1.129E+02	5.305E+01	3.230E+03	4.209E+03	1.034E+04	4.616E+01	1.129E+02	5.305E+01	3.230E+03	2.005E+03	4.520E+03

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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (2)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	4	1	4	1	1	1	4	1	4	1	4	5
X MAXIMUM	1.162E+01	1.469E+01	5.861E+00	5.539E+02	3.332E+02	7.892E+02	1.162E+01	1.469E+01	5.861E+00	5.539E+02	3.569E+02	1.819E+02
Y	2	1	2	1	1	1	2	1	2	1	2	5
Y MAXIMUM	2.163E+01	2.680E+01	1.527E+01	1.011E+03	6.081E+02	1.440E+03	2.163E+01	2.680E+01	1.527E+01	1.011E+03	9.721E+02	2.293E+02
Z	1	1	1	1	1	1	1	1	1	1	4	6
Z MAXIMUM	1.924E+01	7.781E+01	2.757E+01	2.934E+03	1.765E+03	4.181E+03	1.924E+01	7.781E+01	2.757E+01	2.934E+03	4.189E+02	8.697E+02
GRAND TOTAL	4.574E+01	8.762E+01	3.900E+01	3.230E+03	2.005E+03	4.529E+03	4.574E+01	8.762E+01	3.900E+01	3.230E+03	1.406E+03	1.165E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (3)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	4	10	1	1	5	4	4	10	1	1	1	4
X MAXIMUM	1.159E+01	7.617E+00	7.872E+00	5.539E+02	1.819E+02	3.569E+02	1.201E+01	8.263E+00	7.872E+00	4.859E+02	3.310E+02	3.721E+02
Y	2	2	1	1	5	2	2	1	1	1	1	2
Y MAXIMUM	2.161E+01	1.310E+01	1.437E+01	1.011E+03	2.293E+02	9.721E+02	2.501E+01	6.665E+00	1.437E+01	8.868E+02	6.040E+0[illegible]	[illegible]
Z	1	1	1	1	6	4	4	1	1	1		
Z MAXIMUM	1.922E+01	1.298E+01	4.171E+01	2.934E+03	8.697E+02	4.189E+02	1.410E+01	1.935E+01	4.171E+01	2.575E+03	1.754E+[illegible]	
GRAND TOTAL	4.519E+01	2.858E+01	5.154E+01	3.230E+03	1.165E+03	1.406E+03	4.568E+01	2.778E+01	5.154E+01	2.792E+03	2.139E+[illegible]	

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (3)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	4	10	1	1	1	4
X MAXIMUM	1.060E+01	7.650E+00	7.872E+00	3.005E+02	6.011E+02	3.209E+02
Y	2	4	1	1	1	2
Y MAXIMUM	2.460E+01	9.940E+00	1.437E+01	5.485E+02	1.097E+03	1.081E+03
Z	4	1	1	1	1	4
Z MAXIMUM	1.244E+01	2.277E+01	4.171E+01	1.592E+03	3.185E+03	3.768E+02
GRAND TOTAL	4.170E+01	3.335E+01	5.154E+01	1.746E+03	3.502E+03	1.397E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (4)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(	[illegible]
X	4	10	5	1	1	4	9	4	5	4		4
X MAXIMUM	9.820E+00	7.285E+00	8.818E+00	3.005E+02	6.011E+02	3.209E+02	9.036E+00	9.333E+00	8.818E+00	1.304E+02	7.027E	02
Y	2	4	5	1	1	2	2	4	5	2		2
Y MAXIMUM	2.404E+01	1.040E+01	1.112E+01	5.485E+02	1.0[illegible]E+03	1.081E+03	2.013E+01	1.601E+01	1.112E+01	2.612E+02	1 2 1	02
	6	1	6	1	1	4	6	1	6	1		1

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ELEMENT TYPE (3/D   P I P E                ) / / /  ELEMENT NUMBER (    4)
                 PX(J)      VY(J)      VZ(J)      TX(J)      MY(J)      MZ(J)
X                    9          4          5          1          1         10
X MAXIMUM    9.051E+00  1.113E+01  8.818E+00  2.373E+02  6.974E+02  1.930E+02
Y                    2          2          5          1          1          2
Y MAXIMUM    1.316E+01  2.083E+01  1.112E+01  4.331E+02  1.273E+03  6.862E+02
Z                    6          1          6          1          1          1
Z MAXIMUM    1.312E+01  1.769E+01  4.147E+01  1.257E+03  3.695E+03  7.198E+02
GRAND TOTAL  3.120E+01  4.244E+01  5.384E+01  1.429E+03  4.075E+03  1.120E+03
ELEMENT TYPE (3/D   P I P E                ) / / /  ELEMENT NUMBER (    5)
                 PX(I)      VY(I)      VZ(I)      TX(I)      MY(I)      MZ(I)      PX(J)      VY(J)      VZ(J)      TX(J)      MY(J)      MZ(J)
X                    9          4          5          1          1         10          9          4          5          1          1          1
X MAXIMUM    1.179E+01  9.093E+00  1.118E+01  2.373E+02  6.974E+02  1.930E+02  1.179E+01  9.093E+00  1.118E+01  2.373E+02  6.827E+02  1.819E+02
Y                    2          2          5          1          1          2          2          2          5          1          1          2
Y MAXIMUM    1.424E+01  1.725E+01  1.410E+01  4.331E+02  1.273E+03  6.862E+02  1.424E+01  1.725E+01  1.410E+01  4.331E+02  1.246E+03  3.784E+02
Z                    6          1          6          1          1          1          6          1          6          1          1          1
Z MAXIMUM    1.936E+01  1.367E+01  4.963E+01  1.257E+03  3.695E+03  7.198E+02  1.936E+01  1.367E+01  4.963E+01  1.257E+03  3.617E+03  9.638E+02
GRAND TOTAL  4.314E+01  3.355E+01  6.421E+01  1.429E+03  4.075E+03  1.120E+03  4.314E+01  3.355E+01  6.421E+01  1.429E+03  4.381E+03  1.210E+03
ELEMENT TYPE (3/D   P I P E                ) / / /  ELEMENT NUMBER (    6)
                 PX(I)      VY(I)      VZ(I)      TX(I)      MY(I)      MZ(I)      PX(J)      VY(J)      VZ(J)      TX(J)      MY(J)      MZ(J)
X                    9          4          5          1          1          1          9          4          5          1          1          4
X MAXIMUM    1.388E+01  6.504E+00  1.190E+01  2.373E+02  6.827E+02  1.819E+02  1.388E+01  6.504E+00  1.198E+01  2.373E+02  6.456E+02  2.169E+02
Y                    2          2          5          1          1          2          2          2          5          1          1          1
Y MAXIMUM    1.507E+01  1.248E+01  1.510E+01  4.331E+02  1.246E+03  3.784E+02  1.507E+01  1.248E+01  1.510E+01  4.331E+02  1.178E+03  3.891E+02
Z                    6          1          6          1          1          1          6          1          6          1          1          1
Z MAXIMUM    2.416E+01  9.284E+00  5.193E+01  1.257E+03  3.617E+03  9.638E+02  2.416E+01  9.284E+00  5.193E+01  1.257E+03  3.420E+03  1.130E+03
GRAND TOTAL  5.489E+01  2.428E+01  6.925E+01  1.429E+03  4.381E+03  1.210E+03  5.489E+01  2.428E+01  6.925E+01  1.429E+03  4.884E+03  1.390E+03
ELEMENT TYPE (3/D   P I P E                ) / / /  ELEMENT NUMBER (    7)
                 PX(I)      VY(I)      VZ(I)      TX(I)      MY(I)      MZ(I)      PX(C)      VY(C)      VZ(C)      TX(C)      MY(C)      MZ(C)
X                    9          1         10          1          4          1          1          1         10          1          4          1
X MAXIMUM    1.659E+01  7.245E+01  4.841E+00  2.373E+02  2.169E+02  6.456E+02  2.165E+01  6.945E+01  4.841E+00  3.019E+02  2.032E+02  3.790E+02
Y                    2          1          2          1          1          1          1          1          2          1          4          1
Y MAXIMUM    1.615E+01  1.322E+02  3.564E+00  4.331E+02  3.891E+02  1.178E+03  3.951E+01  1.267E+02  3.564E+00  5.510E+02  4.858E+02  6.916E+02
Z                    1          1         15          1          1          1          1          1         15          1          1          1
Z MAXIMUM    3.484E+01  3.838E+02  7.236E+00  1.257E+03  1.130E+03  3.420E+03  1.147E+02  3.679E+02  7.236E+00  1.600E+03  5.912E+02  2.008E+03
GRAND TOTAL  7.150E+01  4.299E+02  1.510E+01  1.429E+03  1.390E+03  4.884E+03  1.432E+02  4.116E+02  1.510E+01  1.761E+03  9.591E+02  2.790E+03
ELEMENT TYPE (3/D   P I P E                ) / / /  ELEMENT NUMBER (    7)
                 PX(J)      VY(J)      VZ(J)      TX(J)      MY(J)      MZ(J)
X                    1          1         10          1          4          1
X MAXIMUM    4.658E+01  5.588E+01  4.841E+00  3.227E+02  3.064E+02  1.284E+03
Y                    1          1          2          1          4          1
Y MAXIMUM    8.500E+01  1.020E+02  3.564E+00  5.889E+02  5.256E+02  2.343E+03
Z                    1          1         15          1          4          1
Z MAXIMUM    2.168E+02  2.960E+02  7.236E+00  1.710E+03  3.596E+02  6.802E+03
GRAND TOTAL  2.820E+02  3.323E+02  1.510E+01  1.884E+03  7.638E+02  7.379E+03
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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (8)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	1	1	4	1	4	1	1	1	4	1	4	1
X MAXIMUM	4.328E+01	5.489E+01	4.834E+00	3.227E+02	3.064E+02	1.284E+03	6.099E+01	3.415E+01	4.834E+00	2.919E+02	1.896E+02	1.927E+03
Y	1	1	2	1	4	1	1	1	2	1	4	1
Y MAXIMUM	7.898E+01	1.002E+02	1.163E+01	5.889E+02	5.256E+02	2.343E+03	1.113E+02	6.232E+01	1.163E+01	5.327E+02	3.253E+02	3.516E+03
Z	1	1	1	1	4	1	1	1	1	1	1	1
Z MAXIMUM	2.293E+02	2.908E+02	6.871E+00	1.710E+03	3.596E+02	6.802E+03	3.231E+02	1.809E+02	6.871E+00	1.546E+03	7.841E+02	1.021E+04
GRAND TOTAL	2.650E+02	3.272E+02	2.282E+01	1.884E+03	7.638E+02	7.379E+03	3.643E+02	2.112E+02	2.282E+01	1.748E+03	9.808E+02	1.10

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (8)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	16	4	4	1	1
X MAXIMUM	6.941E+01	1.982E+01	4.834E+00	2.138E+02	2.664E+02	2.233E+03
Y	1	2	2	1	1	1
Y MAXIMUM	1.267E+02	1.714E+01	1.163E+01	3.822E+02	4.862E+02	4.075E+03
Z	1	1	1	1	1	1
Z MAXIMUM	3.678E+02	4.351E+01	6.871E+00	1.110E+03	1.412E+03	1.183E+04
GRAND TOTAL	4.116E+02	8.901E+01	2.282E+01	1.346E+03	1.569E+03	1.288E+04

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (9)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	4	3	4	1	1	1	4	3	4	1	
X MAXIMUM	6.608E+01	9.132E+00	3.302E+01	2.138E+02	2.233E+03	2.664E+02	6.608E+01	9.132E+00	3.302E+01	2.138E+02	1.624E+03	3.1..
Y	1	2	1	1	1	1	1	2	1	1	1	
Y MAXIMUM	1.206E+02	2.463E+01	5.418E+01	3.822E+02	4.075E+03	4.862E+02	1.206E+02	2.463E+01	5.418E+01	3.822E+02	2.964E+03	5.692E+02
Z	1	1	1	1	1	1	1	1	1	1	1	1
Z MAXIMUM	3.501E+02	1.171E+01	1.573E+02	1.110E+03	1.183E+04	1.412E+03	3.501E+02	1.174E+01	1.573E+02	1.110E+03	8.605E+03	1.652E+03
GRAND TOTAL	3.920E+02	3.686E+01	1.948E+02	1.346E+03	1.288E+04	1.569E+03	3.920E+02	3.686E+01	1.948E+02	1.346E+03	9.290E+03	1.940E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (10)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	4	3	4	1	1	1	4	3	4	1	4
X MAXIMUM	6.328E+01	1.143E+01	3.164E+01	2.138E+02	1.624E+03	3.119E+02	6.328E+01	1.143E+01	3.164E+01	2.138E+02	1.013E+03	3.775E+02
Y	1	2	1	1	1	1	1	2	1	1	1	2
Y MAXIMUM	1.155E+02	3.425E+01	5.436E+01	3.822E+02	2.964E+03	5.692E+02	1.155E+02	3.425E+01	5.436E+01	3.822E+02	1.850E+03	1.206E+03
Z	1	1	1	1	1	1	1	1	1	1	1	1
Z MAXIMUM	3.353E+02	1.415E+01	1.578E+02	1.110E+03	8.605E+03	1.652E+03	3.353E+02	1.415E+01	1.578E+02	1.110E+03	5.369E+03	1.942E+03
GRAND TOTAL	3.757E+02	4.864E+01	1.941E+02	1.346E+03	9.290E+03	1.940E+03	3.757E+02	4.864E+01	1.941E+02	1.346E+03	6.181E+03	2.639E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (11)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	2	1	4	1	4	1	2	1	4	3	4
X MAXIMUM	6.018E+01	7.616E+00	2.965E+01	2.138E+02	1.013E+03	3.775E+02	6.048E+01	7.616E+00	2.965E+01	2.138E+02	1.503E+03	4.778E+02
Y	1	2	1	1	1	2	1	2	1	1	3	4
Y MAXIMUM	1.104E+02	7.339E+01	5.412E+01	3.822E+02	1.850E+03	1.206E+03	1.104E+02	7.339E+01	5.412E+01	3.822E+02	1.270E+03	8.197E+02
Z	1	1	1	1	1	1	1	1	1	1	[illegible]	1
Z [illegible]IMUM	3.204E+02	1.204E+01	1.571E+02	1.110E+03	5.369E+0[illegible]	[illegible]42E+03	3.204E+02	1.204E+01	1.571E+02	1.110E+03	2.591E+0[illegible]	[illegible]96E+03

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ELEMENT TYPE (3/D   P I P E   ) / / /   ELEMENT NUMBER (12)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	2	1	1	1	4	1	2	1	1	3	4
X MAXIMUM	3.912E+01	8.508E+00	1.090E+01	2.721E+02	6.947E+02	4.554E+02	3.912E+01	8.508E+00	1.090E+01	2.721E+02	5.465E+02	3.547E+02
Y	1	2	1	2	1	2	1	2	1	2	1	4
Y MAXIMUM	7.138E+01	8.199E+01	1.989E+01	5.743E+02	1.268E+03	2.036E+03	7.138E+01	8.199E+01	1.989E+01	5.743E+02	8.599E+02	6.086E+02
Z	1	6	1	1	1	1	1	6	1	1	1	1
Z MAXIMUM	2.072E+02	1.890E+01	5.775E+01	1.442E+03	3.680E+03	1.505E+03	2.072E+02	1.890E+01	5.775E+01	1.442E+03	2.497E+03	1.195E+03
GRAND TOTAL	2.401E+02	8.904E+01	6.867E+01	1.727E+03	4.241E+03	2.982E+03	2.401E+02	8.904E+01	6.867E+01	1.727E+03	2.971E+03	1.670E+03

ELEMENT TYPE (3/D   P I P E   ) / / /   ELEMENT NUMBER (13)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	5	1	1	3	4	1	5	1	1	3	5
X MAXIMUM	3.289E+01	4.648E+00	8.312E+00	2.721E+02	5.465E+02	3.547E+02	3.289E+01	4.648E+00	8.312E+00	2.721E+02	3.370E+02	2.410E+02
Y	1	5	1	2	1	4	1	5	1	2	7	5
Y MAXIMUM	6.002E+01	5.859E+00	1.517E+01	5.743E+02	8.599E+02	6.086E+02	6.002E+01	5.859E+00	1.517E+01	5.743E+02	3.274E+02	3.039E+02
Z	1	6	1	1	1	1	1	6	1	1	6	6
Z MAXIMUM	1.742E+02	2.562E+01	4.404E+01	1.442E+03	2.497E+03	1.195E+03	1.742E+02	2.562E+01	4.404E+01	1.442E+03	6.181E+02	1.518E+03
GRAND TOTAL	2.110E+02	3.414E+01	5.339E+01	1.727E+03	2.971E+03	1.670E+03	2.110E+02	3.414E+01	5.339E+01	1.727E+03	1.341E+03	2.066E+03

ELEMENT TYPE (3/D   P I P E   ) / / /   ELEMENT NUMBER (14)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	1	21	1	1	1	5	1	1	3	1	1	1
X MAXIMUM	2.414E+01	[illegible]	[illegible]	[illegible]	[illegible]	2.410E+02	1.603E+01	1.811E+01	9.283E+00	1.449E+02	1.924E+02	1.463E+02
Y	1	2	7	2	7	5	1	1	7	2	2	2
Y MAXIMUM	4.405E+01	[illegible]	9.019E+00	5.743E+02	[illegible]	3.039E+02	2.926E+01	3.304E+01	9.019E+00	4.716E+02	1.061E+02	3.851E+02
Z	1	21	6	1	6	6	1	1	6	1	1	1
Z MAXIMUM	1.279E+02	2.511E+01	1.704E+01	1.442E+03	6.184E+02	1.548E+03	8.495E+01	9.593E+01	1.704E+01	7.676E+02	1.020E+03	7.748E+02
GRAND TOTAL	1.763E+02	4.094E+01	3.694E+01	1.727E+03	1.341E+03	2.066E+03	1.266E+02	1.294E+02	3.694E+01	9.827E+02	1.221E+03	

ELEMENT TYPE (3/D   P I P E   ) / / /   ELEMENT NUMBER (14)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	21	1	3	11	3	1
X MAXIMUM	7.628E+00	2.414E+01	9.283E+00	8.929E-11	2.335E+02	7.815E+02
Y	2	1	7	11	2	1
Y MAXIMUM	1.112E+01	4.405E+01	9.019E+00	1.313E-10	7.326E+02	1.426E+03
Z	21	1	6	1	1	1
Z MAXIMUM	2.541E+01	1.279E+02	1.704E+01	3.152E-10	8.335E+02	4.140E+03
GRAND TOTAL	4.094E+01	1.763E+02	3.694E+01	4.088E-10	1.390E+03	4.988E+03

ELEMENT TYPE (3/D   P I P E   ) / / /   ELEMENT NUMBER (15)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	
X	1	11	11	0	3	1	1	11	11	0	7	
X MAXIMUM	1.673E+01	7.688E+00	5.322E+00	0.	2.335E+02	7.815E+02	1.673E+01	7.688E+00	5.322E+00	0.	2.934E+02	[illegible]
Y	1	11	11	0	2	1	1	11	11	0	1	1
Y MAXIMUM	3.053E+01	1.129E+01	7.814E+00	0.	7.326E+02	1.426E+03	3.053E+01	1.129E+01	7.814E+00	0.	4.186E+02	9.400E+02
Z	1	1	11	0	1	1	1	1	11	0	1	1
Z MAXIMUM	8.864E+01	2.589E+01	9.619E+00	0.	8.335E+02	4.140E+03	8.864E+01	2.589E+01	9.619E+00	0.	1.215E+03	2.729E+03
GRAND TOTAL	1.232E+02	3.715E+01	2.390E+01	0.	1.390E+03	4.988E+03	1.232E+02	3.715E+01	2.390E+01	0.	1.731E+03	3.590E+03

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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (16)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	1	7	0	7	1	1	1	7	0	1	1
X MAXIMUM	1.674E+01	9.539E+00	5.434E+00	0.	2.934E+02	5.151E+02	1.674E+01	9.539E+00	5.434E+00	0.	1.455E-11	3.274E-11
Y	1	1	1	0	1	1	1	1	1	0	1	1
Y MAXIMUM	3.055E+01	1.741E+01	7.752E+00	0.	4.186E+02	9.400E+02	3.055E+01	1.741E+01	7.752E+00	0	4.120E-11	6.518E-11
Z	1	1	1	0	1	1	1	1	1	0	5	1
Z MAXIMUM	8.868E+01	5.054E+01	2.251E+01	0	1.215E+03	2.729E+03	8.868E+01	5.054E+01	2.251E+01	0	7.514E-12	2.391E-10
GRAND TOTAL	1.237E+02	6.649E+01	3.205E+01	0.	1.731E+03	3.590E+03	1.237E+02	6.649E+01	3.205E+01	0.	4.813E-11	2.532E-10

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (17)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	2	9	2	3	4	3	2	9	2	9	2
X MAXIMUM	1.762E+01	1.450E+01	2.304E+01	1.803E+02	1.056E+03	2.450E+02	1.762E+01	1.450E+01	2.304E+01	1.803E+02	4.361E+02	5.011E+02
Y	1	2	1	2	3	2	1	2	1	2	1	2
Y MAXIMUM	3.210E+01	1.398E+02	2.889E+01	1.738E+03	8.923E+02	7.620E+02	3.210E+01	1.398E+02	2.889E+01	1.738E+03	6.279E+02	4.829E+03
Z	1	6	6	6	6	6	1	6	6	6	6	6
Z MAXIMUM	9.319E+01	2.621E+01	1.124E+02	2.978E+02	1.886E+03	5.646E+02	9.319E+01	2.621E+01	1.124E+02	2.978E+02	2.612E+03	4.840E+02
GRAND TOTAL	1.352E+02	1.476E+02	1.550E+02	1.809E+03	3.400E+03	1.264E+03	1.352E+02	1.476E+02	1.550E+02	1.809E+03	3.551E+03	4.975E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (18)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	2	3	2	9	2	3	2	3	2	3	2
X MAXIMUM	3.270E+01	6.769E+00	2.317E+01	1.803E+02	4.361E+02	5.011E+02	3.270E+01	6.769E+00	2.317E+01	1.803E+02	5.289E+02	2.304E+02
Y	3	2	1	2	1	2	3	2	1	2	3	2
Y MAXIMUM	2.763E+01	6.523E+01	2.215E+01	1.738E+03	6.279E+02	4.829E+03	2.763E+01	6.523E+01	2.215E+01	1.738E+03	4.470E+02	2.220E+03
Z	1	2	1	6	6	6	1	2	1	6	1	6
Z MAXIMUM	4.593E+01	5.581E+00	6.430E+01	2.978E+02	2.612E+03	4.840E+02	4.593E+01	5.581E+00	6.430E+01	2.978E+02	7.490E+02	4.471E+02
GRAND TOTAL	9.106E+01	6.777E+01	9.897E+01	1.809E+03	3.551E+03	4.975E+03	9.106E+01	6.777E+01	9.897E+01	1.809E+03	1.485E+03	2.371E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (19)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	3	3	2	2	2	3	3	3	2	4	7	3
X MAXIMUM	4.057E+01	2.164E+01	1.019E+01	1.803E+02	2.304E+02	5.289E+02	4.399E+01	1.338E+01	1.019E+01	7.631E+01	5.715E+01	6.531E[illegible]
Y	3	1	2	2	2	3	3	3	2	2	2	[illegible]
Y MAXIMUM	3.428E+01	2.543E+01	9.815E+01	1.738E+03	2.220E+03	4.470E+02	3.717E+01	1.131E+01	9.815E+01	7.024E+02	2.790E+02	[illegible]
Z	3	1	6	6	6	1	1	1	6	1	[illegible]	[illegible]
Z MAXIMUM	5.096E+01	7.383E+01	1.353E+01	2.978E+02	4.471E+02	7.490E+02	7.943E+01	2.498E+01	1.353E+01	2.635E+02	1.795E+0[illegible]	[illegible]
GRAND TOTAL	9.583E+01	1.153E+02	1.014E+02	1.809E+03	2.371E+03	1.485E+03	1.428E+02	4.579E+01	1.014E+02	7.877E+02	3.902E+0[illegible]	[illegible]

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (19)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	3	2	2	2	1
X MAXIMUM	2.164E+01	4.057E+01	1.019E+01	1.394E+02	1.894E+02	3.834E+02
Y	1	3	2	2	2	1
Y MAXIMUM	2.543E+01	3.428E+01	9.815E+01	1.343E+03	1.825E+03	6.997E+02
Z	1	3	6	2	1	1
Z MAXIMUM	7.383E+01	5.096E+01	1.353E+01	1.149E+02	4.368E+02	2.031E+03
GRAND TOTAL	1.153E+02	9.583E+01	1.014E+02	1.361E+03	1.960E+03	2.302E+03

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ELEMENT TYPE (3/D PIPE) / / / ELEMENT NUMBER (20)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	2	3	2	1	2	3	2	3	2	3	2
X MAXIMUM	2.166E+01	1.117E+01	4.468E+01	1.394E+02	3.834E+02	1.894E+02	2.166E+01	1.117E+01	4.468E+01	1.394E+02	2.012E+03	6.531E+02
Y	1	2	3	2	1	2	1	2	3	2	3	2
Y MAXIMUM	2.544E+01	1.077E+02	3.775E+01	1.343E+03	6.997E+02	1.825E+03	2.544E+01	1.077E+02	3.775E+01	1.343E+03	1.700E+03	6.294E+03
Z	1	6	3	2	1	1	1	6	3	2	3	6
Z MAXIMUM	7.385E+01	1.916E+01	5.612E+01	1.149E+02	2.031E+03	4.368E+02	7.385E+01	1.916E+01	5.612E+01	1.149E+02	2.528E+03	9.885E+02
GRAND TOTAL	1.155E+02	1.140E+02	1.032E+02	1.361E+03	2.302E+03	1.960E+03	1.155E+02	1.140E+02	1.032E+02	1.361E+03	4.038E+03	6.632E+03

S U M M A R Y   O F   S U P P O R T   F O R C E S/M O M E N T S   I N   G L O B A L / L O C A L   S Y S T E M S .

BDRY= BOUNDARY ELEMENTS (SPRING)
THAM= THERMAL ANCHOR MOTION
SLAM= SEISMIC ANCHOR MOTION
SNBR= SNUBBER ELEMENT

L O A D   C A S E   1   BENCHMARK PROBLEM 2 SRSS

KIND OF SUPPORT	NODE NUMBER	GLOBAL COMPONENTS FX	FY	FZ	MX	MY	MZ	LOCAL COMPONENTS FL	ML
BDRY	1	53	0.	0.	0.	0.	0.	53	0
BDRY	1	0	46.	0.	0.	0.	0.	46	0.
BDRY	1	0	0	113	0	0	0	113	0
BDRY	7	0	0	441	0.	0	0	441	0
BDRY	9	257	0	0.	0.	0	0.	257.	0.
BDRY	11	0	123.	0	0.	0.	0	123.	0
BDRY	13	0	98.	0	0.	0	0.	98	0
BDRY	15	111	111	156	0.	0	0.	221.	0
BDRY	17	32.	0	0.	0.	0.	0	32.	0
BDRY	17	0.	124	0.	0.	0	0	124.	0
BDRY	17	0	0	66.	0.	0.	0	66	0
BDRY	21	103.	0.	0.	0.	0	0	103	0
BDRY	21	0	114	0.	0.	0	0.	114	0.
BDRY	21	0.	0	116.	0.	0	0	116	0

Problem 2c

Independent Support Motion Solution
Absolute Group Combination

C O M B I N A T I O N   O F   I N D I V I D U A L   S U P P O R T   G R O U P   R E S P O N S E S

KIND=1 SUPPORT COMBINATION AFTER SAP SUMMATION

KIND≠1 SUPPORT COMBINATION DONE FIRST

GROUP COMBINATION TYPE= 2

ALGEBRAIC METHOD = 1
ABSOLUTE METHOD = 2
SRSS METHOD = 3

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NODE DISPLACEMENTS/ROTATIONS

NODE NUMBER	MODE NUMBER	X-TRANSLATION	Y-TRANSLATION	Z-TRANSLATION	X-ROTATION	Y-ROTATION	Z-ROTATION	ACCELERATIONS IN G'S X-DIRECTION	Y-DIRECTION	Z-DIRECTION
1	TOTAL	7 63867E-09	6.96148E-09	1 55641E-08	0.	0.	0.			
2	TOTAL	8 96306E-03	2 96087E-05	2 19922E-02	7.02403E-04	4.00084E-04	2 84679E-04			
3	TOTAL	2.76147E-02	5 89965E-05	6 79826E-02	9.06483E-04	8.00168E-04	3 81986E-04			
4	TOTAL	3 72577E-02	4 41149E-03	8.00211E-02	7 95017E-04	1 05747E-03	4 26117E-04			
5	TOTAL	4 11418E-02	1.56555E-02	5 56616E-02	6.55607E-04	1.44250E-03	4 50539E-04			
6	TOTAL	4.11385E-02	2 34876E-02	2 88626E-02	6 04802E-04	1 55982E-03	4 40676E-04			
7	TOTAL	4 11332E-02	3 10171E-02	6.07336E-06	5 55635E-04	1 67234E-03	4 18065E-04			
8	TOTAL	2 37136E-02	3 55177E-02	4 33952E-02	4 92257E-04	1.45729E-03	3 42225E-04			
9	TOTAL	3 50486E-06	2 88350E-02	5.55612E-02	3.98413E-04	3.63557E-04	3 14483E-04			
10	TOTAL	4 32290E-03	2.26767E-02	5 56452E-02	3.43869E-04	1.95385E-04	3 12662E-04			
11	TOTAL	7 65364E-03	1 84326E-02	5.57256E-02	2.64261E-04	3.71741E-04	3 24081E-04			
12	TOTAL	1 47958E-02	1.64498E-02	5.58024E-02	1.82269E-04	4.97312E-04	3 47434E-04			
13	TOTAL	2 50916E-02	1 46223E-02	5.58514E-02	1.60111E-04	6.17795E-04	3 42604E-04			
14	TOTAL	7 40647E-02	7.35698E-03	5.59924E-02	1.59900E-04	7.48912E-04	4 54469E-04			
15	TOTAL	7 89124E-02	1 42987E-04	5.62781E-02	2.37752E-04	6.76133E-04	6 36275E-04			
16	TOTAL	4 20217E-02	7.13021E-05	3.54936E-02	5.54149E-04	6.76133E-04	7 34129E-04			
17	TOTAL	4 52006E-09	1.69039E-08	9.08940E-09	7.05058E-04	6.76133E-04	7 98620E-04			

18 TOTAL .	1.48413E-02	2.81037E-02	3.71955E-02	1 81587E-04	5.08445E-04	2 02965E-04
19 TOTAL..	1.48410E-02	2 92205E-02	1.54898E-02	3.07858E-04	5 53901E-04	1 12972E-04
20 TOTAL..	4.93285E-03	8.46971E-03	5.10914E-05	3.33611E-04	1.92768E-04	1.38352E-04
21 TOTAL	1.51682E-08	1.69874E-08	1.57597E-08	0	0.	0
22 TOTAL..	0	0.	0.	0.	0.	0
23 TOTAL..	0	0	0.	0.	0.	0
24 TOTAL.	0	0.	0.	0.	0.	0.
25 TOTAL..	0	0	0	0	0.	0.
26 TOTAL	0	0	0	0.	0.	0.
27 TOTAL.	0	0	0	0.	0.	0
28 TOTAL	0	0	0	0	0.	0
29 TOTAL	0	0	0	0	0	0
30 TOTAL	0.	0	0.	0	0	0
31 TOTAL	0.	0	0	0	0.	0
32 TOTAL .	0	0	0.	0	0.	0.
33 TOTAL..	0	0	0	0	0.	0
34 TOTAL .	0	0.	0.	0	0	0
35 TOTAL.	0.	0	0.	0	0	0

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (1)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	4	1	10	1	1	1	4	1	10	1	1	1
X MAXIMUM	1.248E+01	3.282E+01	1.685E+01	1.036E+03	1.268E+03	3.263E+03	1.248E+01	3.282E+01	1.685E+01	1.036E+03	6.232E+02	1.476E+03
Y	4	1	2	1	1	1	4	1	2	1	1	1
Y MAXIMUM	3.896E+01	4.970E+01	2.561E+01	1.569E+03	1.920E+03	4.941E+03	3.896E+01	4.970E+01	2.561E+01	1.569E+03	9.436E+02	2.235E+03
Z	1	1	1	1	1	1	1	1	1	1	1	1
Z MAXIMUM	2.609E+01	1.260E+02	4.548E+01	3.978E+03	4.869E+03	1.253E+04	2.609E+01	1.260E+02	4.548E+01	3.978E+03	2.393E+03	5.668E+03
GRAND TOTAL	6.961E+01	1.556E+02	7.639E+01	4.498E+03	5.959E+03	1.438E+04	6.961E+01	1.556E+02	7.639E+01	4.498E+03	2.787E+03	6.317E+03

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (2) 16202

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	4	1	1	1	1	1	4	1	1	1	4	5
X MAXIMUM	1.246E+01	2.747E+01	9.733E+00	1.036E+03	6.232E+02	1.476E+03	1.246E+01	2.747E+01	9.733E+00	1.036E+03	3.826E+02	2.714E+02
Y	4	1	2	1	1	1	4	1	2	1	2	4
Y MAXIMUM	3.890E+01	4.159E+01	2.287E+01	1.569E+03	9.436E+02	2.235E+03	3.890E+01	4.159E+01	2.287E+01	1.569E+03	1.456E+03	4.355E+02
Z	1	1	1	1	1	1	1	1	1	1	4	6
Z MAXIMUM	2.608E+01	1.055E+02	3.737E+01	3.978E+03	2.393E+03	5.668E+03	2.608E+01	1.055E+02	3.737E+01	3.978E+03	6.167E+02	1.045E+03
GRAND TOTAL	6.912E+01	1.217E+02	5.605E+01	4.498E+03	2.787E+03	6.317E+03	6.912E+01	1.217E+02	5.605E+01	4.498E+03	2.194E+03	1.542E+03

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (3)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	4	10	1	1	5	4	4	10	1	1	1	4
X MAXIMUM	1.243E+01	1.312E+01	1.472E+01	1.036E+03	2.714E+02	3.826E+02	1.288E+01	1.423E+01	1.472E+01	9.089E+02	6.191E+02	3.989E+02
Y	4	2	1	1	4	2	4	1	1	1	1	2
Y MAXIMUM	3.880E+01	1.975E+01	2.229E+01	1.569E+03	4.355E+02	1.456E+03	4.020E+01	1.034E+01	2.229E+01	1.376E+03	9.374E+02	1.641E+03
Z	1	1	1	1	6	4	4	1	1	1	1	4
Z MAXIMUM	2.606E+01	1.760E+01	5.654E+01	3.978E+03	1.045E+03	6.167E+02	2.076E+01	2.623E+01	5.654E+01	3.490E+03	2.377E+03	6.430E+02
GRAND TOTAL	6.848E+01	4.165E+01	7.057E+01	4.498E+03	1.542E+03	2.194E+03	6.965E+01	3.967E+01	7.057E+01	3.897E+03	2.963E+03	2 [illegible]

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (3)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	4	10	1	1	1	4
X MAXIMUM	1.137E+01	1.317E+01	1.472E+01	5.621E+02	1.124E+03	3.440E+02
Y	2	4	1	1	1	2
Y MAXIMUM	3.685E+01	1.939E+01	2.229E+01	8.511E+02	1.702E+03	1.619E+03
Z	4	1	1	1	1	4
Z MAXIMUM	1.832E+01	3.087E+01	5.654E+01	2.158E+03	4.317E+03	5.545E+02
GRAND TOTAL	6.365E+01	4.871E+01	7.057E+01	2.449E+03	4.886E+03	2.162E+03

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (4)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	HY(C)	
X	9	10	5	1	1	4	9	4	5	4	1	
X MAXIMUM	1.331E+01	1.254E+01	1.316E+01	5.621E+02	1.124E+03	3.440E+02	1.573E+01	1.001E+01	1.316E+01	1.398E+02	1.314E+0:	
Y	2	4	5	1	1	2	2	4	5	4		
Y MAXIMUM	3.600E+01	2.016E+01	1.910E+01	8.511E+02	1.702E+03	1.619E+03	3.015E+01	3.124E+01	1.910E+01	4.365E+02	1.990E+0	
Z	1	1	6	1	1	4	6	1	6	1	1	1
Z MAXIMUM	1.761E+01	1.627E+01	4.984E+01	2.158E+03	4.317E+03	5.545E+02	1.780E+01	2.178E+01	4.984E+01	2.958E+02	5.047E+03	6.453E+02
GRAND TOTAL	6.673E+01	4.169E+01	6.976E+01	2.449E+03	4.886E+03	2.162E+03	5.738E+01	5.384E+01	6.976E+01	7.722E+02	5.598E+03	1.856E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (4)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	9	4	5	1	1	10
X MAXIMUM	1.575E+01	1.194E+01	1.316E+01	4.439E+02	1.304E+03	3.323E+02
Y	2	4	5	1	1	2
Y MAXIMUM	1.972E+01	3.726E+01	1.910E+01	6.721E+02	1.975E+03	1.028E+03
Z	6	1	6	1	1	1
Z MAXIMUM	1.577E+01	2.398E+01	4.984E+01	1.704E+03	5.008E+03	9.757E+02
GRAND TOTAL	4.490E+01	6.461E+01	6.976E+01	2.026E+03	5.661E+03	1.626E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (5)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	9	4	5	1	1	10	9	4	5	1	1	1
X MAXIMUM	2.051E+01	9.749E+00	1.669E+01	4.439E+02	1.304E+03	3.323E+02	2.051E+01	9.749E+00	1.669E+01	4.439E+02	1.277E+03	3.403E+02
Y	2	4	5	1	1	2	2	4	5	1	1	2
Y MAXIMUM	2.133E+01	3.044E+01	2.423E+01	6.721E+02	1.975E+03	1.028E+03	2.133E+01	3.044E+01	2.423E+01	6.721E+02	1.933E+03	5.668E+02
Z	6	1	6	1	1	1	6	1	6	1	1	1
Z MAXIMUM	2.327E+01	1.854E+01	5.964E+01	1.704E+03	5.008E+03	9.757E+02	2.327E+01	1.854E+01	5.964E+01	1.704E+03	4.903E+03	1.307E+03
GRAND TOTAL	6.016E+01	5.153E+01	8.312E+01	2.026E+03	5.661E+03	1.626E+03	6.016E+01	5.153E+01	8.312E+01	2.026E+03	6.009E+03	1.715E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (6)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	9	4	5	1	1	1	9	4	5	1	1	1
X MAXIMUM	2.416E+01	6.974E+00	1.787E+01	4.439E+02	1.277E+03	3.403E+02	2.416E+01	6.974E+00	1.787E+01	4.439E+02	1.208E+03	3.988E+02
Y	2	4	5	1	1	2	2	4	5	1	1	4
Y MAXIMUM	2.258E+01	2.177E+01	2.594E+01	6.721E+02	1.933E+03	5.668E+02	2.258E+01	2.177E+01	2.594E+01	6.721E+02	1.828E+03	7.259E+02
Z	1	1	6	1	1	1	1	1	6	1	1	1
Z MAXIMUM	3.053E+01	1.259E+01	6.242E+01	1.704E+03	4.903E+03	1.307E+03	3.053E+01	1.259E+01	6.242E+01	1.704E+03	4.637E+03	1.531E+03
GRAND TOTAL	7.550E+01	3.749E+01	8.989E+01	2.026E+03	6.009E+03	1.715E+03	7.550E+01	3.749E+01	8.989E+01	2.026E+03	6.588E+03	1.991E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (7)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	9	1	10	1	1	1	1	1	10	1	4	1
X MAXIMUM	2.887E+01	1.355E+02	8.336E+00	4.439E+02	3.988E+02	1.208E+03	4.050E+01	1.299E+02	8.336E+00	5.647E+02	3.036E+02	7.089E+02
Y	2	1	4	1	4	1	1	1	4	1	4	1
Y MAXIMUM	2.419E+01	2.052E+02	6.305E+00	6.721E+02	7.259E+02	1.828E+03	6.131E+01	1.967E+02	6.305E+00	8.550E+02	9.478E+02	1.073E+03
Z	1	1	15	1	1	1	1	1	15	1	1	1
Z MAXIMUM	4.722E+01	5.203E+02	8.010E+00	1.704E+03	1.531E+03	4.637E+03	1.555E+02	4.988E+02	8.010E+00	2.168E+03	8.014E+02	2.722E+03
GRAND TOTAL	9.742E+01	5.967E+02	2.143E+01	2.026E+03	1.991E+03	6.588E+03	1.992E+02	5.709E+02	2.143E+01	2.463E+03	1.473E+03	3.825E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (7)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	1	10	1	4	1
X MAXIMUM	8.712E+01	1.045E+02	8.336E+00	6.036E+02	3.285E+02	2.402E+03
Y	1	1	4	1	4	1
Y MAXIMUM	1.319E+02	1.582E+02	6.305E+00	9.139E+02	1.025E+03	3.636E+03
Z	1	1	15	1	4	1
XIMUM	3.345E+02	4.013E+02	8.010E+00	2.318E+03	5.294E+0[illegible]	[illegible]221E+03

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ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (8)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	1	1	4	1	4	1	1	1	4	1	1	1
X MAXIMUM	8.095E+01	1.027E+02	5.183E+00	6.036E+02	3.285E+02	2.402E+03	1.141E+02	6.388E+01	5.183E+00	5.460E+02	2.768E+02	3.601E+03
Y	1	1	2	1	4	1	1	1	2	1	4	1
Y MAXIMUM	1.226E+02	1.554E+02	1.742E+01	9.139E+02	1.025E+03	3.636E+03	1.727E+02	9.671E+01	1.742E+01	8.266E+02	6.346E+02	5.457E+03
Z	1	1	1	1	4	1	1	1	1	1	1	1
Z MAXIMUM	3.108E+02	3.942E+02	9.314E+00	2.318E+03	5.294E+02	9.221E+03	4.380E+02	2.453E+02	9.314E+00	2.096E+03	1.063E+03	1.384E+01
GRAND TOTAL	3.686E+02	4.532E+02	3.377E+01	2.634E+03	1.268E+03	1.031E+04	5.065E+02	2.011E+02	3.377E+01	2.466E+03	1.434E+03	1

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (8)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	9	4	1	1	1
X MAXIMUM	1.298E+02	3.357E+01	5.183E+00	3.918E+02	4.984E+02	4.176E+03
Y	1	2	2	4	1	1
Y MAXIMUM	1.966E+02	2.560E+01	1.742E+01	7.157E+02	7.546E+02	6.323E+03
Z	1	1	1	1	1	1
Z MAXIMUM	4.985E+02	5.898E+01	9.314E+00	1.504E+03	1.914E+03	1.604E+04
GRAND TOTAL	5.717E+02	1.200E+02	3.377E+01	1.937E+03	2.187E+03	1.794E+04

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (9)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	
X	1	4	1	1	1	1	1	4	1	1	1	
X MAXIMUM	1.236E+02	9.791E+00	5.553E+01	3.918E+02	4.176E+03	4.984E+02	1.236E+02	9.791E+00	5.553E+01	3.918E+02	3.038E+03	
Y	1	2	1	4	1	1	1	2	1	4	1	
Y MAXIMUM	1.871E+02	3.689E+01	8.407E+01	7.157E+02	6.323E+03	7.546E+02	1.871E+02	3.689E+01	8.407E+01	7.157E+02	4.600E+03	[illegible]
Z	1	1	1	1	1	1	1	1	1	1	1	
Z MAXIMUM	4.746E+02	1.592E+01	2.132E+02	1.504E+03	1.604E+04	1.914E+03	4.746E+02	1.592E+01	2.132E+02	1.504E+03	1.166E+04	2.240E+03
GRAND TOTAL	5.449E+02	5.686E+01	2.697E+02	1.937E+03	1.794E+04	2.187E+03	5.449E+02	5.686E+01	2.697E+02	1.937E+03	1.297E+04	2.735E+03

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (10)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	4	1	1	1	1	1	4	1	1	1	1
X MAXIMUM	1.184E+02	1.225E+01	5.572E+01	3.918E+02	3.038E+03	5.834E+02	1.184E+02	1.225E+01	5.572E+01	3.918E+02	1.896E+03	6.858E+02
Y	1	2	1	4	1	1	1	2	1	4	1	2
Y MAXIMUM	1.792E+02	5.130E+01	8.436E+01	7.157E+02	4.600E+03	8.832E+02	1.792E+02	5.130E+01	8.436E+01	7.157E+02	2.870E+03	1.806E+03
Z	1	4	1	1	1	1	1	4	1	1	1	1
Z MAXIMUM	4.545E+02	1.974E+01	2.140E+02	1.504E+03	1.166E+04	2.240E+03	4.545E+02	1.974E+01	2.140E+02	1.504E+03	7.279E+03	2.633E+03
GRAND TOTAL	5.226E+02	7.465E+01	2.686E+02	1.937E+03	1.297E+04	2.735E+03	5.226E+02	7.465E+01	2.686E+02	1.937E+03	8.651E+03	3.830E+03

ELEMENT TYPE (3/D PIPE) /// ELEMENT NUMBER (11)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	2	1	1	1	1	1	2	1	1	3	1
X MAXIMUM	1.131E+02	1.335E+01	5.547E+01	3.918E+02	1.896E+03	6.858E+02	1.131E+02	1.335E+01	5.547E+01	3.918E+02	2.462E+03	5.986E+02
Y	1	2	1	4	1	2	1	2	1	4	3	4
Y MAXIMUM	1.713E+02	1.099E+02	8.398E+01	7.157E+02	2.870E+03	1.806E+03	1.713E+02	1.099E+02	8.398E+01	7.157E+02	1.957E+03	1.599E+03
Z	1	1	1	1	1	1	1	1	1	1	5	1
Z MAXIMUM	4.314E+02	1.632E+01	2.130E+02	1.504E+03	7.279E+03	2.633E+03	4.344E+02	1.632E+01	2.130E+02	1.504E+03	3.114E+03	2.299E+03
GRAND TOTAL	5.001E+02	1.162E+02	2.642E+02	1.937E+03	8.651E+03	3.830E+03	5.004E+02	1.162E+02	2.642E+02	1.937E+03	6.357E+03	3.365E+03

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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (12)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	2	1	1	1	1	1	2	1	1	3	1
X MAXIMUM	7.317E+01	1.491E+01	2.039E+01	5.090E+02	1.299E+03	5.312E+02	7.317E+01	1.491E+01	2.039E+01	5.090E+02	8.919E+02	4.218E+02
Y	1	2	1	2	1	2	1	2	1	2	1	4
Y MAXIMUM	1.108E+02	1.228E+02	3.087E+01	8.602E+02	1.967E+03	3.050E+03	1.108E+02	1.228E+02	3.087E+01	8.602E+02	1.334E+03	1.187E+03
Z	1	6	1	1	1	1	1	6	1	1	1	1
Z MAXIMUM	2.809E+02	2.271E+01	7.829E+01	1.954E+03	4.989E+03	2.040E+03	2.809E+02	2.271E+01	7.829E+01	1.954E+03	3.384E+03	1.620E+03
GRAND TOTAL	3.327E+02	1.322E+02	9.564E+01	2.424E+03	5.948E+03	4.412E+03	3.327E+02	1.322E+02	9.564E+01	2.424E+03	4.200E+03	2.462E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (13)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	5	1	1	3	1	1	5	1	1	3	5
X MAXIMUM	6.152E+01	6.935E+00	1.555E+01	5.090E+02	8.949E+02	4.218E+02	6.152E+01	6.935E+00	1.555E+01	5.090E+02	5.517E+02	3.597E+02
Y	1	4	1	2	1	4	1	4	1	2	7	5
Y MAXIMUM	9.314E+01	1.054E+01	2.354E+01	8.602E+02	1.334E+03	1.187E+03	9.314E+01	1.054E+01	2.354E+01	8.602E+02	4.412E+02	5.221E+02
Z	1	6	1	1	1	1	1	6	1	1	1	6
Z MAXIMUM	2.362E+02	3.079E+01	5.970E+01	1.954E+03	3.384E+03	1.620E+03	2.362E+02	3.079E+01	5.970E+01	1.954E+03	8.245E+02	1.861E+03
GRAND TOTAL	2.905E+02	4.488E+01	7.450E+01	2.424E+03	4.200E+03	2.462E+03	2.905E+02	4.488E+01	7.450E+01	2.424E+03	1.878E+03	2.700E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (14)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	1	21	3	1	3	5	1	1	3	1	1	1
X MAXIMUM	4.515E+01	9.107E+00	1.520E+01	5.090E+02	5.517E+02	3.597E+02	2.999E+01	3.387E+01	1.520E+01	2.710E+02	3.599E+02	2.736E+02
Y	1	2	7	2	7	5	1	1	7	2	2	2
Y MAXIMUM	6.836E+01	1.666E+01	1.216E+01	8.602E+02	4.412E+02	5.221E+02	4.541E+01	5.128E+01	1.216E+01	7.064E+02	6.082E+02	5.772E+02
Z	1	21	1	1	1	6	1	1	1	1	1	
Z MAXIMUM	1.734E+02	2.631E+01	2.271E+01	1.954E+03	8.245E+02	1.861E+03	1.152E+02	1.300E+02	2.271E+01	1.040E+03	1.382E+[illegible]	[illegible]
GRAND TOTAL	2.388E+02	5.288E+01	5.174E+01	2.424E+03	1.878E+03	2.700E+03	1.692E+02	1.765E+02	5.171E+01	1.400E+03	1.714E+[illegible]	[illegible]

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (14)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	21	1	3	1	3	1
X MAXIMUM	9.107E+00	4.515E+01	1.520E+01	1.273E-10	3.823E+02	1.462E+03
Y	2	1	7	11	2	1
Y MAXIMUM	1.666E+01	6.836E+01	1.216E+01	2.064E-10	1.097E+03	2.213E+03
Z	21	1	1	1	1	1
Z MAXIMUM	2.631E+01	1.734E+02	2.271E+01	4.320E-10	1.130E+03	5.612E+03
GRAND TOTAL	5.288E+01	2.388E+02	5.174E+01	5.664E-10	1.999E+03	6.898E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (15)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J	MZ(J)
X	1	11	14	0	3	1	1	11	14	0		
X MAXIMUM	3.129E+01	1.003E+01	6.954E+00	0.	3.823E+02	1.462E+03	3.129E+01	1.003E+01	6.954E+00	0	5.096E+[illegible]	[illegible]
Y	1	11	1	0	2	1	1	11	11	0	1	1
Y MAXIMUM	4.738E+01	1.775E+01	1.229E+01	0.	1.097E+03	2.213E+03	4.738E+01	1.775E+01	1.229E+01	0	6.496E+02	1.459E+03
Z	1	1	11	0	1	1	1	1	11	0	1	1
Z MAXIMUM	1.202E+02	3.510E+01	1.107E+01	0	1.130E+03	5.612E+03	1.202E+02	3.510E+01	1.107E+01	0	1.647E+[illegible]	1.699E+03
GRAND TOTAL	1.684E+02	5.078E+01	3.262E+01	0	1.999E+[illegible]	3.898E+03	1.684E+02	5.078E+01	3.262E+01	0	2.411[illegible]	[illegible]

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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (16)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	1	9	0	9	1	1	1	9	0	1	1
X MAXIMUM	3.131E+01	1.784E+01	9.437E+00	0.	5.096E+02	9.634E+02	3.131E+01	1.784E+01	9.437E+00	0.	1.155E-11	4.729E-11
Y	1	1	1	0	1	1	1	1	1	0	1	1
Y MAXIMUM	4.740E+01	2.701E+01	1.203E+01	0.	6.496E+02	1.459E+03	4.740E+01	2.701E+01	1.203E+01	0.	6.003E-11	9.095E-11
Z	1	1	1	0	1	1	1	1	1	0	5	1
Z MAXIMUM	1.202E+02	6.851E+01	3.051E+01	0.	1.647E+03	3.699E+03	1.202E+02	6.851E+01	3.051E+01	0.	9.550E-12	3.201E-10
GRAND TOTAL	1.690E+02	9.089E+01	4.520E+01	0.	2.441E+03	4.908E+03	1.690E+02	9.089E+01	4.520E+01	0.	6.696E-11	3.405E-10

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (17)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	1	2	9	2	3	4	1	2	9	2	9	2
X MAXIMUM	3.290E+01	2.542E+01	4.010E+01	3.160E+02	1.729E+03	2.627E+02	3.290E+01	2.542E+01	4.010E+01	3.160E+02	7.591E+02	8.781E+02
Y	1	2	1	2	3	2	1	2	1	2	1	2
Y MAXIMUM	4.981E+01	2.094E+02	4.483E+01	2.603E+03	1.375E+03	1.141E+03	4.981E+01	2.094E+02	4.483E+01	2.603E+03	9.741E+02	7.233E+03
Z	1	6	6	6	6	6	1	6	6	6	6	1
Z MAXIMUM	1.263E+02	3.150E+01	1.351E+02	3.579E+02	2.267E+03	6.784E+02	1.263E+02	3.150E+01	1.351E+02	3.579E+02	3.139E+03	6.355E+02
GRAND TOTAL	1.820E+02	2.206E+02	2.045E+02	2.696E+03	4.761E+03	1.862E+03	1.820E+02	2.206E+02	2.045E+02	2.696E+03	4.677E+03	7.446E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (18)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	2	3	2	9	2	3	2	3	2	3	2
X MAXIMUM	5.354E+01	1.186E+01	3.793E+01	3.160E+02	7.591E+02	8.781E+02	5.354E+01	1.186E+01	3.793E+01	3.160E+02	8.660E+02	4.037E+02
Y	3	2	1	2	1	2	3	2	1	2	3	2
Y MAXIMUM	4.258E+01	9.770E+01	3.437E+01	2.603E+03	9.744E+02	7.233E+03	4.258E+01	9.770E+01	3.137E+01	2.603E+03	6.886E+02	3.325E+03
Z	1	2	1	6	6	1	1	2	1	6	1	6
Z MAXIMUM	6.226E+01	7.458E+00	8.716E+01	3.579E+02	3.139E+03	6.355E+02	6.226E+01	7.458E+00	8.716E+01	3.579E+02	1.015E+03	5.374E+02
GRAND TOTAL	1.303E+02	1.014E+02	1.370E+02	2.696E+03	4.677E+03	7.446E+03	1.303E+02	1.014E+02	1.370E+02	2.696E+03	2.118E+03	3.532E+03

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (19)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(C)	VY(C)	VZ(C)	TX(C)	MY(C)	MZ(C)
X	3	3	2	2	2	3	3	3	2	2	7	3
X MAXIMUM	6.643E+01	3.544E+01	1.785E+01	3.160E+02	4.037E+02	8.660E+02	7.203E+01	2.191E+01	1.785E+01	1.277E+02	9.247E+01	1.069E+0[illegible]
Y	3	1	2	2	2	3	3	3	2	2	[illegible]	
Y MAXIMUM	5.282E+01	3.946E+01	1.470E+02	2.603E+03	3.325E+03	6.886E+02	5.728E+01	1.743E+01	1.470E+02	1.052E+03	4.179E+0[illegible]	
Z	3	1	6	6	6	1	1	1	6	1	[illegible]	
Z MAXIMUM	7.286E+01	1.001E+02	1.626E+01	3.579E+02	5.374E+02	1.015E+03	1.077E+02	3.386E+01	1.626E+01	3.572E+02	2.157E+0[illegible]	
GRAND TOTAL	1.404E+02	1.573E+02	1.516E+02	2.696E+03	3.532E+03	2.118E+03	2.010E+02	6.370E+01	1.516E+02	1.179E+03	5.663E+0[illegible]	

ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (19)

	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	3	2	2	2	1
X MAXIMUM	3.544E+01	6.643E+01	1.785E+01	2.442E+02	3.319E+02	7.172E+02
Y	1	3	2	2	2	1
Y MAXIMUM	3.946E+01	5.282E+01	1.470E+02	2.012E+03	2.734E+03	1.086E+03
Z	1	3	6	2	1	1
Z MAXIMUM	1.001E+02	7.286E+01	1.626E+01	1.536E+02	5.921E+02	2.754E+03
GRAND TOTAL	1.573E+02	1.401E+02	1.516E+02	2.041E+03	2.935E+03	3.192E+03

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ELEMENT TYPE (3/D P I P E) / / / ELEMENT NUMBER (20)

	PX(I)	VY(I)	VZ(I)	TX(I)	MY(I)	MZ(I)	PX(J)	VY(J)	VZ(J)	TX(J)	MY(J)	MZ(J)
X	3	2	3	2	1	2	3	2	3	2	3	2
X MAXIMUM	3.547E+01	1.958E+01	7.315E+01	2.442E+02	7.172E+02	3 319E+02	3.547E+01	1 958E+01	7 315E+01	2 442E+02	3 295E+03	1.145E+03
Y	1	2	3	2	1	2	1	2	3	2	3	2
Y MAXIMUM	3.948E+01	1.613E+02	5.817E+01	2.012E+03	1.086E+03	2.734E+03	3.948E+01	1.613E+02	5.817E+01	2 012E+03	2 620E+03	9 427E+03
Z	1	6	3	2	1	1	1	6	3	2	3	6
Z MAXIMUM	1.001E+02	2.303E+01	8.024E+01	1.536E+02	2.754E+03	5.921E+02	1.001E+02	2.303E+01	8 024E+01	1 536E+02	3 614E+03	1.108E+03
GRAND TOTAL	1.576E+02	1.699E+02	1.517E+02	2.041E+03	3.192E+03	2.935E+03	1.576E+02	1 699E+02	1.517E+02	2.041E+03	6 079E+03	9.904E+03

SUMMARY OF SUPPORT FORCES/MOMENTS IN GLOBAL/LOCAL SYSTEMS

BDRY= BOUNDARY ELEMENTS (SPRING)
THAM= THERMAL ANCHOR MOTION
SEAM= SEISMIC ANCHOR MOTION
SNBR= SNUBBER ELEMENT

LOAD CASE 1 BENCHMARK PROBLEM 2 ABSOLUTE

KIND OF SUPPORT	NODE NUMBER	GLOBAL COMPONENTS FX	FY	FZ	MX	MY	MZ	LOCAL COMPONENTS FL	ML
BDRY	1	76	0.	0.	0.	0	0	76.	0.
BDRY	1	0.	70.	0.	0.	0	0.	70	0
BDRY	1	0.	0	156.	0.	0	0	156.	0.
BDRY	7	0.	0	607	0.	0.	0.	607	0.
BDRY	9	350	0.	0	0.	0	0	350.	0
BDRY	11	0	184	0.	0.	0	0	184.	0.
BDRY	13	0	146	0	0.	0	0.	146.	0.
BDRY	15	151	151	212	0	0	0	301.	0.
BDRY	17	45	0	0	0	0	0	45.	0.
BDRY	17	0	169	0	0	0	0	169.	0
BDRY	17	0	0	91	0	0	0	91	0.
BDRY	21	152	0	0	0	0	0	152.	0
BDRY	21	0	170	0	0	0	0	170.	0
BDRY	21	0	0	158	0	0	0	158.	0

ATTACHMENT 3

LETTER NUMBER 2.02.096

Response to RAI 9B

Double-click Icon to view Attachment

ATTACH3 PDF

GE NUCLEAR ENERGY
Structural Mechanics and Materials
175 Curtner Avenue, San Jose, CA 95125

October 9, 2002
dkh0213

cc: D. J. Robare
M. K. Kaul
R. M. Horn

Mr. Fred Mogolesko
Entergy Nuclear Generating Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: Pilgrim Nuclear Power Station, Unit 1 – Piping Analysis Methodology for ISM with SRSS Grouping

Dear Mr. Mogolesko,

This letter report provides the justification for the use of Independent Support Motion (ISM) and the Square-Root-of-Sum-of-Squares (SRSS) Grouping method of analysis, in conjunction with the use of Regulatory Guide 1.61 damping values, for piping systems at Pilgrim Nuclear Power Station, Unit 1. This letter report, supporting documents, analyses results and other details of the present evaluation are documented in the Reference 4.1 GE Nuclear Energy Design Record File (DRF). The requirements of this report are set forth in References 4.2 and 4.3.

This report includes a Background discussion as well as a description of the Technical Basis for the ISM and SRSS Grouping Methodology in conjunction with the utilization of Regulatory Guide 1.61 Damping. This report also includes a technical discussion, based on the seismic analysis of Pilgrim main steam piping Loop A, which quantifies the contribution of the piping modes below 8 Hz to the piping total faulted excitation response.

1.0 BACKGROUND

The piping analyses conducted in 2002 for the Pilgrim Nuclear Power Station, Thermal Power Optimization Program (TPO) are based on the ISM with SRSS grouping combination, response spectrum analysis methodology using Regulatory Guide 1.61 damping. The details of the analytical methodology are described in Section 3.0 of Reference 4.4. The piping analytical methodology is essentially the same as that recommended and utilized by GE Nuclear Energy in the Pilgrim Recirculation Replacement Piping Analysis in the early

1980's, References 4.10, 4.11 and 4.12. A number of the pertinent details contained in these references are provided in Appendix C. In addition, Appendix C contains other historical material related to the application of Regulatory Guide 1.61 damping in GE BWR NSSS piping analysis.

The OBE and SSE seismic input motion spectra for the analysis were taken from Reference 4.5. The piping structural damping was the same as for Regulatory Guide 1.61 and the peak collinear response contributions from each ISM group were combined by the SRSS methodology.

In Reference 4.9, Item No. 9.B., of the U.S. NRC/NRR Request for Additional Information (RAI) associated with the Pilgrim Nuclear Power Station, Unit 1, TPO Program, the staff stated that *"The ISM analysis used the square root of sum of squares (SRSS) approach for combining Group responses. This approach does not correspond to the staff position as stated in NUREG 1061, Volume 4. Likewise, the use of damping values in the analysis is not in accordance with the licensing basis for Pilgrim* (Reference 4.7).

B. Provide the justification for using the RG 1.61 damping values instead of the licensing basis damping values, as shown in the FSAR, for application with the ISM approach."

The technical basis/justification for the ISM with SRSS combination of the grouping peak, collinear response contributions in conjunction with Regulatory Guide 1.61 damping is provided in Section 3.0 below.

2.0 CONCLUSIONS

It is concluded, consistent with the Reference 4.4 analysis, that ISM with SRSS Grouping and Regulatory Guide 1.61 damping be used as the standard analysis procedure for piping systems at Pilgrim.

Experimental tests conducted in the past, Reference 4.6, conclusively demonstrate that damping values for piping systems in BWR plants is higher than that recommended in Regulatory Guide 1.61. The use of Regulatory Guide 1.61 damping values as proposed in Pilgrim Unit 1 Specification document, Reference 4.4, therefore, maintains conservatism and is more realistic than that presented in Table 12.2-3 of Reference 4.8.

The ISM with SRSS Grouping methodology, in contrast to the "Center-of-Gravity" uniform base motion approach used in the "1979 as-built" analyses, is realistic and technically tenable and should be the basis for all piping system analyses.

It is also concluded that the Pilgrim main steam piping modes, with modal frequencies 8 Hz or below, typically contribute over 94% to the total faulted load case piping response. The only exceptions are for several piping supports; i.e., struts and snubbers. However for those exceptions in which the modes at 8 Hz or below cumulatively contribute less than 94% to the total piping response, the total support loads, due to all modes, are less than 70% of the support capability.

The details of the present evaluation, as well as evidence of verification, are documented in the Reference 4.1 GE Nuclear Energy DRF.

3.0 TECHNICAL BASIS FOR THE RECOMMENDED METHODOLOGY

Regulatory Guide 1.61 Damping. In October 1982, GE issued a Design Memo, Reference 4.6, which reported the conclusions of a full-scale test of main steam piping under in-plant conditions conducted in 1978-79 at Wyle Labs, Huntsville, Alabama. The piping system was extensively instrumented to yield responses from multiple sensor locations thus corroborating the results obtained. The purpose of the tests was to assess the damping values associated with the seismic/dynamic response of piping systems under in-plant conditions.

The report was an outcome of the work of the Steering Committee on Piping Systems of the Pressure Vessel Research Committee (PVRC) of the ASME. The Steering Committee was formed to address, among other things, the issue of conservatism in the use of damping values of piping systems in operation at that point in time. It included a survey of the damping values for piping systems available from the considerable accumulation of literature that had been published after the NRC had issued Regulatory Guide 1.61. The literature search showed that values reported by researchers ranged from 2% to 10% with higher excitation levels resulting in higher damping. Additionally, the purpose of the report was *"to present experimental damping data obtained from General Electric BWR piping systems for ASME Service Level B conditions"*. Higher damping values would be expected for higher Service Levels, thus, damping values greater than those derived from the tests could be justified.

Under in-plant conditions the energy loss in piping systems results not only from material viscous damping but also from material non-linear effects such as cyclical hysteresis losses, and system losses from slippage between contacting parts and rotations at joints. The test set-up was designed to simulate these in-plant conditions as closely as possible to enable the tests to capture all sources of energy loss and determine the values of the *effective* viscous damping.

The fundamental conclusion of the test study was *"that NRC Regulatory Guide 1.61 specified limits for damping for both OBE and SSE events are much too conservative."* The GE test data indicated that the damping *"was at least 5 per cent for measured stresses considerably less than the Service Levels A and B limits."*

In view of the conclusions of this report, it would seem appropriate to use at least 5% damping for the piping systems analysis. However, since the input ground motion to the Pilgrim 1 primary structure model does not reflect the conservatism of Regulatory Guide 1.60 recommended ground motion, adopting a much lower damping as set forth in Regulatory

Guide 1.61 will more than offset the loss of conservatism associated with the selection of ground motion.

Moreover, reverting to the damping values used in the "1979 as-built" analysis of Pilgrim 1 Recirculation and Branch Piping (tabulated in Table 12.2-3 of Reference 4.8) is not technically justified, since it unnecessarily adds to the already conservative damping values of Regulatory Guide 1.61 that were used in the 1984 analyses. The use of Regulatory Guide 1.61 damping ensures enough conservatism in the analysis of piping systems and any departure from the values in the Guide should be towards increased damping values rather than towards lower values.

In summary, it should be emphasized that the effective damping in piping systems under OBE and SSE conditions is significantly higher than what is recommended by various regulatory guidelines and what is used in the general analysis practice. Damping values prescribed by Regulatory Guide 1.61 are, therefore, still on the conservative side and may be employed without any technical concern for under-prediction of actual piping responses.

ISM with SRSS Grouping. The recirculation piping is supported at multiple locations on various plant structures such as the RPV (Group 1), Reactor Building and Drywell (Group 2), and Shield Wall and Pedestal (Group 3). These structural groups generally have distinct associated vibration modes whose frequencies are spaced sufficiently far apart. The consequence of this frequency spacing is that the piping support motions from these structural groups are, in a statistical sense, uncorrelated. This fact alone provides sufficient justification for the SRSS combination of the piping responses due to each support group motion.

An additional justification, heuristic in nature, comes from the fact that peak acceleration values of the piping support excitations from each of these structural groups generally occur at different times spaced far enough apart so that the resulting peak responses in the piping systems due to each of these support group input motions are very unlikely to be time consistent and hence additive. The following example illustrates this.

In a previous analysis of Pilgrim primary structure model (model data from DRF No. B11-00617-01, GE Report No. GENE 771-65-1094, Revision 2, March 1995) with fuel modification (use of GE14 fuel), the maximum accelerations at recirculation piping support point groups were obtained. The model used in that analysis is shown in Appendix B, Figure B.1. The primary frequencies of various structural groups are listed in Table B.1. Table B.2 records the times at which peak acceleration responses occur in different structural groups. The following discussion forms the basis for selecting Table B.2 node points on these structural groups.

The RPV support elevation for the various piping systems is 86.94 feet. Node 16 selected in Table B.2 corresponds to this elevation. Reactor Building and Drywell support locations are at 21.7 feet, 23 feet and 51.1 feet. The peak responses and response spectra at these nodes are enveloped by the node with the highest elevation, i.e., Node 7 at elevation 51 feet. Similarly, Shield Wall and Pedestal support locations are at elevations 35.4 feet and 47.4 feet with envelopes of peak responses and response spectra obtained at Node 13, Reference 4.5.

Table B.2 lists the times at which peak acceleration responses are obtained at each of the nodes thus selected on the three structural groups. The minimum spacing between these peak response times occurs between the RPV and the Reactor building responses and is 0.135 seconds.

A few simplifying assumption would help understand how piping responses from excitation at the support points would combine to lead to a maximum total response. Firstly, assume that in the vicinity of their peak response times, each of the three structural groups vibrate at a single frequency associated with these groups, i.e., 6.844 Hz. for RPV, 12.265 Hz. for the Reactor Building and Drywell, and 15.353 Hz. for the Shield Wall and Pedestal. Assume further that the peak response of the supported piping due to each of the separate inputs from these structural groups is spaced similarly in time as the peak responses of the input driving motion.

With these simplifying assumptions, it is obvious that the peak responses from individual support motions would be additive if the time spacing between the peaks is zero. At any non-zero spacing, the combined peak response will be less than the absolute-sum value. Using the analogy of closely spaced modes (Regulatory Guide 1.92) if the time spacing of peak responses from individual support group excitations is less than one-tenth of the largest excitation period, the individual responses should be absolute-summed otherwise combined by SRSS method.

In the preceding representative example data, the longest excitation period is associated with the RPV with a frequency of 6.844 Hz. This corresponds to a period of 0.1461 seconds. One-tenth of this period is 0.0146 seconds, which is much less than the spacing of 0.135 seconds between RPV and Reactor Building response peaks. Thus, SRSS would appropriately characterize the combination method in this case.

<u>Cumulative Contribution of Main Steam Piping Modes Below 8 Hz.</u> The results of the study conducted by Entergy in conjunction the Reference 4.4 piping analysis are summarized in Table A-1 of Appendix A of this letter report. A sketch of the piping model, corresponding to main stream piping Loop A, analyzed in the study, is provided in Figure A1 of Appendix A. Piping responses were calculated for two load cases: (i) SSE only, and (ii) for the critical faulted load case. For each load case, the piping responses were calculated twice, once based on the modes with modal frequencies up through 8 Hz and the second time based on the modes with frequencies up through 100 Hz.

From Table A-1, it is seen that the Pilgrim main steam piping Loop A analysis, with modal frequencies 8 Hz or below, cumulatively contribute over 94% to the total piping responses for the faulted load. The only exceptions are for several piping supports; i.e., struts and snubbers. However for those exceptions in which the piping modes at 8 Hz or below cumulatively contribute less than 94% to the total piping response, the total support loads, due to all modes, are less than 70% of the support capability.

4.0 REFERENCES

4.1 GE Nuclear Energy eDRF No. 0000-0007-4506, "Pilgrim ISM with SRSS Grouping Piping Analysis Methodology", Date Assigned: August 29, 2002.

4.2 GE Nuclear Energy Proposal No. 523-JXA4D-HK1, Rev. 1, "Process Computer Consulting Support GE Proposal No. 523-JXA4D-HK1, Rev. 1", August 8, 2002.

4.3 Entergy Contract # 4500515708, August 12, 2002. (~ Purchase Order for Reference 4.2)

4.4 Entergy Document No. Specification M626, "MS, SRVDL, HPCI and RCIC Lines Piping and Pipe Support Analysis", Date Issued: June 12, 2002.

4.5 Entergy Calculation No. M1193, "Amplified Response Spectra for Use in Specification M626", Date Issued: June 12, 2002. (Calculation performed by Stevenson & Associates)

4.6 GE Design Memo No. 124-100, "Experimentally Determined Damping Values," October 1982, by H.L. Hwang.

4.7 NUREG-1061, Volume 4, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee – Evaluation of Other Dynamic Loads and Load Combinations", Date Published: December 1984.

4.8 Pilgrim Nuclear Power Station, Unit 1 FSAR, Revision 21, October 1997.

4.9 Request No. 9B. of the U.S. Nuclear Regulatory Commission/Nuclear Reactor Regulation, Request for Additional Information (NRC/NRR RAI) associated with the Pilgrim Nuclear Power Station, Unit 1, Thermal Power Optimization (TPO) Program.

4.10 GE Nuclear Energy Letter (dkh-83-026), from D. K. Henrie to Mr. J. S. Roberts (BECo), "Pilgrim Station No. 600 Unit No. 1 Seismic Analysis – Recirculation Piping Replacement Program", November 14, 1983.

4.11 Minutes of December 6, 1983 GE, BECo and Bechtel interface meeting, "Minutes of Seismic Analysis Interface Meeting for Pilgrim Unit 1 Recirc Loop Task".

4.12 GE Nuclear Energy Internal Letter (dkh-84-011), from D. K. Henrie to GE-NE Distribution, "Generic – Methodology for Recirculation Piping Replacement Seismic Analysis", August 15, 1984.

If there are any questions or if I can be of further help, please call me at (408) 925-5964 or Page me at (800) 417-4841.

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APPENDIX A

PILGRIM NUCLEAR POWER STATION
THERMAL POWER OPTIMIZATION (TPO) PROGRAM

ANALYSIS INDEX 5
eDRF # 0000-0007-4506

Main Steam Piping Analysis - Loop A

COMPARISON OF RESULTS

Table A-1

Comparison of Results for Main Steam Model A

Analysis of Cut-off Frequency - 8 Hertz vs 100 Hertz

Case #				78R	78E	Percent Of 100 hertz load in 8 hertz run SSE only	78R	78E	Percent Of 100 hertz load in 8 hertz run Equation 9 Stress
	Results Nodal Stresses Support Loads	Allowable	Direction	3D	3D		3D	3D	
			Spectra	ISM	ISM		ISM	ISM	
			Combine	SRSS	SRSS		SRSS	SRSS	
			Damping	1.61	1.61		1.61	1.61	
			Max Elev	86.9'	86.9'		86.9'	86.9	
	–		Cut-off Freq	100	8		100	8	
			Record Analysis Stress/Load	SSE Only			$[SSE^2 + SRV^2]^{1/2}$		
SRV	Node 3	27000 psi	23771	17549	15775	89.9	26358	24807	94.1
	Node 12	27000 psi	11414	12372	11961	96.7	19347	19013	98.3
Main Steam	Node 1	27000 psi	7929	3224	3134	97.2	9057	8970	99.1
	Node 300	27000 psi	9872	2907	2434	83.7	8836	8421	95.3
Main Steam	SA-1	13300 lbs	10556	11321	8694	76.8	11730	9220	78.6
	SA-2	13300 lbs	6742	6436	4817	74.8	7107	5682	79.9
	SA-3	13300 lbs	5637	4683	3563	76.1	6072	5257	86.6
SRV	MS-S-500	22100 lbs	12231	11711	11335	96.8	13854	13537	97.7
	MS-S-501	22100 lbs	10261	2524	1000	39.6	7269	6890	94 8
	MS-S-502	13300 lbs	12331	3803	2588	68.1	8357	7878	94 3
	MS-S-503	22100 lbs	6366	3401	3246	95 4	8061	7996	99.2
	MS-S-504	13300 lbs	5379	4898	4811	98.2	8258	8207	99.4
	MS-S-505	13300 lbs	7893	2596	2561	98.7	4375	4355	99.5
	MS-S-506	13300 lbs	8776	4542	4518	99.5	6686	6670	99 8
	MS-S-507	13300 lbs	13018	7595	7559	99.5	10311	10285	99.7

Legend/Notes:

1. Direction: 3D indicates that all 3 shock direction analyses (North-South, Vertical, & East-West) are combined by the square root sum of the squares (SRSS) method.
2. Spectra: ERS is enveloped response spectra, ISM is independent support motion

3. Combine: Refers to the combination method for the seismic support groups (or zones) for the ISM analysis, SRSS indicates that the combination is done by the square root sum of the squares method.
4. Damping: 1.61 indicates Regulatory Guide 1.61 damping, 2% OBE and 3% SSE
5. Max Elev: Refers to the maximum elevation used in the analysis, 86.9' indicates spectra enveloped up to and including the reactor pressure vessel nozzle.
6. Cut-off Freq: 8 indicates the analysis was only includes the response for those modes up to 8 hertz and below, 100 indicates the analysis includes the response for all modes up to 100 hertz.
   For each model the fundamental or first frequency is as follows:

Model A, Main Steam Line A, SRV Line A:	5.252 hertz
Model B, Main Steam Line B, SRV Line D:	5.707 hertz
Model C, Main Steam Line C, RCIC line:	6.351 hertz
Model D, Main Steam Line D, SRV Lines B & C, HPCI line:	6.012 hertz

7. SSE: Inertia effects only, safe shutdown earthquake.
8. SRV: Effects of safety relief valve thrust load on attached piping components. Controlling SRV is assumed to be accident case.
9. Stresses: "SSE" stresses are the stresses due to SSE inertia affects only. $[SSE^2 + SRV^2]^{1/2}$ stresses include the affects of pressure, deadweight, SSE, SAM and SRV.
10. Support loads reported for SSE and SRV are the SRSS of SSE and SRV, SSE is calculated as $[X^2 + Y^2 + Z^2]^{1/2}$. Support loads for SSE only loading are calculated as $[X^2 + Y^2 + Z^2]^{1/2}$.
11. "Record Analysis stress/load" is taken from TR-3584-1 (1979) for Main Steam components and from Calculation 5310 (1983) for SRV components. These are the current design basis analyses for the main steam and SRV analyses.

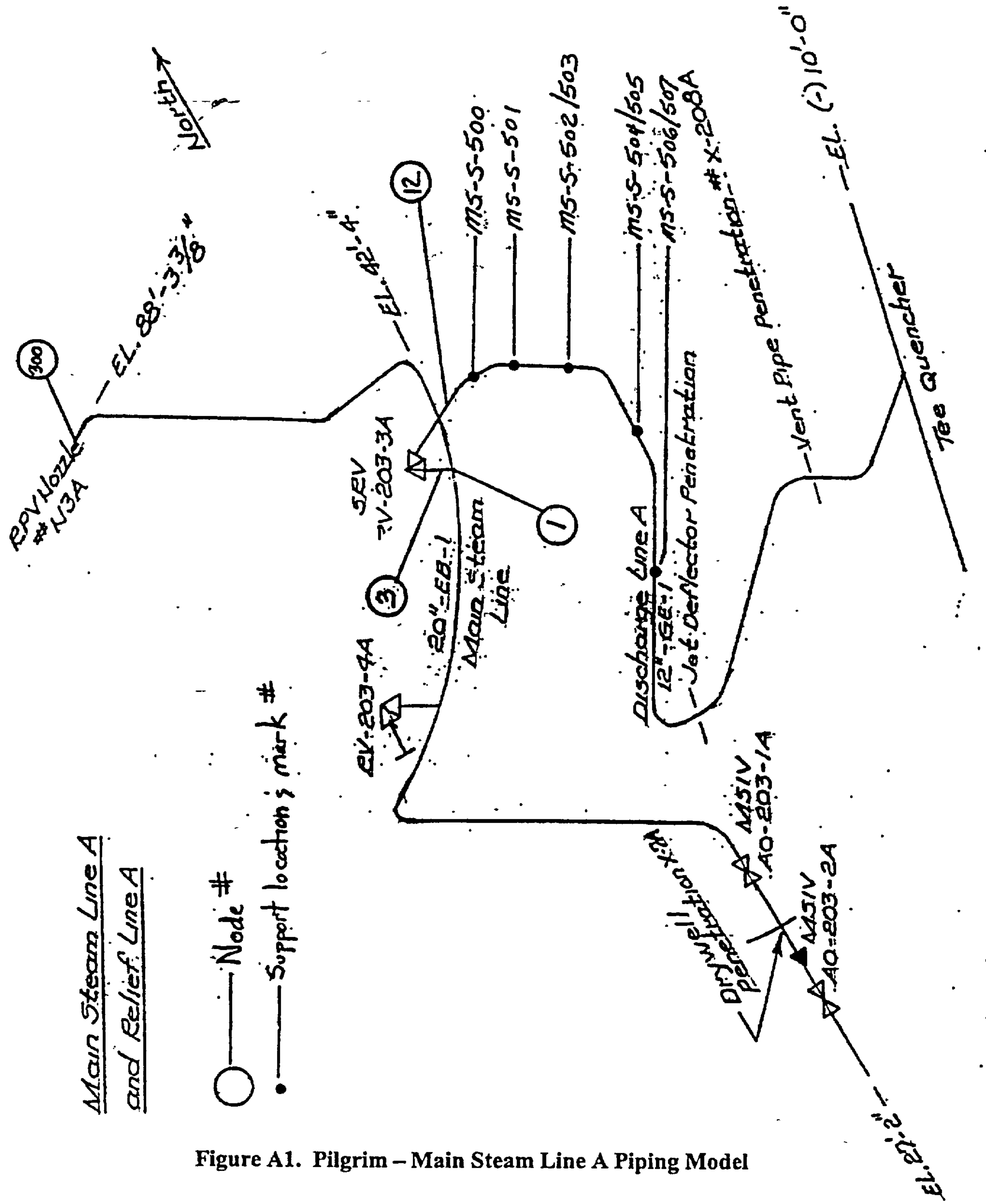

Figure A1. Pilgrim – Main Steam Line A Piping Model

APPENDIX B

PILGRIM NUCLEAR POWER STATION

BASIS FOR SRSS GROUPING FOR PIPING ANALYSIS

ANALYSIS INDEX 2
eDRF # 0000-0007-4506

Table B.1 Pilgrim Primary Structure Seismic Analysis Model Frequencies

Mode	Description	Node of Max. Displacement	Frequency (Hz.)
1	Fuel Mode	31	4.4037
2	Fuel Mode	31	4.6846
3	Reactor Pressure Vessel Mode	14	6.8444
4	Shroud-Reactor Pressure Vessel Mode	23	8.6487
5	Reactor Building Mode	1	12.265
6	RPV-Shield Wall-Pedestal Mode	14	15.353

Table B.2 Time of Maximum Seismic Acceleration in Pilgrim Primary Structure Model Analysis

Node	Structure(s)	Max. Acceleration (ft / sec**2)	Time of Max. Acceleration (Seconds)	Lowest Associated Frequency (Hz.)	Related Period (Seconds)
16	Reactor Pressure Vessel	20.770	5.085	6.844	0.1461
7	Reactor Building and Drywell	8.240	4.950	12.265	0.0815
13	Shield Wall and Pedestal	10.393	9.345	15.353	0.0644

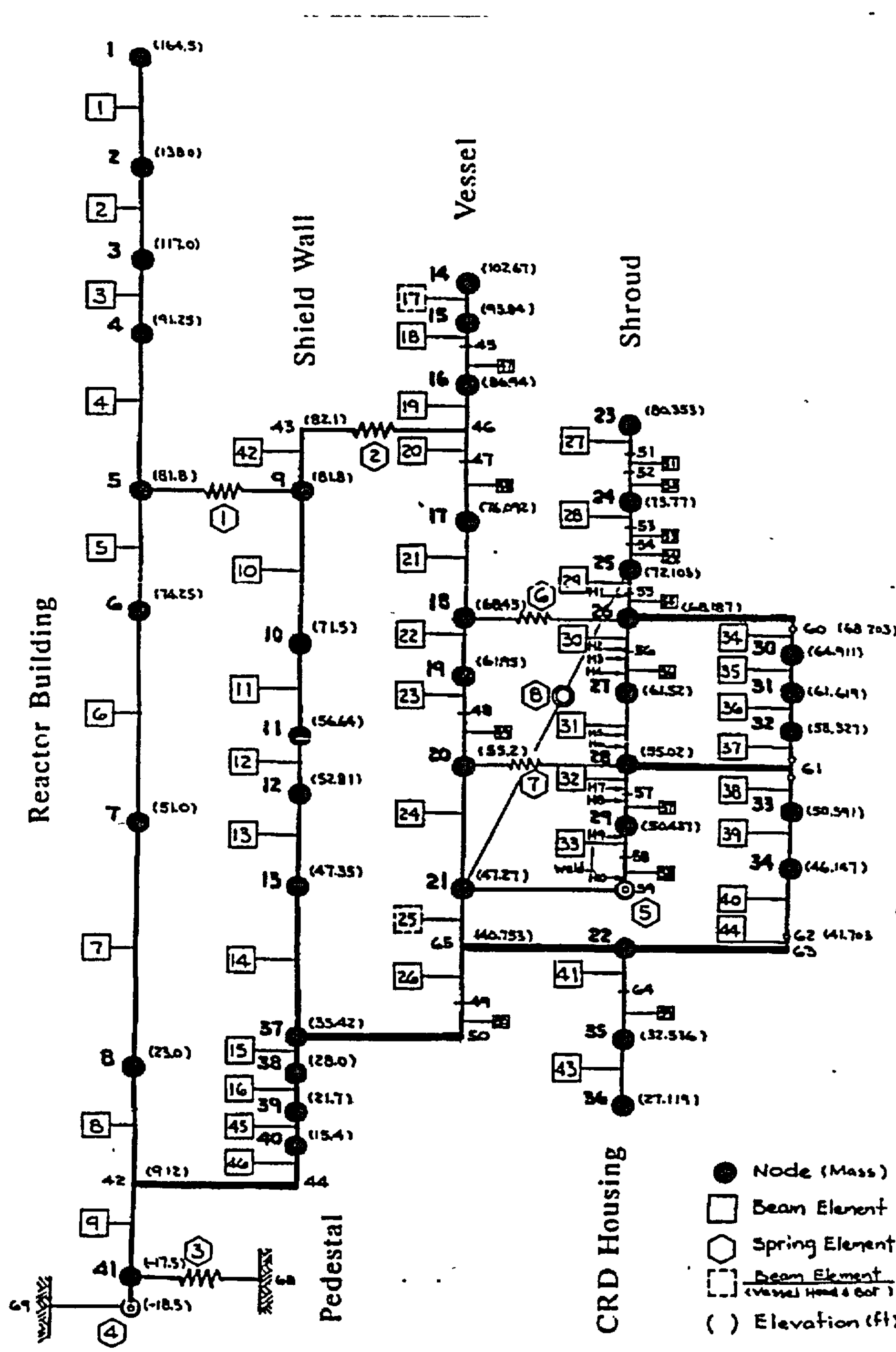

Figure B.1 Pilgrim Primary Structure Seismic Model

ATTACHMENT 4

LETTER NUMBER 2.02.096

Response to RAI 9C

Parametric Case Study – Model A – Main Steam Line A and SRV Line A

Case #			1	42A2	42A	42B	42BR	74	71	77	78	74R	71R	77R	78R	2
Main Steam or SRV	Results Nodal Stresses Support Loads	Allowable	Direction	2D	2D	2D	3D	2D	2D	2D	2D	3D	3D	3D	3D	SRV Only
			Spectra	ERS	ERS	ERS	ERS	ISM	ISM	ISM	ISM	ISM	ISM	ISM	ISM	
			Combine	-	-	-	-	ABS	SRSS	ABS	SRSS	ABS	SRSS	ABS	SRSS	
			Damping	FSAR	FSAR	1.61	1.61	FSAR	FSAR	1.61	1.61	FSAR	FSAR	1.61	1.61	
			Max Elev	42'	86.9'	86.9'	86.9'	86.9'	86.9'	86.9'	86.9'	86.9'	86.9'	86.9'	86.9'	
			Current Stress/Load	Pipe Stress												
Main Steam	Node 1	27000 psi	7929	8886	15431	12102	14003	11463	9306	9463	8120	13680	10603	10887	8950	1087
	Node 300	27000 psi	9872	8561	14121	11260	12990	10945	9325	9352	8344	12619	10266	10422	8943	1277
SRV	Node 3	27000 psi	23771	26679	61868	43547	53462	40163	28207	29363	22273	51507	34945	36475	26358	12797
	Node 12	27000 psi	11414	18361	36682	28044	35402	27905	20827	20916	16897	35030	25071	25255	19347	10935
Support Loads																
Main Steam	SA-1	13300 lbs	10556	10705	30998	20479	27700	19449	13241	13249	9259	26794	17711	18032	12152	3071
	SA-2	13300 lbs	6742	6832	18185	12172	16310	12288	8302	8639	6109	15703	10415	10811	7443	3014
	SA-3	13300 lbs	5637	7611	14387	11012	12575	10719	8878	8432	7228	12420	10082	9557	8016	3865
SRV	MS-S-500	20000 lbs	12231	13626	33291	23060	30092	22368	15461	15746	11711	29429	19820	20130	14300	7401
	MS-S-501	20000 lbs	10261	8193	9825	8730	9546	7944	7563	7427	7221	8786	7994	7871	7445	6817
	MS-S-502	13300 lbs	12331	8589	12073	9929	11571	9861	8731	8666	8101	11457	9684	9517	8581	7441
	MS-S-503	20000 lbs	6366	7875	10720	9211	9962	9293	8519	8279	7892	10140	9064	8725	8166	7308
	MS-S-504	13300 lbs	5379	8170	13111	10895	11661	10381	9034	8517	7800	11781	10020	9312	8322	6649
	MS-S-505	13300 lbs	7893	4376	6968	5816	6240	5514	4800	4523	4143	6256	5324	4946	4421	3522
	MS-S-506	13300 lbs	8776	6431	11487	9307	10019	8907	7532	6976	6211	10267	8515	7780	6753	4907
	MS-S-507	13300 lbs	13018	9228	18421	14557	15820	14259	11832	10837	9444	16623	13567	12273	10432	6974
Summary																
Main Steam and SRV	[redacted] Nodes Failing Pipe Stress			0	7	4	5	4	1	1	0	5	2	2	0	
	Add New Support(s)			No	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	No	
	Supports with Load Increase			6	8	8	8	8	6	6	3	8	7	6	6	
Main Steam	[redacted] Write off [redacted]			2	0	0	0	0	0	0	0	0	0	0	0	
	Minor Modification			0	0	0	0	0	1	0	0	0	0	0	2	
	Major Modification			1	3	3	3	3	2	3	1	3	3	3	1	
SRV	[redacted] Write off [redacted]			0	0	1	0	1	0	0	0	0	0	0	0	
	Minor Modification			2	0	0	1	0	0	0	1	1	0	0	1	
	Major Modification			1	5	4	4	4	3	3	1	4	4	3	2	
Projected Modifications – Model A				4	8	7	8	7	6	6	3	8	7	6	6	
Projected Modifications – Models A, B, C, D				17	34	30	34	30	26	26	13	34	30	26	26	

Legend

1. **Direction:** 2D indicates that the 3 shock direction analyses; North-South (X), Vertical (Y) and East-West(Z), are combined as follows: larger of $[X^2 + Y^2]^{1/2}$ or $[Y^2 + Z^2]^{1/2}$. 3D indicates that all 3-shock direction analyses are combined by the square root sum of the squares method (SRSS).
2. **Spectra:** ERS indicates enveloped response spectra, ISM indicates independent support motion.
3. **Combine:** Refers to the method used for combining seismic support groups (or zones) for the ISM analysis, ABS indicates the support groups are combined by absolute summation while SRSS indicates that the support groups are combined by the square root sum of the squares method.
4. **Damping:** FSAR indicates damping values in accordance with the FSAR: ½% OBE and 1% SSE, 1.61 indicates Regulatory Guide 1.61 damping, 2% OBE and 3% SSE.
5. **Max Elev:** Refers to the maximum elevation used in the analysis, 86.9 indicates spectra enveloped up to and including the reactor pressure vessel nozzle. 42 indicates spectra is the envelope of the Bio Shield building at elevations 35.4 and 47.4 and the Reactor Building at elevations –17.5, 23 and 51. Thus this enveloped spectra approximates the design bases center of gravity method, with an equivalent elevation of ~42', the elevation of the upper horizontal run of the main steam piping.
6. **Write Off:** Indicates that the support load has increased <10% from the current load.
7. **Minor Modification:** Assumes a minor modification will result for those supports where the load increase is > 10%, but <25%.
8. **Major Modification:** Assumes a major modification will result for those supports where the load increase is >25%. When there is a pipe stress failure at least one new support is needed, new supports are considered a major modification.

Notes

1. One new support is assumed to be needed for any case where there is a pipe stress failure.
2. Entries for stress are the values for equation 9 stress (Pressure + Deadweight + $[SSE^2 + SRV^2]^{1/2}$) Values exceeding the allowable of 27000 psi are indicated by shading.
3. Support loads reported are the SRSS of SSE and SRV, SSE is calculated as the larger of $[X^2 + Y^2]^{1/2}$ or $[Y^2 + Z^2]^{1/2}$ + SSE SAM for 2D and as $[X^2 + Y^2 + Z^2]^{1/2}$ + SSE SAM for 3D.
4. "Current stress/load" is taken from TR-3584-1 for Main Steam components and from Calculation 5310 for SRV components. These are the analyses of record for these systems.
5. Supports with load increases <10% are shaded blue.

6. Supports with load increases >10%, but <25%, resulting in minor modifications are shaded yellow.
7. Supports with load increases >25% or with new loads exceeding the snubber allowable, result in major modifications, and are shaded tan.
8. Cases and projected modifications that include new supports are shaded in green.
9. All supports shown are snubbers; the “allowable” listed is the Level C allowable per the snubber manufacturer.
10. The pipe stress allowable is the ASME Code allowable for Level C, the equation 9 stress includes the stress due to pressure, deadweight, SSE and SRV. SSE and SRV stresses are combined by SRSS.

Summary of Parametric Study

Introduction

The Pilgrim Station power uprate project proposes installation of larger safety relief valves (SRVs) during RFO #14 to improve reactor pressure relief capacity. The increased throat diameter of these valves results in higher discharge loads when they are opened. A qualifying analysis must be performed to determine the effect of this change on the design of the affected piping and piping supports. Pilgrim design criteria assumes SRV actuation occurs during a seismic event and the qualifying analyses must combine the effects of the blowdown transient with seismic loading. The associated piping includes the four main steam lines and four safety relief discharge lines inside the containment drywell.

Pilgrim's seismic analysis of record was developed in the 1970's and is less sophisticated than what would be produced today in balancing uncertainty with conservatism. For example, the horizontal input motion used for seismic analysis of the main steam lines inside the containment was simplified by using the response spectrum at "center of gravity" of the piping system at about Elevation 42 feet. This practice was common to facility designs of Pilgrim's era, and does not explicitly quantify the effect of the input motion from the reactor vessel nozzle at Elevation 86.9 feet. Pilgrim FSAR Sections 12.2.3.5.4 and C3.3.1 describe the following parameters for current piping system seismic analysis:

- Response spectrum method (e.g. using enveloped response spectra, or ERS)
- Piping damping at 0.5% OBE and 1% SSE
- Combining seismic inertia and anchor movement effects by the square-root-of-sum-of–squares method
- Calculating seismic analyses (north-south, vertical & east-west) by using the larger of $[X^2 + Y^2]^{1/2}$ and $[Y^2 + Z^2]^{1/2}$

Preparation of a new qualifying analysis for the main steam lines and safety relief discharge lines inside the containment drywell using the current design basis would not reflect current best practice for the most correct calculation of seismic response. It was desired to reduce the uncertainty associated with specifying input motion at the center of gravity of the piping system, but recognized that continued use of the enveloped response spectrum (ERS) methodology was not practical. The input motion at the reactor vessel nozzle would dominate the response spectra input motion and result in calculated pipe stresses and pipe support loads being artificially large with excessive conservatism. Rather than continue with the current design basis for seismic analysis, a decision was made to update the design parameters to reflect current best practice for the most correct calculation of seismic response. A significant benefit would be to explicitly account for the effect of input motion at the vessel nozzle, enhancing the certainty of the results leading to an improved understanding of design margin.

Study Methodology

The goal of the parametric study was to select a set of design parameters that would increase the certainty of the results and provide justification to reduce extra conservatism. An additional benefit of this design parameter selection process is to optimize the piping system modification requirements and reduce the scopes of work with minimal contribution to safety margin enhancement. A parametric study was performed with the piping system model for Main Steam Line A and Safety Relief Line (SRV) Line A as shown in Figure 1. This model is considered representative of the other three Main Steam lines and their associated SRV discharge lines. The study investigates the effects of various sets of input and analysis criteria to determine the sensitivity and impact on the calculated pipe stresses and pipe support loads. Analyses are performed using the ADLPIPE computer program using dead weight, pressure, safety relief valve discharge loading and seismic effects.

The initial step was to benchmark the newly created piping system model to the 1979 analyses of record for main steam and the 1983 analysis of record for the SRV discharge line. Comparing the two sets of reported pipe stresses and pipe support loads demonstrated that the new piping model accurately represents the design basis configuration.

The next step was to select the parameters to be varied in the piping analyses. Five items were chosen for sensitivity investigation as follows:

<u>Combination of input motion response</u>: Seismic analyses are performed for three directions of input motion (north-south, vertical & east-west) and results combined either by the **2D** FSAR method, or by using the square-root-of-sum-of-squares of all three directions, **3D**.

<u>Response spectrum input</u>: Seismic analyses are performed using either the **ERS** method discussed in the FSAR, or using independent support motion **(ISM)** with groups of seismic response spectra associated with different structures.

<u>Combination of co-linear results from different ISM groups</u>: For ISM analyses, co-linear results from different groups are combined either by absolute summation **ABS** or by the square-root-of-sum-of-squares method **SRSS.**

<u>Response Spectra Damping</u>: Piping system damping is set at 0.5% OBE and 1% SSE per the **FSAR** or Regulatory Guide **1.61** damping.

<u>Highest elevation for input motion</u>: Use of elevation **42'** associated with the center of gravity of the piping system compared to elevation **86.9'** at the reactor vessel nozzle.

<u>Results Summary</u>

The baseline case for benchmarking, identified as **42A2,** applies the current design basis for seismic analysis, i.e., 2D, ERS, FSAR damping and input motion corresponding to the spectra at the center of gravity of the piping system, Elevation 42'. Eleven additional analyses were performed to define the incremental differences in pipe stresses and pipe support loads with other sets of design parameters. Since the ultimate goal was to select a set of design criteria that would increase the certainty of the results, seven of these using **2D** and/or **ERS** parameters are academic and not under consideration as a going forward criteria.

It was decided that the final choice for the recommended set of parameters should be made from the cases reflecting the **3D** method of combining input motion responses, and the **ISM** methodology having one of the input motion groups at elevation **86.9'** to reflect the reactor vessel nozzle. These four cases are as follows:

Case	Fixed Parameters			Variable Parameters	
74R	3D	ISM	86.9'	ABS	FSAR
71R	3D	ISM	86.9'	SRSS	FSAR
77R	3D	ISM	86.9'	ABS	1.61
78R	3D	ISM	86.9'	SRSS	1.61

The baseline Case 42A2 with the current seismic design basis and new SRV loading, leads to the necessity for a small number of pipe support modifications. Each of the final four cases being considered in the parametric study results in some piping stresses exceeding allowables and/or the necessity for a larger number of pipe support modifications. These results demonstrate that the final four cases introduce additional energy into the piping system model due to the enhancements to the seismic methodology. Specific results and details are presented in Table 1.

Conclusion

Each of the final four cases is a design improvement compared with Case 42A2, because they increase the certainty of the results. All require modifications to the piping system configuration depending on the level of conservatism reflected in the damping and method of combining ISM results. Case 78R is the preferred option because use of the SRSS methodology can be shown to be technically correct and appropriate for this Pilgrim application, and because the use of R.G. 1.61 damping moves closer to representing the actual piping system response during seismic excitation. Since it is generally accepted that piping systems do not fail under seismic inertia loading, the use of the lower FSAR damping resulting in higher calculated pipe support loads and extensive modifications, would not necessarily produce a commensurate safety benefit.

Notes for Attachment 4 Table

General

Table 1 presents the equation 9 pipe stresses for selected nodes (i.e., generally maximum stress locations) from both the main steam and SRV portion of the model, along with the maximum loads imposed on the 11 snubbers in the model. The 27,000 psi allowable is based on the Level C allowable for ASME Code equation 9. This is the controlling case relative to stress margin. Equation 9 includes the stress due to internal pressure, deadweight, SSE and SRV. The SSE and SRV stresses are combined by SRSS, and then added directly to the pressure and deadweight stresses.

All the supports included in Table 1 are snubbers. The allowable reported for the snubbers, either 13,300 lbs or 20,000 lbs is the manufacturer's Level C allowable. The support loads reported in Table 1 are the SRSS of SSE seismic and SRV.

Cases 1 and 42A2

Selected results from our current design basis calculation are tabulated in the column under the heading Case 1. Case 42A2 is the beginning point of the parametric study. The record analyses are Teledyne Report TR-3584-1 for the Main Steam piping (1979) and Teledyne Calculation 5310 SRV/DL A-X208 for the SRV piping (1983). Case 42A2 mimics the record analyses; with the following differences:

- ADLPIPE is used rather than STARDYNE
- The SRV forcing functions are based on the revised throat area of the new valves to be installed.
- The ADLPIPE model uses refined support models.
- The ADLPIPE analysis is run to 100 hertz rather than 33 hertz to incorporate the effects of zero period acceleration (zpa), which was not included in the record analysis.
- Case 42A2 combines SSE inertia and SSE anchor movement by absolute summation as opposed to SRSS.

The results from Case 42A2 correlate well with the analysis of record, Case 1, using the so-called center gravity approach for seismic input, indicating the new ADLPIPE model accurately reflects the design basis configuration. The spectra used were a single curve generated by GE for elevation approximately 42 feet, as was the practice at the time of the analysis. Spectra for all other cases are based on the actual locations of support points in the piping model, taken from Pilgrim Specification C-114-ER-Q-E0 and peak spread $\pm$ 15%.

Cases 42A, 42B and 42BR

Cases 42A, 42B and 42BR are all enveloped response spectra cases where spectra for the RPV nozzle (elevation 86.9') has been added. The combination of the three shock directions is varied between the 2D method (the method specified by the FSAR, see above) and the 3D method in which all 3 orthogonal shock direction analyses are combined by the SRSS method. The damping varies between the FSAR requirements and Reg Guide 1.61 requirements. The results of all three of these cases indicated significant pipe stress failures and support load increases. Use of any of these cases would result in adding one or more new supports and potentially extensive modifications to existing pipe supports. Based on the results of these three runs it was concluded, that in order to upgrade the analysis to include the RPV nozzle spectra, an independent support motion (ISM) seismic analysis would need to be performed.

Cases 74, 71, 77 and 78

Cases 74, 71, 77, and 78 are all ISM analyses using three seismic support zones. Zone 1 is the RPV spectra and is applied only at the nozzle. Zone 2 is controlled by the bioshield spectra and zone 3 represents the lower elevations of the reactor building. All four cases use a 2D combination of the shock directions, per the FSAR. Cases 74 and 71 use FSAR damping, with the support zones combined by absolute summation and SRSS respectively. Cases 77 and 78 use Reg. Guide 1.61 damping with the support zones combined by absolute summation and SRSS respectively. From the results presented in Table 1 it can be concluded, as expected, that FSAR damping and absolute summation yield higher results than Reg. Guide 1.61 damping and SRSS.

Case 71, with FSAR damping and absolute summation, has pipe stresses exceeding the allowable limit by nearly 50%. In addition, 7 out of 11 supports are projected to have major modifications. Cases 71 and 77 yield similar results; both cases fail pipe stress by about 7% and have 6 projected major modifications. The results of Case 78 indicate that pipe stresses are acceptable and minimal modifications would be required on supports. Based on the results of these four runs it was decided to investigate the added conservatism introduced by 3d analysis.

Cases 74R, 71R, 77R, and 78R

The final four Cases 74R, 71R, 77R and 78R are the same as Cases 74, 71, 77 and 78 respectively, except that the 3 seismic shock directions are combined by SRSS. These analyses yield higher results on the order of 25%. Thus, Cases 74R, 71R and 77R fail the pipe stress criteria by as much as 90%, snubber loads are as much as twice the Level C allowable. Load Case 78R yields acceptable pipe stresses and six projected support modifications and is the preferred option for the following reasons:

- The use of SRSS methodology with ISM results can be shown to be technically correct and appropriate for this Pilgrim application (Re: GE white paper)
- The use of R.G. 1.61 damping moves closer to representing the actual piping system response during seismic excitation (Re: GE white paper)
- It is generally accepted that piping systems do not fail under seismic inertia loading, hence the use of the lower FSAR damping resulting in higher calculated pipe support loads and extensive modifications, would not necessarily produce a commensurate safety benefit.

Load Case 2

Load Case 2 reports the results for the SRV force time history analysis only, it does not include any seismic results. The SRV results included in the pipe stresses and support loads for all cases are the same, since none of the parameters that are varied in the different runs affect the SRV analysis. They are included in Table 1 so that one can determine what percent of the total load is from SRV and what percent from SSE. From Case 2 it is evident that for the supports on the main steam piping the seismic load is dominating; whereas for the SRV piping the SRV force time history loads tends to dominate.

ATTACHMENT 6

LETTER NUMBER 2.02.096

GENERAL ELECTRIC NON-PROPRIETARY DOCUMENT

General Electric NEDO-33050, Revision 1
Safety Analysis Report for Pilgrim Nuclear Power Station
Thermal Power Optimization

Double-click Icon to view Attachment

ATTACH6.PDF

202096

GE Nuclear Energy

175 Curtner Ave., San Jose, CA 95125

NEDO-33050
Revision 1
Class I
DRF-0000-0000-0017
October 2002

SAFETY ANALYSIS REPORT
FOR
PILGRIM NUCLEAR POWER STATION
THERMAL POWER OPTIMIZATION

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

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GLOSSARY OF TERMS

Term	Definition
AC	Alternating Current
ADS	Automatic Depressurization System
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ART	Adjusted Reference Temperature
ASME	American Society Of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
B&PV	Boiler and Pressure Vessel
BHP	Brake Horsepower
BOC	Beginning of Cycle
BOP	Balance Of Plant
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CD	Condensate Demineralizer
CFR	Code Of Federal Regulations
CGCS	Combustible Gas Control System
CLTP	Current Licensed Thermal Power
CO	Condensation Oscillation
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRGT	Control Rod Guide Tube
CS	Core Spray
CSC	Containment Spray Cooling
CSS	Core Support Structure
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DC	Direct Current
E1A	Stability Enhanced Option 1A
ECCS	Emergency Core Cooling System
EFPY	Effective Full Power Years
ELTR1	NEDC-32424P-A
ELTR2	NEDC-32523P-A
ENOI	Entergy Nuclear Operations, Inc.
EOC	End-of-Cycle
EOL	End of Life
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate

Term	Definition
EQ	Environmental Qualification
FAC	Flow Accelerated Corrosion
FIV	Flow-Induced Vibration
FPCCS	Fuel Pool Cooling And Cleanup System
fps	Feet Per Second
FW	Feedwater
GDC	General Design Criteria
GE	General Electric Company
GENE	GE Nuclear Energy
GL	Generic Letter
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HP	High Pressure
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilation, and Air Conditioning
HWC	Hydrogen Water Chemistry
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICA	Interim Corrective Actions
IEEE	Institute Of Electrical And Electronics Engineers
IPE	Individual Plant Examination
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
ISP	Integrated Surveillance Program
kA	Kilo-amp
ksi	Kips Per Square Inch
kV	Kilovolt
kW	Kilowatt
LCO	Limiting Conditions For Operation
LHGR	Linear Heat Generation Rate
LOCA	Loss-Of-Coolant-Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCC	Motor Control Circuit/Center
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MeV	Million Electron Volts
MHC	Mechanical-Hydraulic Control
Mlb	Millions Of Pounds

Term	Definition
MOV	Motor Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSL	Main Steam Line
MSLBA	Main Steam Line Break Accident
MVA	Million Volt Amps
MWe	Megawatt-Electric
MWt	Megawatt-Thermal
NDE	Non-Destructive Examination
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Setpoint
NUMARC	Nuclear Utilities Management and Resources Council
NUREG	Nuclear Regulations (NRC Document)
OLMCPR	Operating Limit Minimum Critical Power Ratio
OOS	Out-Of-Service
P-T	Pressure-Temperature
PBOC	Pipe Break Outside Containment
PCS	Pressure Control System
PCT	Peak Clad Temperature
PNPS	Pilgrim Nuclear Power Station
PRFO	Pressure Regulator Failure Open
PSA	Probabilistic Safety Assessment
psi	Pounds Per Square Inch
psia	Pounds Per Square Inch - Absolute
psig	Pounds Per Square Inch - Gauge
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RIS	Regulatory Issue Summary
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RT_{NDT}	Reference Temperature Of Nil-Ductility Transition
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer

Term	Definition
SAFER/GESTR-LOCA	A Computer Program, A Model For Analysis Of System Response To Loss-Of-Coolant Accident
SBO	Station Blackout
SBPCS	Steam Bypass Pressure Control System
scfm	Standard Cubic Feet Per Minute
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SPC	Suppression Pool Cooling
SR	Surveillance Requirement
SRM	Source Range Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
SSP	Supplemental Surveillance Capsule Program
SSW	Salt Service Water
TBCCW	Turbine Building Closed Cooling Water
TCV	Turbine Control Valve
TFSP	Turbine First Stage Pressure
TIP	Traversing In-Core Probe
TLTR	NEDC-32938P, Thermal Power Optimization Licensing Topical Report
TPO	Thermal Power Optimization
TSAR	Thermal Power Optimization Safety Analysis Report
TSV	Turbine Stop Valve
UFM	Ultrasonic Flow Measurement
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
USAS	United States of America Standard
USE	Upper Shelf Energy
VPCF	Velocity Profile Correction Factor
VWO	Valves Wide Open
Wd	Recirculation Drive Flow

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify increasing the licensed thermal power at the Pilgrim Nuclear Power Station (PNPS) to 2028 MWt. The requested license power level is 1.5% above the Current Licensed Thermal Power (CLTP) level of 1998 MWt.

This report follows the format and content for Boiling Water Reactor (BWR) Thermal Power Optimization (TPO) licensing reports documented in NEDC-32938P, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," called "TLTR." Per the outline of the TPO Safety Analysis Report (TSAR) in the TLTR Appendix A, every safety issue that should be addressed in a plant-specific TPO licensing report is addressed in this report. For issues that have been evaluated generically, this report references the appropriate evaluation and establishes that the evaluation is applicable to the plant.

Only previously NRC-approved or industry-accepted methods were used for the analyses of accidents and transients. Therefore, because the safety analysis methods have been previously addressed, they are not addressed in this report. Also, event and analysis descriptions that are provided in other licensing documents or the Updated Final Safety Analysis Report (UFSAR) are not repeated. This report summarizes the results of the safety evaluations needed to justify a licensing amendment to allow for TPO operation.

The TLTR addresses power increases of ≤ 1.5% of CLTP, which will produce up to approximately 2% increase in steam flow to the turbine-generator. A higher steam flow is achieved by increasing the reactor power along the current rod and core flow control lines. A limited number of operating parameters are changed, some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised, and tests similar to some of the original startup tests are performed.

Evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. This report demonstrates that PNPS can safely operate at a power level of 2028 MWt.

The evaluations were conducted in accordance with the criteria of TLTR Appendix B.

1. All safety aspects of the plant that are affected by a 1.5% increase in the thermal power level were evaluated, including the Nuclear Steam Supply System (NSSS) and Balance-of-Plant (BOP) systems.
2. Evaluations and reviews were based on licensing criteria, codes and standards applicable to the plant at the time of the TSAR submittal. There is no change in the previously established licensing basis for the plant, except for the increased power level, the increased flowrate for the Automatic Depressurization System (ADS) due to the increased throat size

in the safety relief valves (SRVs), and the use of the Independent Support Motion method of analysis for the piping affected by the increased SRV discharge loads.

3. Evaluations and/or analyses were performed using NRC-approved or industry-accepted analysis methods for the UFSAR accidents and transients affected by TPO.
4. Evaluations and reviews of the NSSS systems and components, containment structures, and BOP systems and components show continued compliance to the codes and standards applicable to the current plant licensing basis (i.e., no change to comply with more recent codes and standards is proposed due to TPO), except that the Independent Support Motion method of analysis is proposed for piping affected by SRV discharge loads.
5. NSSS components and systems were reviewed to confirm that they continue to comply with the functional and regulatory requirements specified in the UFSAR and/or applicable reload license.
6. Any modification to safety-related or non-safety-related equipment will be implemented in accordance with 10 CFR 50.59.
7. All plant systems and components affected by an increased thermal power level were reviewed to ensure there is no significant increase in challenges to the safety systems.
8. A review was performed to assure that the increased thermal power level continues to comply with the existing plant environmental regulations.
9. An assessment, as defined in 10 CFR 50.92(c), was performed to establish that no significant hazards consideration exists as a result of operation at the increased power level.
10. A review of the latest UFSAR and of design changes / 50.59 reviews implemented, but not yet shown in the UFSAR, ensures adequate evaluation of the licensing basis for the effect of TPO through the date of that evaluation. Additionally, 50.59 reviews for changes not yet implemented were reviewed for the effects of increased power.

The plant licensing requirements have been reviewed, and it is concluded that this TPO can be accommodated (1) without a significant increase in the probability or consequences of an accident previously evaluated, (2) without creating the possibility of a new or different kind of accident from any accident previously evaluated, and (3) without exceeding any existing regulatory limits applicable to the plant, which might cause a significant reduction in a margin of safety. Therefore, the requested TPO uprate does not involve a significant hazards consideration.

1.0 INTRODUCTION

1.1 OVERVIEW

This document addresses a Thermal Power Optimization (TPO) power uprate of 1.5% of the Current Licensed Thermal Power (CLTP), consistent with the magnitude of the thermal power uncertainty reduction for the Pilgrim Nuclear Power Station (PNPS). This report follows the format and content for Boiling Water Reactor (BWR) Thermal Power Optimization (TPO) licensing reports documented in NEDC-32938P, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," (TLTR) (Reference 1). Power uprates in GE BWRs of up to 120% of original licensed thermal power are based on the generic guidelines and approach defined in the Safety Evaluation Reports provided in References 2 and 3 (ELTR1 and ELTR2). Since their NRC approval, numerous extended power uprate (EPU) submittals have been based on these reports. The outline for the TPO Safety Analysis Report (TSAR) in TLTR Appendix A follows the same pattern as that used for the extended power uprates. All the issues that should be addressed in a plant-specific TPO licensing report are included in this TSAR. For issues that have been evaluated generically, this report references the appropriate evaluation and establishes that it is applicable to the plant.

1.2 PURPOSE AND APPROACH

1.2.1 TPO Analysis Basis

PNPS was originally licensed at 1998 MWt. The current safety analysis basis assumes, where required, that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level; many of the original safety analyses were performed at 105% steam flow (~104.2% thermal power). The analyses performed at 102% of CLTP remain applicable at the TPO RTP, because the 2% factor from Regulatory Guide (RG) 1.49, "Power Levels of Nuclear Power Plants," is effectively reduced by the improvement in the FW flow measurements. Some analyses may be performed at TPO RTP, because the uncertainty factor is accounted for in the methods, or the additional 2% margin is not required (e.g., Anticipated Transient Without Scram (ATWS)).

The TPO uprate is based on the evaluation of the improved FW flow rate measurement provided in Section 1.4. Figure 1-1 illustrates the TPO power/flow operating map for PNPS. The changes to the power/flow operating map are consistent with the generic descriptions given in TLTR Section 5.2. The approach to achieve a higher thermal power level is to increase core flow along the established Maximum Extended Load Line Limit Analysis (MELLLA) rod lines. This strategy allows the plant to maintain most of the existing available core flow operational flexibility while assuring that low power related issues such as stability and ATWS do not change because of the TPO uprate.

No increase in the previously licensed maximum core flow limit is associated with the TPO uprate. When end of full power reactivity condition (all rods out) is reached, end-of-cycle coastdown may be used to extend the power generation period. Previously licensed performance improvement features are presented in Section 1.3.2.

The TPO uprate is accomplished with no increase in the nominal vessel dome pressure. This minimizes the effect of uprating on reactor thermal duty, evaluations of environmental conditions, and minimizes changes to all instrument setpoints related to system pressure, etc. The high-pressure (HP) turbine steam path will be replaced to accommodate the proposed increase in steam flow. The TPO uprate does not affect the pressure control function of the turbine bypass valves. In addition, the throat size of the safety relief valves (SRVs) is being increased, which results in a 7.5% increase in SRV capacity.

1.2.2 Margins

The TPO analysis basis ensures that the power-dependent instrument error margin identified in RG 1.49 is maintained. NRC-approved or industry-accepted computer codes and calculational techniques are used in the safety analyses for the TPO uprate. A list of the Nuclear Steam Supply System (NSSS) computer codes used in the evaluations is provided in Table 1-1. Similarly, factors and margins specified by the application of design code rules are maintained, as are other margin-assuring acceptance criteria used to judge the acceptability of the plant.

1.2.3 Scope of Evaluations

The scope of evaluations is discussed in TLTR Appendix B. Tables B-1 through B-3 illustrate those analyses that are bounded by current analyses, those that are not significantly affected, and those that require updating. The disposition of the evaluations as defined by Tables B-1 through B-3 is applicable to PNPS. This TSAR includes all of the evaluations for the plant specific application. Many of the evaluations are supported by generic reference, some supported by rational considerations of the process differences, and some plant specific analyses are provided.

The scope of the evaluations are summarized in the following sections:

2.0 Reactor Core and Fuel Performance: Overall heat balance and power-flow operating map information is provided. Key core performance parameters are confirmed for each fuel cycle, and will continue to be evaluated and documented for each fuel cycle.

3.0 Reactor Coolant and Connected Systems: Evaluations of the NSSS components and systems are performed at the TPO conditions. These evaluations confirm the acceptability of the TPO changes in process variables in the NSSS.

4.0 Engineered Safety Features: The effects of TPO changes on the containment, Emergency Core Cooling System (ECCS), Standby Gas Treatment, and other Engineered Safety Features are evaluated for key events. The evaluations include the containment responses during limiting abnormal events, ECCS Loss-Of-Coolant-Accident (LOCA), and safety relief valve containment dynamic loads. The throat size of the SRVs is being increased, which results in a 7.5% increase in SRV capacity and increased discharge loads. The effect of increased SRV loads on piping and containment is addressed.

5.0 Instrumentation and Control: The instrumentation and control signal ranges and analytical limits for setpoints are evaluated to establish the effects of TPO changes in process parameters. If required, analyses are performed to determine the need for setpoint changes for various functions. In general, setpoints are changed only to maintain adequate operating margins between plant operating parameters and trip values.

6.0 Electrical Power and Auxiliary Systems: Evaluations are performed to establish the operational capability of the plant electrical power and distribution systems and auxiliary systems to ensure that they are capable of supporting safe plant operation at the TPO RTP level.

7.0 Power Conversion Systems: Evaluations are performed to establish the operational capability of various (non-safety) balance-of-plant (BOP) systems and components to ensure that they are capable of delivering the increased TPO power output.

8.0 Radwaste and Radiation Sources: The liquid and gaseous waste management systems are evaluated at TPO conditions to show that applicable release limits continue to be met during operation at the TPO RTP level. The radiological consequences are evaluated to show that applicable regulations are met for TPO including the effect on source terms, on-site doses and off-site doses during normal operation.

9.0 Reactor Safety Performance Evaluations:

[Redacted]

The standard reload analyses consider the plant conditions for the cycle of interest.

10.0 Other Evaluations: High energy line break and environmental qualification evaluations are performed at bounding conditions for the TPO range to show the continued operability of plant equipment under TPO conditions. The Probabilistic Safety Assessment (PSA) / Individual Plant Examination (IPE) is not updated, because the change in plant risk from the TPO uprate is insignificant. This conclusion is supported by the recently issued NRC Regulatory Issue Summary (RIS) 2002-03 (Reference 4). In response to feedback received during the public workshop held on August 23, 2001, the Staff wrote, "The NRC has generically determined that measurement uncertainty recapture power uprates have an insignificant impact on plant risk. Therefore, no risk information is requested to support such applications" (Guidance G.9).

1.2.4 Exceptions to the TLTR

None.

1.2.5 Concurrent Changes Unrelated to TPO

None.

1.3 TPO PLANT OPERATING CONDITIONS

1.3.1 Reactor Heat Balance

The following typical heat balance diagram at the TPO conditions is presented:

Figure 1-2 Reactor Heat Balance – TPO Power, 100% Core Flow

The small changes in thermal-hydraulic parameters for the TPO are illustrated in Table 1-2. These parameters are generated for TPO by performing coordinated reactor and turbine-generator heat balances that relate the reactor thermal-hydraulic parameters to the increased plant FW and steam flow conditions. Input from PNPS operation is considered (e.g., steam line pressure drop) to match expected TPO uprate conditions.

1.3.2 Reactor Performance Improvement Features

The following performance improvement and equipment out-of-service features currently licensed at PNPS are acceptable at the TPO thermal power: Increased Core Flow (ICF) and MELLLA.

1.4 BASIS FOR TPO UPRATE

The uncertainty of the PNPS heat balance calculation using Crossflow ultrasonic flow measurement (UFM) instrumentation to correct the plant feedwater flow measurement will be 0.5% or less in accordance with the requirements of Topical Report CENPD-397-P (Reference 5).

The topical report provides justification for increased flow measurement accuracy using the Crossflow UFM system and documents the theory, design and operating features of the Crossflow UFM. The NRC safety evaluation report (Reference 6) documents the staff's acceptance of the topical report for referencing in license applications, and also provides additional guidelines for licensees to use the Crossflow UFM for a power uprate.

The uncertainty of the PNPS heat balance calculation using Crossflow UFM instrumentation to correct the plant feedwater flow measurement will be < 0.5%. This uncertainty will be verified when the plant specific design is completed. The uncertainty evaluation will be performed at a 95% confidence level. The evaluation will address errors for feedwater flow and temperature, steam dome pressure, Control Rod Drive (CRD) flow and temperature, Reactor Water Cleanup (RWCU) system flow and temperature, recirculation pump power and efficiency, steam moisture carryover, system thermal losses, inaccurate steam tables, correction factor tolerance, and operation variances.

In addition to the requirements of the topical report, the PNPS Design Change Process will ensure that the following criteria requested by the NRC are met:

1) Maintenance and Calibration
    a) Written to ensure periodic In-service Inspections are made to verify operability requirements are met.
    b) Crossflow Out of Service.
    c) Written to periodically calibrate the Crossflow UFM equipment.
2) Operational and Maintenance History
    a) PNPS does not have an existing Crossflow UFM installation. It does have a crossbeam UFM system that was installed in 1999. The system has operated extremely well with no unscheduled outages or system failures.
3) Uncertainty Determination Methodology
    a) The uncertainty calculations will be performed using the methodology described in the topical report.
    b) The uncertainty calculation will clearly specify the requirements for 95% confidence interval flow measurements including
        i) Inside pipe diameter measurement and associated uncertainty
        ii) Transducer spacing measurement and associated uncertainty
        iii) Velocity profile correction factor (VPCF) and justification
        iv) Crossflow time delay calibration data and associated uncertainty.
    c) Crossflow operating procedures will be in place to ensure the assumptions and requirements of the uncertainty calculation are valid.
4) Site Specific Piping Configuration
    a) The vendor will provide flow laboratory test data justifying the site-specific piping configuration is bounded by the topical report.
    b) The vendor will supervise the installation of the equipment to ensure the installation guidelines in the UFM topical report are followed.

1.5 Summary and Conclusions

This evaluation has investigated a TPO uprate to 101.5% of CLTP. The strategy for achieving higher power is to extend the current power/flow map. The plant licensing challenges have been reviewed to demonstrate how the TPO uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The TPO uprate described herein involves no significant hazards consideration.

Table 1-1 Computer Codes Used For TPO Analyses

Task	Computer Code	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	(1)	NEDE-24011
Reactor Internal Pressure Differences	ISCOR	09	(1)	NEDE-32227, Oct. 1993 NEDC-32082P, Aug 1992 MFN-212-78, May 12, 1978
Anticipated Transient Without Scram	ODYN	10	Y	NEDE-24154P-A, Feb. 2000
ECCS-LOCA	LAMB	08	Y	NEDO-20566A
	GESTR	08	Y	NEDE-23785-1-PA, Rev. 1
	SAFER	04	Y	NEDE-30996P-A (2) (3)
	ISCOR	09	Y	NEDE-30130-P-A, NEDE-24011-P-A-14-US, (1)
	TASC	03	Y	(3)

NA – Not Applicable

NOTES:

(1) The heat balance application of ISCOR is not considered to be NRC approved Simple reactor system heat balance equations are used in ISCOR. This methodology is used as part of the current PNPS reload licensing analysis and there are no changes for TPO The reactor core coolant hydraulics implemented in ISCOR is documented and approved in NEDE-24011-A. Further reference is made to NEDE-30130-P-A, which was reviewed and approved for steady state analysis by the NRC. The steady state thermal-hydraulic correlations used in ISCOR are discussed in Section 4 of GESTAR II, NEDE-24011P-A, which is NRC approved.

(2) Letter, J.F. Klapproth (GE) to USNRC, "Transmittal of GE Proprietary Report NEDC-32950P 'Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," dated January 2000 by letter dated January 27, 2000

(3) Letter, S A. Richards (NRC) to J.F. Klapproth, "Review of NEDC-32084P, 'TASC-03A, A Computer Program for Transient Analysis of a Single Fuel Channel' (TAC NO. MB0564)," March 13, 2002.

Table 1-2 Thermal-Hydraulic Parameters at TPO Uprate Conditions

Parameter	Current Licensed Thermal Power	TPO Uprate Power
Thermal Power (MWt) (Percent Of Current Licensed Power)	1998 100	2028 101.5
Steam Flow (Mlb/hr) (Percent Of Current Rated)	7.984 100	8.117 101.7
FW Flow (Mlb/hr) (Percent Of Current Rated)	7.954 100	8.087 101.7
Dome Pressure (psia)	1050	1050
Dome Temperature (°F)	550.5	550 5
FW Temperature (°F)	363.0	364.3
Full Power Core Flow Range (Mlb/hr) (Percent Of Current Rated)	51.8 to 74.2 75.0 to 107.5	52.9 to 74.2 76.7 to 107.5

Table 1-3 Summary of Effect of TPO Uprate on Licensing Criteria

Key Licensing Criteria	Effect of 1.5% Thermal Power Increase	Explanation of Effect
LOCA challenges to fuel (10 CFR 50, Appendix K)	No increase in peak clad temperature (PCT), no change of maximum Linear Heat Generation Rate (LHGR) required	Previous analysis accounted for ≥102% of licensed power, bounding TPO operation with a bounding vessel pressure.
Change of Operating Limit Minimum Critical Power Ratio (OLMCPR)	< 0.01 increase	Minor increase due to slightly higher power density and increased MCPR safety limit (slightly flatter radial power distribution)
Challenges to reactor pressure vessel (RPV) overpressure	No increase in peak pressure.	No increase because previous analysis allowed ≥102% overpower, bounding TPO operation
Primary containment pressure during a LOCA	No increase in peak containment pressure.	Previous analysis allowed ≥102% overpower, bounding TPO operation No vessel pressure increase. No increase in energy to the pool.
Pool temperature during a LOCA	No increase in peak pool temperature	Previous analysis allowed ≥102% overpower, bounding TPO operation. No vessel pressure increase. No increase in energy to the pool.
Offsite Radiation Release, design basis accidents	No increase (remains within 10 CFR 100).	Existing analysis at 102% overpower, bounds TPO operation. No vessel pressure increase.
Onsite Radiation Dose, normal operation	~1.5% increase, must remain within 10 CFR 20	Slightly higher inventory of radionuclides in steam/FW flow paths
Heat discharge to environment	~1°F temperature increase	Small % power increase.
Equipment Qualification	Remains within current pressure, radiation, and temperature envelopes.	No change in Harsh Environment terms (bounded by previous design using ≥102% power); minimal change in normal operating conditions.
Fracture Toughness (10 CFR 50, Appendix G)	< 2°F increase in RT_{NDT}	Small increase in neutron fluence.
Stability	No direct effect of TPO uprate because applicable stability regions and lines are extended beyond the absolute values associated with the current boundaries to preserve MWt-core flow boundaries as applicable for each stability option.	No increase in maximum rod line boundary. Characteristics of each reload core continue to be evaluated as required for each stability option.
ATWS peak vessel pressure	Insufficient pressure margin exists to American Society of Mechanical Engineers (ASME) Code "Emergency" category stress limit.	Results of plant-specific analysis using increased SRV capacity meet peak vessel pressure acceptance criteria.
Vessel and NSSS equipment design pressure	No change.	Comply with existing ASME Code stress limits of all categories

Figure 1-1 Power/Flow Map for PNPS at TPO Uprate Power

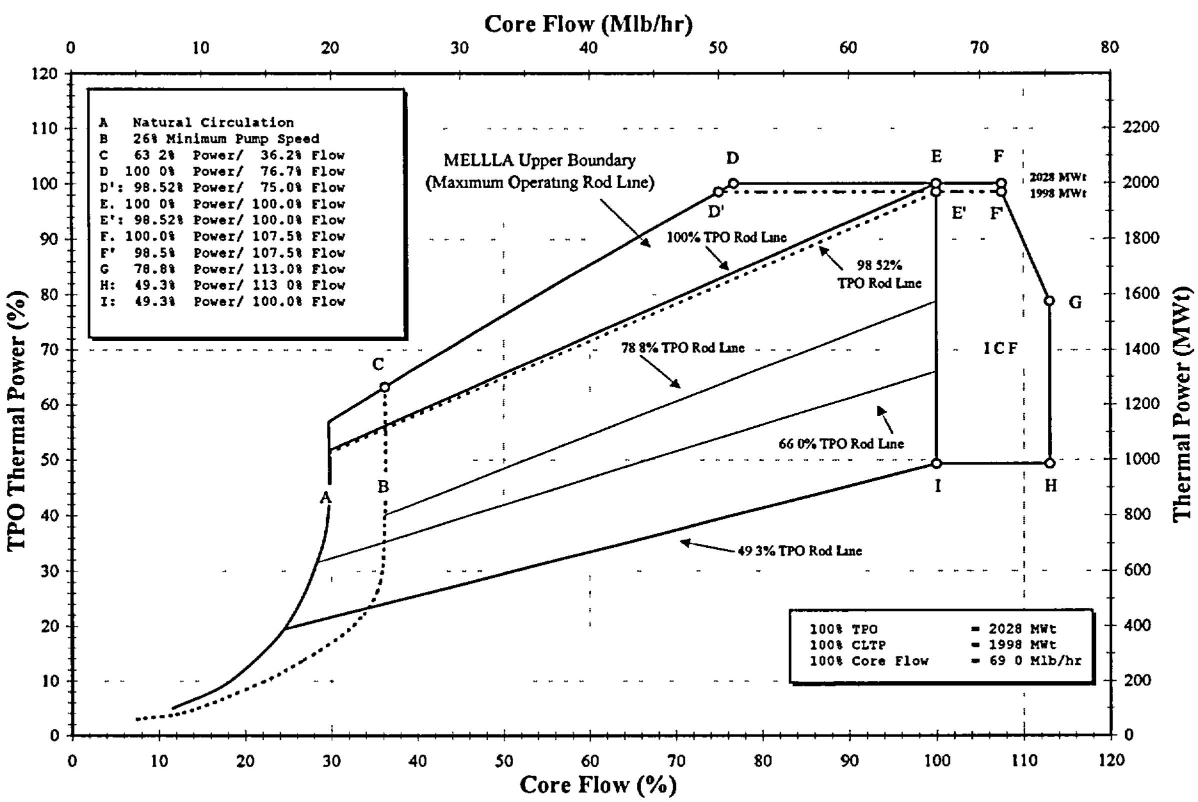
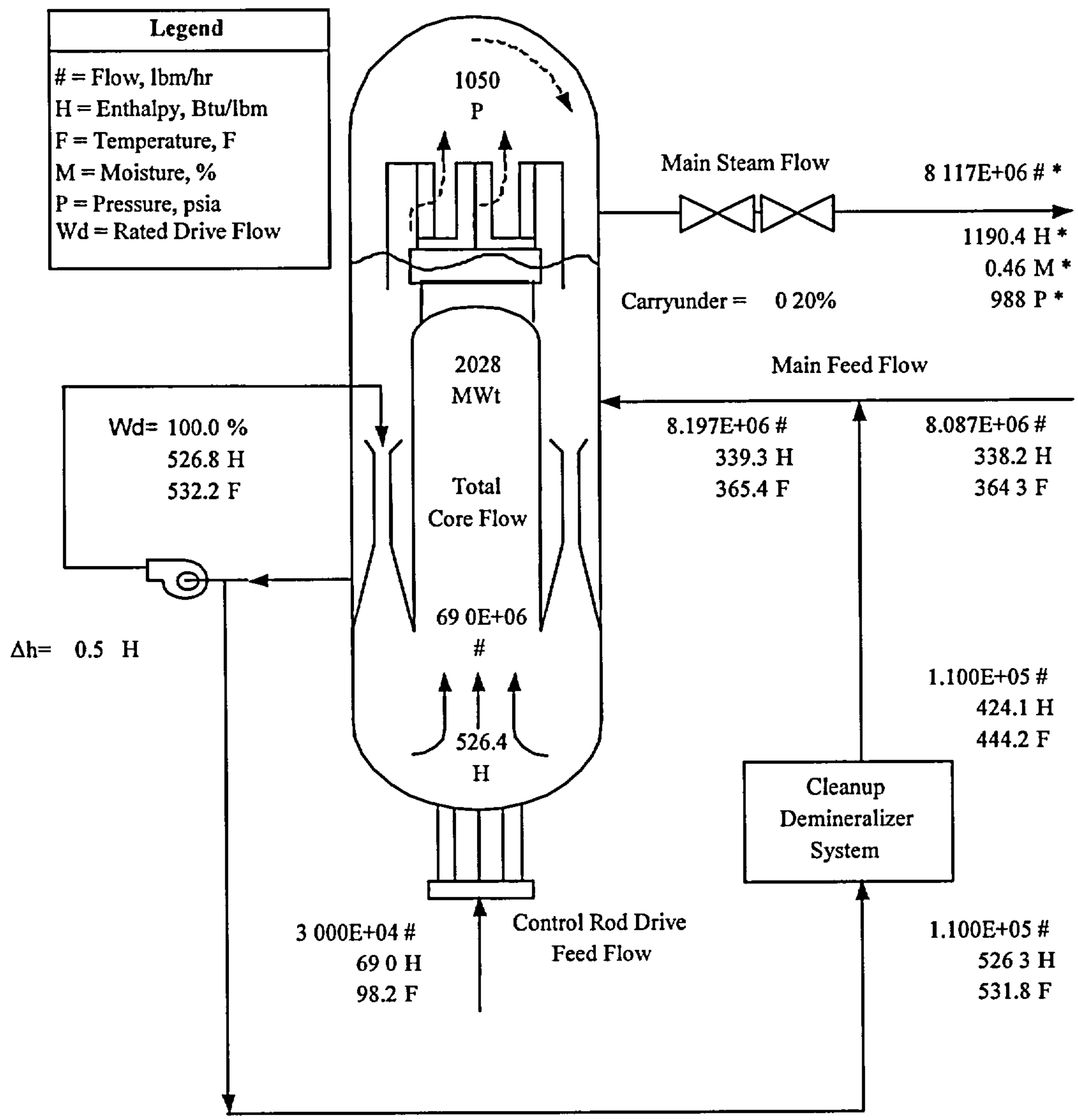


Figure 1-2 Reactor Heat Balance – TPO Power, 100% Core Flow



* Conditions at upstream side of TSV

Core Thermal Power	2028 0
Pump Heating	5.5
Cleanup Losses	-3.3
Other System Losses	-0 6
Turbine Cycle Use	2029.6 MWt

2.0 REACTOR CORE AND FUEL PERFORMANCE

2.1 FUEL DESIGN AND OPERATION

At the TPO Rated Thermal Power (RTP) conditions, all fuel and core design limits are met by the deployment of fuel enrichment and burnable poison, control rod pattern management, and core flow adjustments. New fuel designs are not needed for the TPO to ensure safety. However, revised loading patterns, slightly larger batch sizes, and potentially new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length. NRC-approved limits for burnup on the fuel are not exceeded. Therefore, the reactor core and fuel design will be addressed by the first normal reload for TPO operation.

2.2 THERMAL LIMITS ASSESSMENT

Operating thermal limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (e.g., transients, LOCA). This section addresses the effects of TPO on thermal limits. Cycle-specific core configurations, which are evaluated for each reload, confirm TPO RTP capability and establish or confirm cycle-specific limits.

The historical 25% of RTP value for the Technical Specification Safety Limit, some thermal limits monitoring Limiting Conditions for Operation (LCOs) thresholds, and some Surveillance Requirements (SRs) thresholds is based on

[Redacted]

The historical 25% RTP value is a conservative basis, as described in the plant Technical Specifications,

[Redacted]

Therefore, the Safety Limit percent RTP basis, thermal limits monitoring LCOs, and SR percent RTP thresholds remain at 25% RTP for the TPO uprate.

2.2.1 Safety Limit MCPR

The Safety Limit Minimum Critical Power Ratio (SLMCPR) is dependent upon the nominal average power level and the uncertainty in its measurement. Consistent with approved practice, a revised SLMCPR is calculated for the first TPO fuel cycle and confirmed for each subsequent cycle. The historical uncertainty allowance and calculational methods are discussed in TLTR Section 5.7.2.1.

2.2.2 MCPR Operating Limit

TLTR Appendix E shows that the changes in the OLMCPR for a TPO uprate
[Redacted]

Because the cycle-specific SLMCPR is also defined, the actual required OLMCPR can be established. This ensures an adequate fuel thermal margin for TPO uprate operation.

2.2.3 MAPLHGR and Maximum LHGR Operating Limits

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and maximum LHGR limits are maintained as described in TLTR Section 5.7.2.2. No significant change results due to TPO operation. The LHGR limits are fuel dependent and are not affected by the TPO. The ECCS performance is addressed in Section 4.3.

2.3 REACTIVITY CHARACTERISTICS

All minimum shutdown margin requirements apply to cold shutdown (≤ 212°F) conditions and are maintained without change. Checks of cold shutdown margin based on Standby Liquid Control System (SLCS) boron injection capability and shutdown using control rods with the most reactive control rod stuck out are made for each reload. The TPO uprate has no significant effect on these conditions; the shutdown margin is confirmed in the reload core design.

Operation at the TPO RTP could result in a minor decrease in the hot excess reactivity during the cycle. This loss of reactivity does not affect safety, and does not affect the ability to manage the power distribution through the cycle to achieve the target power level. However, the lower hot excess reactivity can result in achieving an earlier all-rods-out condition. Through fuel cycle redesign, sufficient excess reactivity can be obtained to match the desired cycle length.

2.4 STABILITY

Because the rod line is not changed for the TPO uprate, there is minimal effect on stability beyond the normal cycle-to-cycle core characteristic variations that are evaluated with each reload. PNPS utilizes reactor stability Enhanced Option I-A (E1A), which is not affected by the TPO uprate. Reload stability evaluations continue to ensure acceptable stability performance and protection for future cores operating at TPO uprated conditions.

2.5 REACTIVITY CONTROL

The generic discussion in TLTR Sections 5.6.3 and J.2.3.3 applies to PNPS. The CRD and CRD Hydraulic systems and supporting equipment are not affected by the TPO uprate and no further evaluation of CRD performance is necessary.

3.0 REACTOR COOLANT AND CONNECTED SYSTEMS

3.1 NUCLEAR SYSTEM PRESSURE RELIEF / OVERPRESSURE PROTECTION

The pressure relief system prevents overpressurization of the nuclear system during abnormal operational transients. The SRVs along with other functions provide this protection. Evaluations and analyses for the CLTP have been performed at 102% of CLTP to demonstrate that the reactor vessel conformed to ASME Boiler and Pressure Vessel (B&PV) Code and plant Technical Specification requirements. There is no increase in nominal operating pressure for the PNPS TPO uprate. There are no changes in the SRV or safety valve setpoints or valve out-of-service options; however, the total SRV capacity will be increased by increasing the existing SRV throat diameters. There is no change in the methodology or the limiting overpressure event. Therefore, the generic evaluation contained in the TLTR is applicable

The analysis for each fuel reload, which is current practice, confirms the capability of the system to meet the ASME design criteria.

3.2 REACTOR VESSEL

The RPV structure and support components form a pressure boundary to contain reactor coolant and moderator, and form a boundary against leakage of radioactive materials into the drywell. The RPV also provides structural support for the reactor core and internals.

3.2.1 Fracture Toughness

TLTR Section 5.5.1.5 describes the RPV fracture toughness evaluation process. The end of life (EOL) fluence is calculated for the TPO uprate conditions and from the fluence for current conditions to evaluate the vessel against the requirements of 10 CFR 50, Appendix G. The results of these evaluations indicate that:

- The upper shelf energy (USE) remains greater than 50 ft-lb for the design life of the vessel and maintains the margin requirements of 10 CFR 50, Appendix G. The minimum EOL USE for beltline materials is 85 ft-lb for 40 years of operation.
- The beltline material reference temperature of the nil-ductility transition (RT_{NDT}) remains below 200°F.
- The 27 effective full power year (EFPY) (40 year life) shift is decreased from the 32 EFPY shift. The adjusted reference temperature (ART) values for 27 EFPY are provided in Table 3-1.
- The surface fluence decreases for 27 EFPY (40 year life) including TPO. Because 1/4T fluence contributes to the resulting ART, there is no change to ART or Shift for up to and including 27 EFPY. The Pressure-Temperature (P-T) curves currently licensed for PNPS for 32 EFPY account for a Shift value of 105°F. The Shift value calculated for TPO is

unchanged up to 32 EFPY. Therefore, the current 32 EFPY P-T curves are valid with TPO.

- The reactor vessel material surveillance program consists of three capsules. The three capsules have been in the reactor vessel since plant startup. One of these capsules was removed after approximately 4.17 EFPY of operation. PNPS is part of the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) / Supplemental Surveillance Capsule Program (SSP) and will comply with the withdrawal schedule specified for surrogate surveillance capsules that now represent PNPS. Therefore, the 10 CFR 50, Appendix H surveillance capsule schedule for the ISP/SSP will govern. Implementation of the TPO uprate has no effect on the BWRVIP withdrawal schedule.

The maximum operating dome pressure for the TPO uprate is unchanged from current operation. Therefore, no change in the hydrostatic and leakage test pressures is required. The vessel is still in compliance with the regulatory requirements at TPO uprate conditions.

3.2.2 Reactor Vessel Structural Evaluation

The effect of the TPO uprate was evaluated to ensure that the RPV components comply with the existing structural requirements of the ASME B&PV Code. For the components under consideration, the code of construction, the 1965 Edition of the Code with addenda to and including Summer 1966, was used as the governing Code. However, for a component that underwent a previous design modification, the governing code for that component was the code used in the stress analysis of the modified component.

Evaluations were performed using the procedures defined in TLTR Appendix I.

[Redacted]

The stress and fatigue analyses were performed for the design, the Normal and Upset, and the Emergency and Faulted conditions.

3.2.2.1 Design Conditions

For the TPO uprate, the RPV design requirements are unchanged from the original design requirements specified in the RPV purchase documents.

3.2.2.2 Normal and Upset Conditions

The reactor coolant temperature and flows (except core flow) at the TPO conditions are changed from those at current rated conditions. Evaluations were performed at conditions that bound the small change in operating conditions. A primary plus secondary stress analysis indicates that TPO stresses still meet the requirements of the ASME Code. Elastic-plastic methods were implemented for some components and the Code requirements for these methods were met. The fatigue usage was also evaluated for limiting components with a usage factor greater than 0.5. The fatigue analysis results for the limiting components are provided in Table 3-2. The analysis results for the TPO uprate show that all components meet the ASME Code requirements. The current basis for the PNPS Upset transient conditions bounds the transient conditions for TPO operation.

3.2.2.3 Emergency and Faulted Conditions

The TPO uprate does not change the Emergency and Faulted conditions for PNPS because the previous evaluations bound the analysis required for such cases under TPO operation. The same analyzed conditions are sufficient for the TPO evaluation. Therefore, the existing Emergency and Faulted stress analysis continues to meet the requirements of the ASME Code. The current assessment of the "original" Certified Stress Report applies to PNPS for the TPO uprate.

3.3 REACTOR INTERNALS

The reactor internals include core support structure (CSS) and non-core support structure (non-CSS) components.

3.3.1 Reactor Internal Pressure Difference

The Reactor Internal Pressure Differences (RIPDs) are more strongly affected by the maximum licensed core flow rate than by the power level. The maximum flow rate is not changed for the TPO uprate. The effect due to the changes in loads for both Normal and Upset conditions is reported in Section 3.3.2. The Emergency and Faulted evaluations of RIPD for TPO uprate are bounded by the current analyses performed at 102% of CLTP conditions. Fuel bundle lift margins are only calculated for the faulted conditions, to demonstrate that fuel bundles would not be lifted under the worst conditions, thus they are not calculated. As an older plant, the PNPS licensing basis does not require the hydraulic lift forces to be combined with seismic loads. Thus, the hydraulic control rod guide tube (CRGT) lift forces are not calculated.

3.3.2 Reactor Internals Structural Evaluation

The reactor internal components were evaluated for structural integrity due to load changes associated with the TPO uprate. Consistent with the TLTR, only Normal and Upset conditions were evaluated. The evaluation considered the effect of TPO on pressure, temperature, weight, seismic, and flow loads, as applicable, and was performed consistent with the design bases for the components. The TPO loads were either bounded by the design basis values or the changes were insignificant. Therefore, the reactor internal components remain qualified for the TPO uprate.

3.3.3 Steam Separator and Dryer Performance

The steam separator and dryer performance evaluation is described in TLTR Section 5.5.1.6. [Redacted] the expected performance of the steam separators and dryer was evaluated [Redacted] to ensure that the quality of the steam leaving the RPV continues to meet the operational criteria at the TPO uprate conditions. TPO RTP operation results in an increase in saturated steam generated in the reactor core. For constant core flow, the increased steam flow results in a slight increase in the separator inlet quality and dryer face velocity and a decrease in the water level inside the dryer skirt, all of which affect the steam separator-dryer performance. The results of the evaluation demonstrate that the steam separator-dryer performance exceeds the original design limit for steam moisture content at both CLTP and TPO RTP for bounding conditions of high radial power peaking and low core flow. Reducing radial power peaking and/or increasing core flow can reduce the moisture content to an acceptable value. Steam moisture content will be monitored during initial TPO startup testing (Section 10.4) to determine acceptable operational moisture content.

3.4 FLOW INDUCED VIBRATION

The process for the RPV internals vibration assessment is described in TLTR Section 5.5.1.3. An evaluation determined the effects of flow-induced vibration (FIV) on the reactor internals at TPO RTP and 107.5% rated core flow. The vibration levels for the TPO uprate conditions were estimated from vibration data recorded during startup testing of PNPS and during other tests. These expected vibration levels were compared with established vibration acceptance limits. The following components were evaluated for the TPO uprate:

Component(s)	Process Parameter(s)	TPO Evaluation
Shroud Shroud Head and Separator Steam Dryer	Steam flow at TPO RTP is ~2% greater than CLTP	Less than 10% increase in FIV
Jet Pumps	Jet pump flow at TPO RTP is identical to CLTP	No Change
Jet Pump Sensing Lines	Vane passing frequency of recirculation pumps	No Change In Possibility of Resonance
FW Sparger	FW flow at TPO RTP is ~2% greater than CLTP	Less than 10% increase in FIV

The calculations for the TPO uprate conditions indicate that vibrations of all essential to safe shutdown reactor internal components are within the GE acceptance criteria. The evaluation is based on the as-designed condition of the components. Any deviation from the as-designed condition will be evaluated and dispositioned in accordance with the BWRVIP Inspection and Flaw Evaluation Guidelines before TPO implementation. Further, the Cumulative Usage Factor (CUF) due to FIV is negligible for the TPO uprate.

The analysis is conservative for the following reasons:

- The GE criteria of 10,000 psi peak stress intensity is much more conservative than the ASME allowable peak stress intensity of 13,600 psi for service cycles equal to 10^{11}.
- The modes are absolute summed.
- The maximum vibration amplitude in each mode is used in the absolute sum process, whereas in reality the vibration amplitude fluctuates.

Therefore, the flow-induced vibrations for all evaluated components in the as-designed condition remain within acceptable limits.

The safety-related Main Steam (MS) and FW piping have minor increased flow rates and flow velocities resulting from the TPO uprate. The MS and FW piping experience minor increased vibration levels, approximately proportional to the square of the flow velocities and also in proportion to any increase in fluid density. The decrease in FW fluid density for TPO uprate conditions as a result of the < 2°F temperature increase is insignificant.

3.5 PIPING EVALUATION

3.5.1 Reactor Coolant Pressure Boundary Piping

The methods used for the piping and pipe support evaluations are described in TLTR Appendix K. These approaches are identical to those used in the evaluation of previous BWR power uprates of up to 20% power. The effect of the TPO uprate with no nominal vessel dome pressure increase is negligible for the Reactor Coolant Pressure Boundary (RCPB) portion of all piping except for portions of the FW lines, MS lines, and piping connected to the FW and MS lines. The following table summarizes the evaluation of the piping inside containment.

Component(s) / Concern	Process Parameter(s)	TPO Evaluation
Recirculation System Pipe Stresses Pipe Supports	Nominal dome pressure at TPO uprate power is identical to CLTP Recirculation flow at TPO uprate power is identical to CLTP Small increase in core pressure drop of < 1psi Recirculation fluid temperature decreases <1°F	Negligible change in pipe stress Negligible effect on pipe support loads
MS and Attached Piping (Inside Containment) except for the SRV Discharge Line (SRVDL) piping (see below) (e g., Reactor Core Isolation Cooling (RCIC) / High Pressure Coolant Injection (HPCI) piping (Steam Side), MS drain lines, RPV head vent line piping located inside containment) Pipe Stresses Pipe Supports Flow Accelerated Corrosion (FAC)	Nominal dome pressure at TPO uprate power is identical to CLTP Steam flow at TPO uprate power is ~2% greater than CLTP No change in MS line pressure and temperature	Current licensing basis envelopes TPO conditions; therefore, piping systems are acceptable for TPO. Minor increase in the potential for FAC (FAC concerns are covered by existing piping monitoring program)
SRVDL piping	Steam flow at TPO uprate power is ~2% greater than CLTP	SRVDL is being addressed by the Independent Support Motion methodology
FW Piping(Inside Containment) Pipe Stresses Pipe Supports FAC	Nominal dome pressure at TPO uprate power is identical to CLTP FW flow at TPO uprate power is ~2% greater than CLTP Minor increase in FW line pressure < 1 psi Fluid temperature increases < 2°F	Current licensing basis envelopes TPO conditions, therefore, piping systems are acceptable for TPO. Minor increase in the potential for FAC (FAC concerns are covered by existing piping monitoring program)

Component(s) / Concern	Process Parameter(s)	TPO Evaluation
RPV bottom head drain line, RCIC piping, HPCI piping, Low Pressure Coolant Injection (LPCI) piping, CS piping, SLCS piping, and RWCU piping Pipe Stresses Pipe Supports	Nominal dome pressure at TPO uprate power is identical to CLTP Small increase in core pressure drop of < 1 psi Recirculation fluid temperature decreases ~1°F	Negligible change in pipe stress Negligible effect on pipe support loads

Main Steam and Attached Piping System Evaluation

Except for the SRVDL piping, the MS and attached piping system (inside containment) is acceptable, because the changes in flow, pressure, and temperature were considered insignificant under the TPO condition and system pressure and temperature are within original design limits. Except for the SRVDL piping, the current licensing basis for the MS piping system (inside containment) analyzed for pressure and temperature envelopes the TPO operating pressure and temperature. For the SRVDL piping, the modified SRVs permit ~7.5% additional flow. The subsequent increased loads will be analyzed and addressed prior to implementation of the TPO uprate. The ~2% increase in the MS flow does not have a significant effect on the MS piping system stress and support loads. Therefore, all safety aspects of the MS piping system (inside containment) are within current licensing basis evaluations.

Flow Accelerated Corrosion

The carbon steel MS piping can be affected by FAC. FAC is affected by changes in fluid velocity, temperature and moisture content. PNPS has an established program for monitoring pipe wall thinning in single and two-phase high energy carbon steel piping. The variation in velocity, temperature, and moisture content resulting from the uprate are minor changes to parameters affecting FAC.

No changes to piping inspection scope and frequency are required to ensure adequate margin for the changing process conditions. The continuing inspection program takes into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program provides assurance that the TPO uprate has no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to FAC.

Feedwater Piping System Evaluation

The FW Piping system (inside containment) is acceptable, because the changes in flow, pressure, and temperature were considered insignificant under the TPO condition and system pressure and temperature are within original design limits. The current licensing basis for the FW piping system (inside containment) analyzed for pressure and temperature envelopes the TPO operating pressure and temperature. The < 2% increase in the FW flow does not have a significant effect on the FW piping system stress and support loads, because there are no fast closing valves in the

FW piping system (inside containment). Therefore, all safety aspects of the FW piping system (inside containment) are within current licensing basis evaluations.

Flow Accelerated Corrosion

The carbon steel FW piping can be affected by FAC. FAC in the FW piping is affected by changes in fluid velocity and temperature. PNPS has an established program for monitoring pipe wall thinning in single and two-phase high energy carbon steel piping. The variation in velocity and temperature resulting from the TPO uprate are minor changes to parameters affecting FAC.

No changes to piping inspection scope and frequency are required to ensure adequate margin exists for the TPO process conditions. The continuing inspection program takes into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program provides assurance that the TPO uprate has no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to FAC.

3.5.2 Non-RCPB Safety Class Piping Evaluation

This section addresses the adequacy of the non-RCPB safety class piping design for operation at the TPO conditions. Large-bore and small-bore Safety Class, ISI Class 2 and 3 piping and supports not addressed in Section 3.5.1 were evaluated for acceptability at the TPO conditions.

The Design Basis Accident (DBA) LOCA dynamic loads including the pool swell loads, vent thrust loads, condensation oscillation (CO) loads, and chugging loads were originally defined based on analyses at 102% of CLTP. The structures attached to the containment, such as the piping systems, vent penetrations, and valves are based on these DBA LOCA hydrodynamic loads. For the TPO conditions, the DBA LOCA containment response loads do not change.

3.5.3 Balance-of-Plant Non-Safety Class Piping Evaluation

The evaluation of the BOP piping and supports was performed by comparing the TPO operating conditions to the design pressure and temperature in Piping Specification M300. The original design Code for BOP piping and also for most safety systems was USAS B31.1.0 and different levels of Quality and Non-Destructive Examination (NDE) were applied for Safety-Related versus Non-Safety-Related Piping.

The effects of the TPO uprate have been evaluated for the following BOP piping systems:

- MS-Outside Containment
- MS Turbine By-Pass Piping
- MS Isolation Valve Drain Piping
- Extraction Steam
- FW–Outside Containment and to the Inboard FW Check Valve

- Condensate
- Condensate Demineralizer (Less than 0.1°F change and not significant)
- FW Heater Drain

Pipe Stresses

Operation at TPO uprate conditions results in slightly higher flow rates internal to the pipes (typically 5% or less). Pressure increases are in the order of 5 psi or less and temperature increases are 4°F or less. Comparison of the design and TPO conditions show that the TPO pressures and temperatures are generally within the design limits. The largest deviation is a 6-psi (~3%) and 2°F (~0.5%) increase over design for an extraction steam line. This is not considered significant and other deviations are smaller. Resulting changes in stress are either within design or increases over design are so small that they are not significant.

Pipe Supports

Because there is no change in the MS temperature, there is no change in the MS pipe support loads. The supports for piping that contains fluid that increases in temperature, (e.g. the FW piping) have slightly increased pipe support loadings. However, when considering the loading combination with other loads that are not affected by the TPO uprate, such as seismic and deadweight, the combined support load increase is insignificant.

Flow Accelerated Corrosion

The integrity of high energy piping systems is assured by proper design in accordance with the applicable codes and standards. Piping thickness of carbon steel components can be affected by FAC. PNPS has an established program for monitoring pipe wall thinning in single phase and two-phase high-energy carbon steel piping. FAC rates may be influenced by changes in fluid velocity, temperature, and moisture content.

Operation at the TPO RTP results in some changes to parameters affecting FAC in those systems associated with the turbine cycle (e.g., condensate, FW, MS). The evaluation of and inspection for FAC in BOP systems is addressed by compliance with Generic Letter (GL) 89-08, "Erosion/Corrosion in Piping." The plant FAC program currently monitors the affected systems. Continued monitoring of the systems provides confidence in the integrity of susceptible high energy piping systems. Appropriate changes to piping inspection frequency will be implemented to ensure adequate margin exists for those systems with changing process conditions. This action takes into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program provides assurance that the TPO has no adverse effect on high energy piping systems potentially susceptible to pipe wall thinning due to FAC. In addition, much of the FAC-susceptible piping has been replaced with FAC resistant 1¼% Cr, ½% Mo piping.

3.6 Reactor Recirculation System

The Reactor Recirculation system evaluation process is described in TLTR Section 5.6.2. The TPO uprate has a minor effect on the recirculation system and its components. The TPO uprate does not require an increase in the maximum core flow. No significant reduction of the maximum flow capability occurs due to the TPO uprate because of the small increase in core pressure drop (< 1 psi). An evaluation has confirmed that no significant increase in recirculation system vibration occurs from the TPO operating conditions.

3.7 Main Steam Line Flow Restrictors

The generic evaluation provided in TLTR Appendix J is applicable to PNPS. The requirements for the main steam line flow restrictors remain unchanged for TPO uprate conditions. Even though rated steam flow is slightly increased, no change in steam line break flow rate occurs because the operating pressure is unchanged. All safety and operational aspects of the main steam line flow restrictors are within previous evaluations.

3.8 Main Steam Isolation Valves

The generic evaluation provided in TLTR Appendix J is applicable to PNPS. The requirements for the main steam isolation valves (MSIVs) remain unchanged for TPO uprate conditions. All safety and operational aspects of the MSIVs are within previous evaluations.

3.9 Reactor Core Isolation Cooling

The RCIC system provides inventory makeup to the reactor vessel when the vessel is isolated from the normal high pressure makeup systems. The generic evaluation provided in TLTR Section 5.6.7 is applicable to PNPS. The TPO uprate does not affect the RCIC system operation, initiation, or capability requirements.

3.10 Residual Heat Removal System

The Residual Heat Removal (RHR) system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post accident conditions. The RHR system is designed to function in several operating modes. The generic evaluation provided in TLTR Sections 5.6.4 and J.2.3.13 is applicable to PNPS.

The following table summarizes the effect of the TPO on the design basis of the RHR system.

Operating Mode	Key Function	TPO Evaluation
LPCI Mode	Core Cooling	See Section 4.2.4
Suppression Pool Cooling (SPC) and Containment Spray Cooling (CSC) Modes	Normal SPC function is to maintain pool temperature below the limit. For Abnormal events or accidents, the SPC mode maintains the long-term pool temperature below the design limit. The CSC mode sprays water into the containment to reduce post-accident containment pressure and temperature.	Containment Analyses have been performed at 102% CLTP.
Shutdown Cooling (SDC) Mode	Removes sensible and decay heat from the reactor primary system during a normal reactor shutdown.	The slightly higher decay heat has negligible effect on the SDC mode, which has no safety function.
Fuel Pool Cooling Assist	Supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capability of the Fuel Pool Cooling system.	See Section 6.3.1

The ability of the RHR system to perform required safety functions is demonstrated with analyses based on 102% of CLTP. Therefore, all safety aspects of the RHR system are within previous evaluations. The requirements for the RHR system remain unchanged for TPO uprate conditions.

3.11 Reactor Water Cleanup System

The generic evaluation of the RWCU system provided in TLTR Sections 5.6.6 and J.2.3.4 is applicable to PNPS. The performance requirements of the RWCU system are negligibly affected by TPO uprate. There is no significant effect on operating temperature and pressure conditions in the high-pressure portion of the system. Steady power level changes for much larger power uprates have shown no effect on reactor water chemistry and the performance of the RWCU system. Power transients are the primary source of challenge to the system, so safety and operational aspects of water chemistry performance are not affected by the TPO.

Table 3-1
Adjusted Reference Temperatures – 40 Year Life (27 EFPY)

Pilgrim TPO **Beltline Plate and Weld ART Values for 27 EFPY**

Plate

Thickness = 5 53 inches

27 EFPY Peak I D fluence = 1 3E+18 n/cm^2
27 EFPY Peak 1/4 T fluence = 9 1E+17 n/cm^2
27 EFPY Peak 1/4 T fluence = 9 1E+17 n/cm^2

Weld

Thickness = 5 53 inches

27 EFPY Peak I D fluence = 1 3E+18 n/cm^2
27 EFPY Peak 1/4 T fluence = 9 1E+17 n/cm^2
27 EFPY Peak 1/4 T fluence = 9.1E+17 n/cm^2

COMPONENT	HEAT OR HEAT/LOT	%Cu	%Ni	CF	Initial RTndt °F	1/4 T Fluence n/cm^2	27 EFPY Δ RTndt °F	σ_I	σ_Δ	Margin °F	27 EFPY Shift °F	27 EFPY ART °F
PLATES:												
Lower Shell												
G-3109-2	C-2957-2	0.10	0 47	65	0	6 8E+17	22	0	11	22	45	45
G-3109-1	C-2957-1	0 10	0 48	65	-3	6 8E+17	22	0	11	22	45	42
G-3109-3	C-2973-1	0 11	0 63	74	-4	6 8E+17	26	0	13	26	51	47
Lower-Intermediate Shell												
G-3108-3	C-2945-2	0 10	0 66	65	-12	9 1E+17	26	0	13	26	52	40
G-3108-1	C-2921-2	0 14	0 60	100	-30	9 1E+17	40	0	17	34	74	44
G-3108-2	C-2945-1	0 10	0 65	65	-7	9 1E+17	26	0	13	26	52	45
WELDS:												
Vertical Welds: Lower Shell												
2-338A	27204 Linde 1092 #3714	0 203	1 018	227	-34	3 8E+17	57	0	28	56	113	79
2-338B	27204 Linde 1092 #3714	0 203	1 018	227	-34	5.3E+17	69	0	28	56	125	91
2-338C	27204 Linde 1092 #3714	0 203	1 018	227	-34	4 0E+17	58	0	28	56	114	80
Vertical Welds: Lower-Int. Shell												
1-338A	27204/12008 Linde 1092	0 219	0 996	231	-48	8 5E+17	89	0	28	56	145	97
1-338B	27204/12008 Linde 1092	0 219	0 996	231	-48	3 6E+17	56	0	28	56	112	64
1-338C	27204/12008 Linde 1092	0.219	0 996	231	-48	8.5E+17	89	0	28	56	145	97
Girth:												
1-344	21935 Linde 1092 #3869	0.183	0 704	172	-50	6 8E+17	59	0	28	56	115	65

Table 3-2
Reactor Vessel Fatigue Usage Factors of Limiting Components

	P + Q Stress (ksi)		Fatigue Usage Factor	
Component	**Current**	**TPO**	**Current**	**TPO (40 yr)**
Recirculation Outlet Nozzle	34.20	34.78	0.720	0.747
Feedwater Nozzle	57.1 / 16.2*	58.6 / 16.6*	U < 0.80	U < 1.0
CRD Nozzle – Stub Tube	54.70	57.44	0.800	0.870

* Second value has thermal bending stress removed.

4.0 ENGINEERED SAFETY FEATURES

4.1 CONTAINMENT SYSTEM PERFORMANCE

TLTR Appendix G presents the methods, approach, and scope for the TPO uprate containment evaluation for LOCA. Except for subcompartment pressurization analysis, the previous containment evaluations are bounding for TPO uprate because they were performed at 102% of CLTP. The annulus pressurization due to TPO was calculated to increase by less than 0.1% from CLTP conditions, which is considered negligible. The methodology and results of previous analyses have been reported in previous PNPS licensing documentation. Although the nominal operating conditions change slightly because of the TPO uprate, the required initial conditions for containment analysis inputs remain the same as previously documented.

The following table summarizes the effect of the TPO uprate on various aspects of the containment system performance.

<table>
<tr><th>Topic</th><th>Key Parameters</th><th>TPO Effect</th></tr>
<tr><td>Short Term Pressure and Temperature Response</td><td></td><td rowspan="11">Current Analysis
Based on 102% of CLTP [1]</td></tr>
<tr><td>Gas Temperature</td><td>Break Flow and Energy</td></tr>
<tr><td>Pressure</td><td>Break Flow and Energy</td></tr>
<tr><td>Long-Term Suppression Pool Temperature Response</td><td></td></tr>
<tr><td>Bulk Pool</td><td>Decay Heat</td></tr>
<tr><td>Local Temperature with SRV Discharge</td><td>Decay Heat</td></tr>
<tr><td>Containment Dynamic Loads</td><td></td></tr>
<tr><td>Loss-of-Coolant Accident Loads</td><td>Break Flow and Energy</td></tr>
<tr><td>Safety-Relief Valve Loads</td><td>Decay Heat</td></tr>
<tr><td>Containment Dynamic Loads</td><td></td><td></td></tr>
<tr><td>Subcompartment Pressurization</td><td>Break Flow and Energy</td><td>Negligible increase</td></tr>
<tr><td>Containment Isolation</td><td></td><td>The ability of containment isolation valves and operators to perform their required functions are not affected because existing evaluations have been performed at 102% of CLTP.</td></tr>
</table>

[1] SRV modifications increase the SRV flow rate.

4.1.1 Generic Letter 89-10 Program

The motor-operated valve (MOV) requirements of GL 89-10 were reviewed, and no changes to the functional requirements of GL 89-10 were identified as a result of operating at the TPO RTP. Because the previous analyses that are reactor pressure dependent were performed using the reactor SRV pressure settings, there are no increases in the pressure or temperature at which MOVs are required to operate. Therefore, the GL 89-10 MOVs remain capable of performing their design basis function.

4.1.2 Generic Letter 95-07 Program

The commitments relating to the GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," have been reviewed and no changes are identified as a result of operating at the TPO RTP level. The process parameters used in the screening criteria and valve-specific analyses do not change as a result of the TPO uprate. Therefore, these valves remain capable of performing their design basis function at TPO conditions.

4.1.3 Generic Letter 96-06

The PNPS response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," was reviewed for the TPO uprate. The containment design temperatures and pressures in the current GL 96-06 evaluation are not exceeded under post-accident conditions for the TPO uprate. Therefore, the PNPS response to GL 96-06 remains valid under TPO uprate conditions.

4.2 EMERGENCY CORE COOLING SYSTEMS

4.2.1 High Pressure Coolant Injection

The HPCI system is a turbine driven system designed to pump water into the reactor vessel over a wide range of operating pressures. For the TPO uprate, there is no change to the normal reactor operating pressure or the SRV setpoints. The primary purpose of the HPCI is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. The generic evaluation of the HPCI system provided in Section 5.6.7 of the TLTR is applicable to PNPS. The ability of the HPCI system to perform required safety functions is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the HPCI system are within previous evaluations and the requirements are unchanged for TPO uprate conditions.

4.2.2 High Pressure Core Spray

The High Pressure Core Spray (HPCS) system is not applicable to PNPS.

4.2.3 Core Spray and Low Pressure Core Spray

The Low Pressure Core Spray (LPCS) system is not applicable to PNPS.

The Core Spray (CS) system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant makeup during a large break LOCA or any small break LOCA after the reactor vessel has depressurized. It also provides spray cooling for long-term core cooling in the event of a LOCA. The generic evaluation of the CS system provided in TLTR Section 5.6.10 is applicable to PNPS. The ability of the CS system to perform required safety functions is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the CS system are within previous evaluations and the requirements are unchanged for TPO uprate conditions.

4.2.4 Low Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to provide reactor vessel coolant makeup during a large break LOCA or any small break LOCA after the reactor vessel has depressurized. The generic evaluation of the LPCI mode provided in TLTR Section 5.6.10 is applicable to PNPS. The ability of the RHR system to perform required safety functions required by the LPCI mode is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the RHR system LPCI mode are within previous evaluations and the requirements are unchanged for TPO uprate conditions.

4.2.5 Automatic Depressurization System

The Automatic Depressurization System (ADS) uses relief or safety/relief valves to reduce the reactor pressure following a small break LOCA when it is assumed that the high pressure systems have failed. This allows the CS and LPCI to inject coolant into the reactor vessel. The ADS initiation logic and valve control is not affected by the TPO uprate. The generic evaluation of the ADS provided in TLTR Section 5.6.8 is applicable to PNPS. The ability of the ADS system to perform required safety functions is demonstrated with previous analyses based on 102% of CLTP. For the TPO uprate, the capacity of the existing SRVs was not sufficient to meet the additional overpressure protection requirements. Therefore, modifications were made to increase the flow capacity by ~7.5%, which provides more than adequate overpressure protection for TPO operating conditions. Operational pressure setpoints do not change, ensuring adequate simmer margin during TPO uprate operation. Therefore, all safety aspects of the ADS are within previous evaluations and the requirements are unchanged for TPO uprate conditions.

4.2.6 ECCS Net Positive Suction Head

The most limiting case for net positive suction head (NPSH) typically occurs at the peak long-term suppression pool temperature. The generic evaluation of the containment provided in TLTR Appendix G is applicable to PNPS. Because previous containment analyses were based on 102% of CLTP, there is no change in the available NPSH for systems using suppression pool water. Therefore, the TPO does not affect compliance to the ECCS pump NPSH requirements.

4.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The current 10 CFR 50, Appendix K LOCA analysis for PNPS has been performed at 102% of CLTP.

TLTR Appendix D, Table D-1 shows conservative estimates of the increase in PCT for a TPO uprate of 1.5%

[Redacted]

Because the pre-TPO SAFER/GESTR-LOCA analysis did not have sufficient margin to the statistical Upper Bound PCT limit of 1600°F, a plant specific analysis for PNPS was performed at the TPO RTP level. The TPO analysis was performed with the same ECCS performance parameters as the pre-TPO analysis. The results of the analysis show that the Upper Bound PCT increase would be less than 1°F with the TPO uprate.

4.4 MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM

The Main Control Room (MCR) atmosphere is minimally affected by the TPO uprate. An increase in rated reactor power of 1.5% would increase the estimated dose to the MCR occupants by ~1.5%. The current design basis analyses for accident dose accumulation to the MCR operators have been reviewed. Increasing the calculated dose values by 1.5% would not exceed the MCR habitability limits in 10 CFR 50, Appendix A, GDC 19. Therefore, the system remains capable of performing its safety function for the TPO uprate.

4.5 STANDBY GAS TREATMENT SYSTEM

The Standby Gas Treatment System (SGTS) minimizes the offsite and control room dose rates during venting and purging of the containment atmosphere under abnormal conditions. The current capacity of the SGTS was selected to maintain the secondary containment at a slightly negative pressure during such conditions. This capability is not changed by TPO uprate conditions. The SGTS charcoal beds can accommodate DBA conditions from 102% of CLTP. Therefore, the system remains capable of performing its safety function for the TPO uprate.

4.6 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

PNPS does not have an MSIV gland leakage system. The gland leak connections have been capped to seal them.

4.7 POST-LOCA COMBUSTIBLE GAS CONTROL SYSTEM

The Combustible Gas Control System (CGCS) maintains the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the flammability limit. The generic evaluation of the CGCS provided in TLTR Section J.2.3.10 is applicable to PNPS. The metal available for reaction is unchanged by the TPO uprate and the hydrogen production due to radiolytic decomposition is unchanged because the system was previously evaluated for accident conditions from 102% of CLTP. Therefore, the current evaluation is valid for the TPO uprate.

5.0 INSTRUMENTATION AND CONTROL

5.1 NSSS MONITORING AND CONTROL

The instruments and controls that directly interact with or control the reactor are usually considered within the NSSS. The NSSS process variables and instrument setpoints that could be affected by the TPO uprate were evaluated.

5.1.1 Neutron Monitoring System

5.1.1.1 Average Power Range Monitors, Intermediate Range Monitors, and Source Range Monitors

The Average Power Range Monitors (APRMs) are re-calibrated to indicate 100% at the TPO RTP level of 2028 MWt. The APRM high flux scram and the upper limit of the rod block setpoints, expressed in units of percent of licensed power, are not changed. The flow-biased APRM trips, expressed in units of absolute thermal power (i.e., MWt), remain the same. Thus, the MCPR reduction or maximum LHGR ratio to the limiting value is unchanged for potential transient increases of power from the operating limit to the APRM rod block alarm or flow-referenced scram trip. This approach for the PNPS TPO uprate follows the guidelines of TLTR Section 5.6.1 and Appendix F, which is consistent with the practice approved for GE BWR uprates in ELTR1 (Reference 2).

For the TPO uprate, no adjustment is needed to ensure the Intermediate Range Monitors (IRMs) have adequate overlap with the Source Range Monitors (SRMs) and APRMs. However, normal plant surveillance procedures may be used to adjust the IRMs overlap with the SRMs and APRMs. The IRM channels have sufficient margin to the upscale scram trip on the highest range when the APRM channels are reading near their downscale alarm trip because the change in APRM scaling is so small for the TPO uprate.

5.1.1.2 Local Power Range Monitors and Traversing Incore Probes

At the TPO RTP level, the flux at some Local Power Range Monitors (LPRMs) increases. However, the small change in the power level is not a significant factor to the neutronic service life of the LPRM detectors and radiation level of the traversing incore probes (TIPs). It does not change the number of cycles in the lifetime of any of the detectors. The LPRM accuracy at the increased flux is within specified limits, and the LPRMs are designed as replaceable components. The TIPs are stored in shielded rooms and a small increase in radiation levels can be accommodated by the radiation protection program for normal plant operation.

The rod block monitor (RBM) instrumentation is referenced to an APRM channel. Because the APRM has been rescaled, there is only a small effect on the RBM performance due to the LPRM performance at the higher average local flux. The RBM instrumentation is not significantly affected by TPO uprate conditions, and no change is needed.

5.1.2 Rod Worth Minimizer

The Rod Worth Minimizer (RWM) does not perform a safety-related function. The function of the RWM is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. The power-dependent setpoints for the RWM are included in Section 5.3.8.

5.2 BOP MONITORING AND CONTROL

Operation of the plant at the TPO RTP level has minimal effect on the BOP system instrumentation and control devices. The improved FW flow measurement, which is the basis for the reduction in power uncertainty, is addressed in Section 1.4. All of the control systems and instrumentation have sufficient range/adjustment capability for use at the TPO uprate conditions.

5.2.1 Pressure Control System

The Pressure Control System (PCS) provides a fast and stable response to steam flow changes so that reactor pressure is controlled within a normal operating range. The PCS consists of the pressure regulation system, turbine control valve (TCV) system, and steam bypass valve system. The main turbine speed/load control function is performed by the main turbine-generator mechanical-hydraulic control (MHC) system. The steam bypass valve pressure control function is performed by the same system.

Satisfactory reactor pressure control by the pressure regulator and TCVs requires an adequate flow margin between the TPO RTP operating condition and the steam flow capability of the TCVs at their maximum stroke (i.e., valves wide open (VWO)). Because PNPS does not have sufficient margin in the TCVs to accommodate the proposed TPO flow conditions, the HP turbine steam path will be replaced. The replacement components will provide for a reduced pressure drop, which will allow the existing TCVs to handle the new flow conditions. Minor modifications to the TCV cams may be required and will be included in the HP turbine design package. The new components will provide for a TCV design flow margin of 5%.

The existing MHC controls, as designed for the current 100% CLTP conditions, are adequate and require no component changes for the TPO uprate conditions (See Section 5.2.2).

No modification is required to the steam bypass valves. No modifications are required to the operator interface indications, controls, or alarm annunciators provided in the main control room. The required adjustments are limited to "tuning" of the control settings that may be required to operate optimally at the TPO uprate power level.

PCS tests, consistent with the guidelines in Appendix L of the TLTR, will be performed during the power ascension phase (Section 10.4).

5.2.2 Turbine Control System

The Turbine Control System utilizes an MHC system consisting of:

- Normal governing devices (two initial pressure regulators, speed governor, and startup control devices),
- Pre-emergency devices (acceleration relay),
- Emergency devices for turbine and plant protection (overspeed governor, backup overspeed, master trip, two vacuum trips, motoring protection, thrust bearing wear detector, and electrical fault protection relays), and
- Special control and test devices.

The design basis of the MHC system is to generate coordinated positioning signals for the control, intercept, and bypass valves to control reactor pressure and turbine load. The MHC system operates the main stop valves, control valves, bypass valves, crossover combination intercept-intermediate valves, and other protective devices.

No modification is required to the MHC Turbine Control System. The required adjustments are limited to "tuning" of the control settings that may be required to operate optimally at the TPO uprate power level.

PCS tests, consistent with the guidelines in TLTR Appendix L, will be performed during the power ascension phase (Section 10.4).

5.2.3 Feedwater Control System

An evaluation of the ability of the FW/level control system and FW control valves to maintain adequate water level control at the TPO uprate conditions has been performed. The ~2% increase in FW flow associated with the TPO uprate is within the current control margin of these systems. No changes in the operating water level or water level trip setpoints are required for the TPO uprate. Per the guidelines of Appendix L of the TLTR, the performance of the FW/level control systems will be recorded at 95% and 100% of CLTP and confirmed at the TPO RTP during power ascension. These checks will demonstrate acceptable operational capability and will utilize the methods and criteria described in the original startup testing of these systems.

5.2.4 Leak Detection System

The setpoints associated with leak detection have been evaluated with respect to the ~2% higher steam flow and < 2°F increase in FW temperature for the TPO uprate. Each of the systems, where leak detection potentially could be affected, is addressed below.

Main Steam Tunnel Temperature Based Leak Detection

The < 2°F increase in FW temperature for the TPO uprate will have an insignificant effect on the leak detection trip avoidance margin, because the main steam line (MSL) temperature has more influence on area temperature. As described in Section F.4.2.8 of the TLTR, the high steam tunnel temperature setpoint remains unchanged.

RWCU System Temperature Based Leak Detection

There is no significant effect on RWCU system temperature or pressure due to the TPO uprate. Therefore, there is no effect on the RWCU temperature based leak detection.

RCIC System Temperature Based Leak Detection

The TPO uprate does not increase the nominal vessel dome pressure or temperature. Therefore, there is no change to the RCIC system temperature or pressure, and thus, the RCIC temperature based leak detection system is not affected.

HPCI System Temperature Based Leak Detection

The TPO uprate does not increase the nominal vessel dome pressure or temperature. Therefore, there is no change to the HPCI system temperature or pressure, and thus, the HPCI temperature based leak detection system is not affected.

RHR System Temperature Based Leak Detection

The TPO uprate does not increase the nominal vessel dome pressure or temperature. Therefore, there is no change to the RHR system temperature or pressure, and thus, the RHR temperature based leak detection system is not affected.

Non-Temperature Based Leak Detection

The non-temperature based leak detection systems are not affected by the TPO uprate.

5.3 TECHNICAL SPECIFICATION INSTRUMENT SETPOINTS

The determination of instrument setpoints is based on plant operating experience, conservative licensing analyses or limiting design/operating values. Standard GE setpoint methodologies (Reference 7) are used to generate the allowable values (AVs) and nominal trip setpoints (NTSPs) related to the analytical limit (AL) changes, as applicable. Each actual trip setting is established to preclude inadvertent initiation of the protective action, while assuring adequate allowances for instrument accuracy, calibration, drift, and applicable normal and accident design basis events.

In general, if the AL does not change in units shown in the Technical Specifications, then no change in its associated plant AV and NTSP is required. In some cases, changes in the AV and

NTSP instrument setting will occur in the measured units. No changes in the ALs based on results from the TPO evaluations and safety analyses are expected to occur. Changes in the setpoint margins due to changes in instrument accuracy and calibration errors caused by the change in environmental conditions around the instrument due to the TPO uprate are negligible. Maintaining constant nominal dome pressure for the uprate minimizes the potential effect on these instruments by maintaining the same fluid properties at the instruments. The setpoint evaluations are based on the guidelines in TLTR Sections 5.8 and F.4 and on Section 5.3 of Reference 7.

5.3.1 High-Pressure Scram

The high-pressure scram terminates a pressure increase transient not terminated by direct or high flux scram. Because there is no increase in nominal reactor operating pressure with the TPO uprate, the scram AL on reactor high pressure is unchanged.

5.3.2 TSV Closure Scram and TCV Fast Closure Scram Bypasses

The Turbine Stop Valve (TSV) closure scram and TCV fast closure scram bypasses allow these scram to be bypassed, when reactor power is sufficiently low, such that the scram function is not needed to mitigate a Turbine-Generator (T/G) trip. As discussed in Section 5.2.1, PNPS is replacing the HP turbine rotor and diaphragms to accommodate the increased TPO steam flow and at the same time add margin to the TCVs, which presently operate near VWO conditions. The pressure drop across the new HP turbine will decrease slightly, with a corresponding increase in the pressure drop across the TCVs for a given steam flow.

A new analysis to calculate the turbine first-stage pressure (TFSP) setpoint that enables the T/G trip scram at high power will be performed as part of the turbine replacement project to ensure the correlation between core thermal power and TFSP remains conservative for use in the 3D-monocore analysis. The TFSP setpoint for the T/G scram is primarily based on operational (trip avoidance) considerations. PNPS currently operates with a TFSP scram bypass of 112 psig in accordance with the Technical Specifications. This corresponds to 659 MWt (33% of CLTP / 32.5% of TPO RTP). It is expected that the new TFSP setpoint will remain the same value in terms of absolute main turbine steam flow (lb/hr), and indicated as a pressure signal (psig).

5.3.3 High-Pressure Recirculation Pump Trip

The anticipated transient without scram recirculation pump trip (ATWS-RPT) trips the pumps during plant transients with increases in reactor vessel dome pressure. The ATWS-RPT provides negative reactivity by reducing core flow during the initial part of an ATWS. The evaluation in Section 9.3.1 demonstrates that the current high pressure ATWS-RPT AL is acceptable for the TPO uprate.

5.3.4 Safety Relief Valve

Because there is no increase in nominal vessel dome pressure, the SRV ALs are not changed.

5.3.5 Main Steam Line High Flow Isolation

Although the MS flow increases by ~2%, the current MSL high steam flow differential pressure AL is not changed for the TPO uprate. The corresponding AL in terms of steam flow is decreased to approximately 147.5% of the TPO rated steam flow. Because of the large spurious trip margin, sufficient margin exists to allow for normal plant testing of the MSIVs and turbine stop and control valves. This approach is consistent with TLTR Section F.4.2.5.

5.3.6 Fixed APRM Scram

The fixed APRM ALs, expressed in percent of RTP, do not change for the TPO uprate. The generic evaluation and guidelines presented in TLTR Section F.4.2.2 are applicable to PNPS. The limiting transient that relies on the fixed APRM trip is the MSIV closure transient with indirect scram. As described in TSAR Section 9.1, this event has been analyzed assuming 102% of CLTP and is reanalyzed on a cycle specific basis.

5.3.7 APRM Flow-Biased Scram

The flow-referenced APRM trip and alarm setpoints are unchanged in units of absolute core thermal power versus recirculation drive flow. Because the setpoints are expressed in percent RTP, they decrease in proportion to the power uprate or CLTP RTP / TPO RTP. This is the same approach taken for generic BWR uprates described in ELTR1. There are no significant effects on the instrument errors or uncertainties from the TPO uprate.

5.3.8 Rod Worth Minimizer Low Power Setpoint

The RWM Low Power Setpoint (LPSP) is used to enforce the rod patterns established for the control rod drop accident at low power levels. The RWM LPSP for PNPS remains the same value in terms of percent RTP for the TPO.

5.3.9 Rod Block Monitor

The value in the plant Technical Specifications for the RBM power-biased setpoints is maintained the same in terms of percent RTP. Therefore, no setpoint calculation is required. The trip setpoints (corresponding to the various power-dependent setpoint levels) are evaluated as part of the cycle-specific reload analysis.

5.3.10 Low Steam Line Pressure MSIV Closure (RUN Mode)

The purpose of this function is to initiate MSIV closure on low steam line pressure when the reactor is in the RUN mode. This AL is not changed for the TPO as discussed in TLTR Section F.4.2.7.

5.3.11 Reactor Water Level Instruments

The generic discussion in TLTR Section F.4.2.10 is applicable to the PNPS TPO uprate. Use of the current ALs maintains acceptable safety system performance. The low reactor water level ALs for scram, high pressure injection, and ADS/ECCS are not changed for the TPO uprate. The high water level ALs for trip of the main turbine and FW pumps are also not changed for the TPO uprate.

Water level change during operational transients (e.g., trip of a recirculation pump, FW controller failure, loss of one FW pump) is slightly affected by the TPO uprate. The plant response following the trip of one FW pump does not change significantly, because the maximum operating rod line is not being increased. Therefore, the final power level following a single FW pump trip at TPO uprate conditions would remain the same relative to the remaining FW flow as exists at CLTP.

5.3.12 Main Steam Line Tunnel High Temperature Isolations

As noted in Section 5.2.4 above, the high steam tunnel temperature AL remains unchanged for the TPO uprate.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 AC POWER

Plant electrical characteristics are given in Table 6-1.

6.1.1 Off-Site Power

The review of the existing off-site electrical equipment concluded the following:

- The isolated phase bus duct is adequate for both rated voltage and low voltage current output.
- The main transformers and the associated switchyard components (rated for maximum transformer output) are adequate for the TPO uprate-related transformer output.

A grid stability analysis is being performed to demonstrate conformance to General Design Criteria 17 (10 CFR 50, Appendix A). GDC 17 addresses on-site and off-site electrical supply and distribution systems for safety-related components. There is no anticipated significant effect on grid stability or reliability. There are no modifications associated with the TPO uprate, which would revise the logic of the electrical distribution systems or increase electrical loads beyond the nameplate ratings of the equipment.

6.1.2 On-Site Power

The on-site power distribution system consists of transformers, numerous buses, and switchgear. Alternating current (AC) power to the distribution system is provided from the transmission system or from onsite diesel generators. The on-site power distribution system loads were reviewed under both normal and emergency operating scenarios. In both cases, loads are computed based primarily on equipment nameplate data or brake horsepower (BHP). These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. Operation at the TPO RTP level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating running kW or BHP. Therefore, there are negligible changes to the load and voltage drop values and no change to the short circuit current values.

Station loads under normal operation/distribution conditions are computed based on equipment nameplate data with conservative demand factors applied. The only identifiable change in electrical load demand is associated with condensate and FW pumps. These pumps experience increased flow and pressure due to the TPO uprate conditions. Because these changes are small, the motor demand for each of these loads remains bounded by the existing design. Accordingly, there are negligible changes in the on-site distribution system design basis loads or voltages due to the TPO conditions. The system environmental design bases are unchanged. Operation at the TPO RTP level is achieved by utilizing existing equipment operating at or below the nameplate rating; therefore, under normal conditions, the electrical supply and distribution components (e.g., switchgear, MCCs, cables) are adequate.

Station loads under emergency operation and distribution conditions (emergency diesel generators) are based on equipment nameplate data, except for the ECCS pumps where a conservatively high flow BHP is used. Emergency operation at the TPO RTP level is achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps; therefore, under emergency conditions the electrical supply and distribution components are adequate.

No increase in flow or pressure is required of any AC-powered ECCS equipment for the TPO uprate. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) does not increase, and the current emergency power system remains adequate. The systems have sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

6.2 DC POWER

The direct current (DC) loading requirements in the Updated Final Safety Analysis Report (UFSAR) were reviewed, and no reactor power-dependent loads were identified. The DC power distribution system provides control and motive power for various systems and components. In both normal and emergency operating scenarios, loads are computed based on equipment nameplate data or BHP. These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. Operation at the TPO RTP level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating running kW or BHP. Therefore, there are no changes to the load, voltage drop or short circuit current values.

6.3 FUEL POOL

The following subsections address fuel pool cooling, crud and corrosion products in the fuel pool, radiation levels, and structural adequacy of the fuel racks. The overall conclusion is that the changes due to TPO are within the design limits of the systems and components, and the fuel pool cooling system meets the UFSAR requirements at the TPO conditions.

6.3.1 Fuel Pool Cooling

The Spent Fuel Pool (SFP) heat load increases slightly as a result of operation at the TPO RTP level. The TPO uprate does not affect the heat removal capability of the Fuel Pool Cooling and Cleanup System (FPCCS). The TPO heat load is within the design basis heat load for the FPCCS, and does not result in a delay in removing the RHR Augmented Fuel Pool Cooling system from service (i.e., the outage day the FPCCS can maintain the SFP temperature below 125°F such that the Augmented Fuel Pool Cooling mode of the RHR system is not required).

The SFP cooling adequacy is determined by calculating the heat load generated by a full core discharge plus remaining spaces filled with used fuel discharged at regular intervals. The analysis assumes 24-month fuel cycle lengths and GE-11 and GE-14 fuel as the basis. This evaluation considers the expected heat load in the SFP pool after TPO operation based on NRC methodology. The Augmented Fuel Pool Cooling system has adequate margin to handle a full core offload plus the bundles from previous refuelings. The FPCCS operating parameters do not change for the TPO uprate.

The FPCCS heat exchangers are sufficient to remove the decay heat after normal refueling and following operation at the TPO RTP. The RHR Augmented Fuel Pool Cooling system is available, if needed, to maintain the SFP water temperature below design limit.

6.3.2 Crud Activity and Corrosion Products

The crud activity and corrosion products associated with spent fuel can increase very slightly due to the TPO. The increase is insignificant and SFP water quality is maintained by the FPCCS.

6.3.3 Radiation Levels

The normal radiation levels around the SFP may increase slightly during fuel handling operations. This increase is acceptable and does not significantly increase the operational doses to personnel or equipment.

6.3.4 Fuel Racks

The fuel racks are designed for higher temperatures than are anticipated from the effects of the TPO uprate. There is no effect on the design of the fuel racks, because the original design SFP temperature is not exceeded.

6.4 WATER SYSTEMS

The safety-related and non-safety-related cooling water loads potentially affected by TPO are addressed in the following sections. The environmental effects of TPO are controlled such that none of the present limits (e.g., maximum allowed cooling water discharge temperature) are increased.

6.4.1 Service Water Systems

6.4.1.1 Safety-Related Loads

The Salt Service Water (SSW) system functions as the heat sink for all the systems cooled by the Reactor Building Closed Cooling Water (RBCCW) and Turbine Building Closed Cooling Water (TBCCW) systems during all planned operations in all operating states by continuously providing adequate cooling water flow to the secondary sides of the RBCCW and TBCCW heat exchangers. The safety-related performance of the SSW system during and following the most demanding design basis accident (LOCA) does not change, because the original LOCA analysis was performed based on 102% of CLTP. There is no change in the safety-related heat loads and the requirements are within the capacity of the RHR and associated SSW system.

6.4.1.2 Non-Safety-Related Loads

As discussed above, the SSW system functions as the heat sink for all the systems cooled by the RBCCW and TBCCW systems. During all planned operations in all operating states, the SSW continuously provides adequate cooling water flow to the secondary sides of the RBCCW and TBCCW heat exchangers. During emergency conditions, most of the SSW flow is automatically diverted from the TBCCW to the RBCCW heat exchanger and its safety-related loads. Sufficient margin exists in the SSW system to ensure that normal operation under TPO conditions does not adversely affect the operation of the SSW system.

6.4.2 Main Condenser/Circulating Water/Normal Heat Sink Performance

The main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. TPO operation increases the heat rejected to the condenser and may reduce the difference between the operating pressure and the required minimum condenser vacuum. The performance of the main condenser was evaluated for operation at the TPO RTP. The evaluation is based on a design duty over the actual yearly range of circulating water inlet temperatures, and confirms that the condenser, circulating water system, and heat sink are adequate for TPO operation.

6.4.2.1 Discharge Limits

The National Pollutant Discharge Elimination System (NPDES) permit for PNPS limits operation to an absolute condenser discharge temperature of 102°F, or circulating water differential temperatures less than 32°F. PNPS occasionally needs to trim power to remain within these limits when the ocean temperatures rise to 72°F or more during the summer. PNPS remains within state discharge limits during operation at TPO conditions.

6.4.3 Reactor Building Closed Cooling Water System

The heat loads on the RBCCW system do not increase significantly due to TPO because they depend on either reactor vessel water temperature or flow rates in the systems cooled by the RBCCW. The change in reactor vessel water temperature is minimal and there is no change in nominal reactor operating pressure. The RBCCW system experiences a slight heat load increase, primarily in the Fuel Pool heat exchangers. However, the system has adequate design margin to remove the additional heat. In addition, the performance of the RBCCW system during and following the DBA-LOCA does not change, because the original LOCA analysis was performed based on 102% of CLTP. Therefore, the RBCCW system is acceptable for TPO uprate.

6.4.4 Turbine Building Closed Cooling Water System

The power-dependent heat loads on the TBCCW system that are increased by the TPO are the coolers associated with the isophase bus, turbine, and generator. The remaining TBCCW heat loads are not strongly dependent upon reactor power and do not significantly increase. The TBCCW system has sufficient capacity to assure that adequate heat removal capability is available for TPO operation.

6.4.5 Ultimate Heat Sink

The ultimate heat sink (UHS) is the Atlantic Ocean. Although TPO operation increases the amount of heat discharged to the UHS by a small amount, there is no increase in the ocean temperature as a result of operation at TPO conditions.

There are administrative limits for the use of the UHS such as allowable inlet and discharge temperatures, as well as the temperature rise between them. These limits are monitored and plant operation is adjusted as these limits are approached. Operating at TPO conditions does not change these limits, nor does it result in any change to the UHS.

6.5 STANDBY LIQUID CONTROL SYSTEM

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that all or some of the control rods cannot be inserted. It is a manually operated system that pumps a highly enriched sodium pentaborate solution into the vessel to achieve a subcritical condition. The generic evaluation presented in TLTR Sections 5.6.5 (SLCS) and L.3 (ATWS Evaluation) is applicable to the PNPS TPO uprate. The TPO uprate of 1.5% power does not affect the solution storage requirements, system injection capability, or the equivalent solution injection rate. Because the shutdown margin is reload dependent, the shutdown margin and the required reactor boron concentration are confirmed for each reload core.

The SLCS ATWS performance is evaluated in TSAR Section 9.3.1. The evaluation shows that the TPO has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 POWER DEPENDENT HEATING, VENTILATION AND AIR CONDITIONING

The Heating, Ventilation, and Air Conditioning (HVAC) systems that are potentially affected by the TPO uprate consist mainly of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, and the drywell.

TPO results in a minor increase in the heat load caused by the slightly higher FW process temperature (< 2°F). The increased heat load is within the margin of the steam tunnel area coolers. In the drywell, the increase in heat load due to the FW process temperature is within the drywell cooler capacity. In the turbine building, the maximum temperature increases in the FW heater bay and condenser areas are less than 2°F due to the increase in the FW process temperatures. In the reactor building, the increase in heat load due to a slight SFP cooling process temperature increase is within the margin of the area coolers. Other areas are unaffected by the TPO because the process temperatures and electrical heat loads remain constant.

Therefore, the power dependent HVAC systems are adequate to support the TPO uprate.

6.7 FIRE PROTECTION

Operation of the plant at the TPO RTP level does not affect the fire suppression or detection systems. There are no changes in physical plant configuration or combustible loading as a result of the TPO uprate. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the TPO uprate conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by the TPO uprate.

6.7.1 10 CFR 50 Appendix R Fire Event

Section L.4 of the TLTR presents a generic evaluation of Appendix R events for an increase of 1.5% of CLTP.

[Redacted]

The current analysis based on CLTP has an available margin of 180°F to the clad temperature limit and 49 psi to the containment pressure limit. Therefore, the generic results are clearly applicable and no further plant specific Appendix R analysis is necessary for the TPO uprate.

6.8 OTHER SYSTEMS AFFECTED BY TPO UPRATE

Based on experience and previous NRC reviews, all systems that are significantly affected by TPO are addressed in this report. Other systems not addressed by this report are not significantly affected by TPO. The systems unaffected by TPO at PNPS are confirmed to be consistent with the generic description provided in the TLTR.

Table 6-1 TPO Plant Electrical Characteristics

Parameter	Value
Guaranteed Generator Output (MWe)	709 8
Rated Voltage (kV)	24
Power Factor	0.91
Guaranteed Generator Output (MVA)	780
Current Output (kA)	18.764
Isolated Phase Bus Duct Rating:	
Main Section (kA)	20.0
Branch Section (A)	850
Main Transformers Rating (MVA)	880
TPO Uprate Transformer Output (MVA)	751

7.0 POWER CONVERSION SYSTEMS

The power conversion systems for PNPS were designed to accept the system and equipment flows resulting from continuous operation at CLTP rated steam flow with some margin. Each system was evaluated separately. Where design margin is limited, modifications to systems and components are being implemented such that all systems are able to support operation at the TPO RTP level at the VWO condition.

7.1 TURBINE-GENERATOR

The PNPS main T/G was designed with a maximum flow-passing and generator capability at rated conditions to ensure that the design rated output is achieved. Since initial plant operation in 1972, PNPS has shown that it can maintain rated output with some limited exceptions, i.e., when flow variation and power oscillations warrant that administrative controls be implemented to maintain some flow and pressure control margin.

The PNPS turbine-generator currently operates at near VWO at the design throttle steam flow of 7,974,960 lb/hr, a throttle pressure of 987.0 psia, and a design electrical power output of 697,267 kW.

Because this flow margin is insufficient to support the proposed TPO uprate, the HP turbine steam path will be replaced (See Section 5.2.1). The new design incorporates a flow margin of 5% for manufacturing tolerances and reactor pressure control margin.

For the TPO uprate RTP of 2028 MWt (101.5% of CLTP), the design throttle steam flow is increased to 8,117,000 lb/hr at a throttle pressure of 988.0 psia. The increased throttle flow is approximately 101.7% of current rated. The uprated electrical output based on the new HP turbine design is 709,028 kW. Other plant efficiency improvements will further improve plant output. The increased electrical output remains within the current capacity limits of the generator.

Calculations were performed to determine the TPO uprate turbine steam path conditions. From the thermodynamic models, turbine and generator stationary and rotating components were evaluated for increased loadings, pressure drops, thrusts, stresses, overspeed capability, and other design considerations to assure that design limits are not exceeded and that operation remains acceptable at the TPO uprate condition. In addition, valves, control systems, and other support systems were evaluated. The results of these evaluations show that for the turbine generator and auxiliaries, other than the HP steam path, only minor modifications are needed to support operation at the TPO uprate condition. These modifications will be incorporated in the new HP turbine design package.

The rotor missile analysis remains unchanged at the TPO uprate condition based on the NRC approved methodology in NUREG-1048, which applies to units with GE monoblock rotors.

Based on the calculated results of control system failure, which is on the order of 10^{-8} per year, the missile probability is acceptable.

The overspeed calculation compares the entrapped steam energy contained within the turbine and the associated piping, after the stop valves trip, and the sensitivity of the rotor train for the capability of overspeeding. The entrapped energy increases slightly for the TPO uprate conditions. Therefore, the overspeed trip settings will be changed from 110% and 111% to 110.6% and 111.6%.

7.2 CONDENSER AND STEAM JET AIR EJECTORS

The condenser capability was evaluated for performance at the TPO uprate conditions based on current circulating water system flow. The design margin in the condenser heat removal capability can accommodate the additional heat rejected for operation at the TPO uprate conditions. Operational conditions such as cleanliness, tube plugging, and circulating water temperature cause more significant variations in the condenser back pressure than the small additional TPO heat rejection.

The design of the steam jet air ejectors (SJAEs) was based on the removal of non-condensable gases produced in the reactor and air leakage into the condenser for the VWO operating conditions. Air leakage into the condenser does not increase as a result of the TPO uprate. The small increase in hydrogen and oxygen flows from the reactor does not affect the SJAE capacity because the design was based on operation at significantly greater than required flows. Therefore, the condenser air removal system is not affected by the TPO uprate and the mechanical vacuum pumps and SJAEs are adequate for operation at the TPO uprate conditions.

7.3 TURBINE STEAM BYPASS

The Steam Bypass Pressure Control System (SBPCS) was originally designed for a steam flow capacity of a nominal 25% of the 100% rated flow at CLTP. The steam bypass capacity at the TPO RTP is a nominal 25% of the 100% TPO RTP steam flow rate. The steam bypass system is a normal operating system and non-safety-related. While the bypass capacity as a percent of rated steam flow is reduced, the actual steam bypass capacity is unchanged. The transient analyses that credit the turbine bypass system availability use the actual capacity. The TPO transient analysis (Section 9.1) results are acceptable. Therefore, the turbine bypass capacity is adequate for TPO operation.

7.4 FEEDWATER AND CONDENSATE SYSTEMS

The FW and condensate systems are designed to provide FW at the temperature, pressure, quality, and flow rate required by the reactor. These systems are not safety-related; however, their performance may have an effect on plant availability and the capability to operate reliably at the TPO uprate conditions.

A review of the PNPS FW heaters, heater drains, condensate demineralizers, and pumps (FW and condensate) demonstrated that the components are capable of performing in the proper design range to provide the slightly higher TPO uprate FW flow rate at the desired temperature and pressure. The review also concluded that the FW control valves can maintain water level control at the TPO uprate conditions.

The performance evaluations were based on an assessment of the capability of the condensate and FW system equipment to remain within the design limitations of the following parameters:

- Pump NPSH
- Ability to avoid suction pressure trip
- Flow capacity
- Bearing cooling capability
- Rated motor horsepower
- Full load motor amps
- Vibration

The FW system run-out and loss of FW heating events would see very small changes from the TPO uprate as shown by the experience with substantially larger power uprates.

7.4.1 Normal Operation

The system operating flows for the TPO uprate increase approximately 2%. The three condensate pumps and three FW pumps are sized for approximately 40% flow capacity each, providing sufficient capacity to accommodate the slight increase in TPO flow. The heat exchangers were conservatively sized, with tube side flow of the most limiting FW heater increasing from 7.8 to 7.93 fps at TPO conditions. This remains well below the original guideline design limit of 10 fps. The FW regulating valves were replaced in 1999 and are sized to provide reliable operation and control. They are presently operating with a differential pressure of approximately 320 psi, and can accommodate any expected pressure change across the condensate and FW system. Adequate trip margin, during steady state conditions, exists between the calculated minimum pump suction pressure and the minimum pump suction pressure based on required NPSH. The small increase in flows ensures that TPO RTP does not significantly affect the operating conditions of the condensate and FW systems.

7.4.2 Transient Operation

To account for FW demand transients, the condensate and FW systems were evaluated to ensure sufficient margin above the TPO uprated flow is available. All three condensate and FW pumps operate at 100% CLTP. The pumps are each rated for 40% of rated flow, ensuring adequate margin for transient conditions.

Following a single FW pump trip, the reactor recirculation system would runback recirculation flow, such that the steam production rate is within the flow capacity of the remaining FW pumps. The runback setting prevents a reactor low water level scram, and is sufficient to maintain adequate margin to the potential power/flow instability region.

7.4.3 Condensate Demineralizers

There is no measurable effect on the Condensate Demineralizers (CDs) resulting from the TPO.

PNPS has a full flow CD system that was designed for the shut-off head of the condensate pumps. TPO operation results in a ~2% flow increase, and a slight reduction in the condensate system pressure as the pumps operate further out on their curves. The ion and debris loading of the condensate stream does not measurably change as a result of TPO. The CDs are routinely cleaned on an 80 to 100-day cycle, rather than on pressure drop. Therefore, no change in CD cleaning frequency is expected.

8.0 RADWASTE AND RADIATION SOURCES

8.1 LIQUID AND SOLID WASTE MANAGEMENT

The liquid radwaste system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse, discharge, or shipment.

The single largest source of liquid and wet solid waste is from cleaning/replacement of CD resins. The TPO uprate results in ~2% increased flow rate through the CDs, but is not expected to result in more frequent resin cleaning. CD resins are currently cleaned on an 80 to 100-day cycle. The resin replacement schedule of 18 months is driven by ion depletion and is not expected to change. Any slight reduction in CD service time does not affect plant safety. The RWCU filter demineralizer may also require more frequent replacements due to slightly higher levels of activation and fission products.

The floor drain collector subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. Neither subsystem experiences a significant increase in volume due to operation at the TPO uprate condition.

The activated corrosion products in the waste stream are expected to increase proportionally to the TPO uprate. However, the total volume of processed waste is not expected to increase appreciably because ionic depletion is the basis for resin changeout and TPO has no effect on that variable. A review of plant operating effluent reports leads to the conclusion that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I will continue to be met. Therefore, the TPO uprate does not adversely affect the processing of liquid radwaste and there are no significant environmental effects.

8.2 GASEOUS WASTE MANAGEMENT

The gaseous waste systems collect, control, process, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system and various building ventilation systems. The systems are designed to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

The waste gases originating in the reactor coolant consist mainly of hydrogen, oxygen, and nitrogen with trace amounts of radioactive gases. The function of the offgas system is to process these radioactive noble gases, airborne halogens, and particulates, and to reduce their activity through decay.

Reactor building ventilation systems control airborne radioactive gases by using devices such as High Efficiency Particulate Air (HEPA) and charcoal filters, and radiation monitors that activate isolation dampers or trip supply and exhaust fans, or by maintaining negative or positive air pressure to limit migration of gases. The activity of airborne effluents released through building vents does not increase significantly due to the TPO uprate because the amount of fission products released into the coolant depends on the number and nature of the fuel rod defects and is not dependent on reactor power.

The release limit is an administratively controlled variable and is not a function of core power. The gaseous effluents are well within limits at CLTP operation and remain well within limits following implementation of the TPO uprate. There are no significant environmental effects due to the TPO uprate.

The offgas system was evaluated for the TPO uprate, including the effects of hydrogen water chemistry (HWC) and noble metal injection. Radiolysis of water in the core region, which forms H_2 and O_2, increases linearly with core power, thus increasing the heat load on the recombiner and related components. The offgas system is conservatively designed for 90 scfm of hydrogen and 45 scfm of oxygen from radiolytic decomposition of water. At 100% CLTP, the flows are 58.7 scfm and 34.6 scfm, respectively. Implementation of HWC reduces both hydrogen and oxygen flows. The increases in H_2 and O_2 due to the TPO uprate remain well within the capacity of the system. The system radiological release rate is administratively controlled, and is not changed with operating power. Therefore, the TPO uprate does not affect the offgas system design or operation.

8.3 RADIATION SOURCES IN THE REACTOR CORE

TLTR Appendix H describes the methodology and assumptions for the evaluation of radiological effects for the TPO uprate.

During power operation, the radiation sources in the core are directly related to the fission rate. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy released per unit of reactor power. Therefore, for TPO, the percent increase in the operating source terms is no greater than the percent increase in power.

The post-operation radiation sources in the core are primarily the result of accumulated fission products. Two separate forms of post-operation source data are normally applied. The first is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This source term is defined in terms of MeV/sec per watt of reactor thermal power (or equivalent) at various times after shutdown. Therefore, the total gamma energy source increases in proportion to reactor power.

The second set of post-operation source data consists primarily of nuclide activity inventories for fission products in the fuel. These are needed for post-accident and spent fuel pool evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The core fission product inventories for these evaluations are based on an assumed fuel irradiation time, which develops "equilibrium" activities in the fuel (typically three years). Most radiologically significant fission products reach equilibrium within a 60-day period. The calculated inventories are approximately proportional to core thermal power. Consequently, for TPO, the inventories of those radionuclides, which reached or approached equilibrium, are expected to increase in proportion to the thermal power increase. The inventories of the very long-lived radionuclides, which did not approach equilibrium, are both power and exposure dependent. They are expected to increase proportionally with power if the fuel irradiation time remains within the current basis. Thus, the long-lived radionuclides are expected to increase proportionally to power. The radionuclide inventories are provided in terms of Curies per Megawatt of reactor thermal power at various times after shutdown.

[Redacted]

8.4 Radiation Sources in Reactor Coolant

8.4.1 Coolant Activation Products

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation is the dominant source in the turbine building and in the lower regions of the drywell. Because these sources are produced by interactions in the core region, their rates of production are proportional to power. As a result, the activation products, observed in the reactor water and steam, increase in approximate proportion to the increase in thermal power. The activation products in the steam and coolant are bounded by the existing design basis concentration.

8.4.2 Activated Corrosion and Fission Products

The reactor coolant contains activated corrosion products from metallic materials entering the water and being activated in the reactor region. Under the TPO uprate conditions, the FW flow increases with power, the activation rate in the reactor region increases with power, and the filter efficiency of the CDs may decrease as a result of the FW flow increase. The net result may be an increase in the activated corrosion product production.

Fission products in the reactor coolant are separable into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. The noble gases released during plant operation result from the escape of minute fractions of the fission products from the fuel rods. This activity is the noble gas offgas that is included in PNPS design. The offgas rates for TPO uprate operations are well below the original design basis. Therefore, the design basis release rates are bounding for the TPO uprate.

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. As is the case for the noble gases, the releases from the fuel increase approximately proportional to the TPO power increase. Activity levels in the reactor water are expected to be higher than previous calculated data due to the TPO uprate.

Although the activated corrosion product and fission product activities increase in approximate proportion to the TPO power increase, the sum of the total activated corrosion product activity and the total fission product activity due to the TPO uprate remain a fraction of the original design basis activity in the reactor water. Therefore, the activated corrosion product and fission product activities design bases in the reactor water are unchanged for the TPO uprate.

Moisture carryover from the reactor may change during some portions of the fuel cycle due to radial power distribution and core flow. The effect will be minimized by limiting radial peaking factors at lower core flows. Any potential change to turbine building radiation levels resulting in exposure to plant personnel will be controlled by the plant As Low As is Reasonably Achievable (ALARA) Program.

8.5 Radiation Levels

Normal operation radiation levels increase slightly for the TPO uprate. PNPS was designed with substantial conservatism for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the design, source terms, and analytical techniques.

Post-operation radiation levels in most areas of the plant increase by no more than the percentage increase in power level. In a few areas near the SFP cooling system piping and the reactor water piping, where accumulation of corrosion product crud is expected, as well as near some liquid radwaste equipment, the increase could be slightly higher. Regardless, individual worker exposures will be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas. Procedural controls compensate for increased radiation levels.

The change in core activity inventory resulting from the TPO uprate (Section 8.3) increases post-accident radiation levels by no more than approximately the percentage increase in power level. The slight increase in the post-accident radiation levels has no significant effect on the plant or the habitability of the Technical Support Center or Emergency Operations Facility. A review of areas requiring post-accident occupancy (per NUREG-0737 Item II.B) concluded that access needed for accident mitigation is not significantly affected by the TPO uprate.

8.6 NORMAL OPERATION OFF-SITE DOSES

As discussed in Section 8.2, the normal operation gaseous activity levels remain essentially unchanged for the TPO uprate. The Technical Specification limits implement the guidelines of 10 CFR 50, Appendix I. A review of the normal radiological effluent doses shows that at CLTP, the annual doses are less than 1% of the doses allowed by Technical Specification limits. The TPO uprate does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium or liquid effluents. In addition, radiation from shine is not a significant exposure pathway. Present offsite radiation levels are a negligible portion of background radiation. Therefore, the normal offsite doses are not significantly affected by operation at the TPO RTP level and remain below the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

9.1 ANTICIPATED OPERATIONAL OCCURRENCES

TLTR Appendix E provides a generic evaluation of the AOOs for TPO uprate plants.

[Redacted]

Also included are the analytical methods to be used and operating conditions to be assumed. The AOO events are organized into two major groups: Fuel Thermal Margin Events and Transient Overpressure Events.

TLTR Table E-2 illustrates the effect of a 1.5% power uprate on the OLMCPR.

[Redacted]

The overpressure event is currently performed with the assumption of 2% overpower. Therefore, the overpressure event is bounding for the TPO uprate. The loss of FW transient was evaluated up to 102% of CLTP and the evaluation showed acceptable margin to the safety criterion.

The reload transient analysis includes the worst overpressure event, which is usually the closure of all MSIVs with high neutron flux scram.

The evaluations and conclusions of Appendix E are applicable to the PNPS TPO uprate. Therefore, it is sufficient for the plant to perform the standard reload analyses at the first fuel cycle that implements the TPO uprate.

9.2 DESIGN BASIS ACCIDENTS

The radiological consequences of a DBA are basically proportional to the quantity of radioactivity released to the environment. This quantity is a function of the fission products released from the core as well as the transport mechanisms from the core to the release point. The radiological releases at the TPO RTP are generally expected to increase in proportion to the core inventory increase, which is in proportion to the power increase.

Radiological consequences due to postulated DBA events have been evaluated and analyzed to show that NRC regulations are met for 2% above the CLTP. Therefore, the radiological consequences associated with a postulated DBA from TPO uprate conditions are bounded by these analyses. The evaluation/analysis was based on the methodology, assumptions, and analytical techniques described in the RGs, the Standard Review Plan (SRP), where applicable, and in previous Safety Evaluations (SEs).

9.3 Special Events

9.3.1 Anticipated Transient Without Scram

PNPS meets the ATWS mitigation equipment requirements defined in 10 CFR 50.62:

1. Installation of an Alternate Rod Insertion (ARI) system.
2. Boron injection equivalent to 86 gpm.
3. Installation of automatic RPT logic (i.e., ATWS-RPT).

There are no changes to the operating pressure or maximum rod line for the TPO uprate. The performance characteristics of the equipment do not change because operating conditions do not change.

The PNPS-specific analysis at the CLTP demonstrates that the following ATWS acceptance criteria are met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig.
2. Peak clad temperature within the 10 CFR 50.46 limit of 2200°F.
3. Peak clad oxidation within the requirements of 10 CFR 50.46.
4. Peak suppression pool temperature less than 185°F.
5. Peak containment pressure less than the containment design pressure of 62 psig.

TLTR Section 5.3.5, TLTR Appendix L, and the GE response to an NRC Request for Additional Information (RAI) on the TLTR (Reference 8) present a generic evaluation [Redacted] of an ATWS to a change in power typical of the TPO uprate. The evaluation was based on [Redacted] For a TPO uprate, if a plant has sufficient margin for the projected changes in peak parameters given in TLTR Section L.3.5 as augmented by Reference 8,

[Redacted]

PNPS currently has a margin of 17°F to the pool temperature limit. This margin is well in excess of the [Redacted] defined in TLTR Appendix L and Reference 8. Therefore, no PNPS-specific ATWS analysis for suppression pool temperature was performed for the TPO uprate. However, PNPS does not have sufficient margin to the ASME Service Level C peak vessel bottom pressure limit of 1500 psig at CLTP to apply the generic criteria stated in TLTR Appendix L and Reference 8. Therefore, a plant specific ATWS analysis was performed for the TPO uprate. The key inputs to the ATWS analysis are provided in Table 9-5.

The ATWS analysis was performed as discussed in Section L.3 of ELTR1. The analyzed events have been shown to be the limiting events for ATWS calculations. The limiting case was a PRFO event, initiated at the beginning of cycle (BOC) conditions. As shown in Figure 9-1, the calculated peak vessel bottom pressure is 1495 psig for the TPO uprate. This result meets the above ATWS acceptance criteria. Therefore, the plant pressure response to an ATWS event at the TPO conditions is acceptable.

9.3.2 Station Blackout

TLTR Appendix L provides a generic evaluation of a potential loss of all alternating current power supplies based on previous plant response and coping capability analyses for typical power uprate projects. The previous power uprate evaluations have been performed according to the applicable bases for the plant (e.g., the bases, methods, and assumptions of RG 1.155 and/or NUMARC 87-00). This evaluation is for confirmation of continued compliance to 10 CFR 50.63, "Loss of all alternating current power." It is recognized that this evaluation is dependent upon many plant-specific design and equipment parameters.

Specifically, the following main considerations were evaluated:

- The adequacy of the condensate/reactor coolant inventory.
- The capacity of the Class 1E batteries.
- The Station Blackout (SBO) compressed Nitrogen requirements.
- The ability to maintain containment integrity.
- The effect of loss of ventilation on rooms that contain equipment essential for plant response to an SBO event.

Applicable operator actions have previously been assumed consistent with the plant Emergency Procedure Guidelines. These are the currently accepted procedures for each plant and SBO analysis. For the TPO uprate, there is no significant change in the time available for the operator to perform these assumed actions.

[Redacted] PNPS currently has margins of 15,000 gallons to the available condensate storage inventory volume and 50°F to the containment peak temperature limit. These margins are well in excess of the [Redacted] defined in TLTR Appendix L. Therefore, no PNPS-specific SBO analysis is performed for the TPO uprate.

Table 9-1 Key Inputs for ATWS Analysis

ATWS Input Variable	Baseline Condition Value	TPO Condition Value
Reactor power (MWt)	1998	2028
Core Flow (% Rated)	75	76.7
Reactor dome pressure (psia)	1050	1050
SRV opening setpoint pressure (psig) [1]	1126	1126 (3 valves), 1136 (1 valve) [3]
Total SRV Capacity at 1080 psig (Mlbm/hr) [2]	3.21	3.45
High pressure ATWS-RPT Technical Specification Limit (psig)	1180	1180
Number of SRVs Out of Service (OOS)	0	0

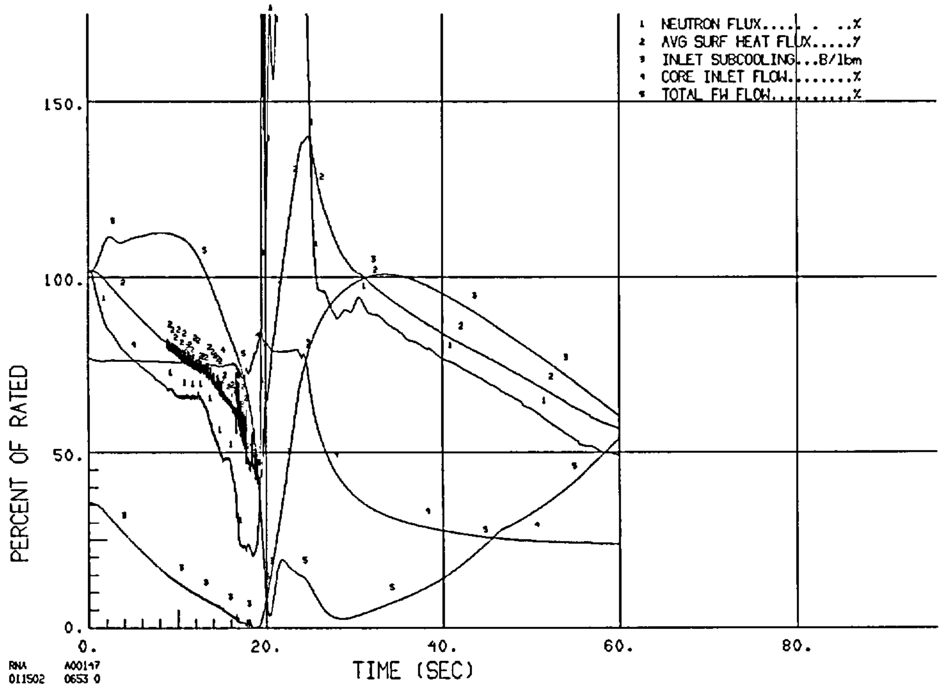
Notes:

(1) Technical Specification Limit values are shown (A conservative 22 psi drift/uncertainty allowance is applied to nominal values).

(2) The increase in total SRV capacity at the TPO condition is a result of the increase in the existing SRV throat diameters.

(3) For the TPO ATWS analysis, one SRV was assumed to have a lift setpoint of 1136 psig. This assumption provides conservative results for the ATWS pressurization events, i.e., a slightly higher peak pressure. PNPS is not changing the actual SRV setpoints as part of the TPO uprate.

Figure 9-1 Limiting ATWS Event (PRFO at BOC)



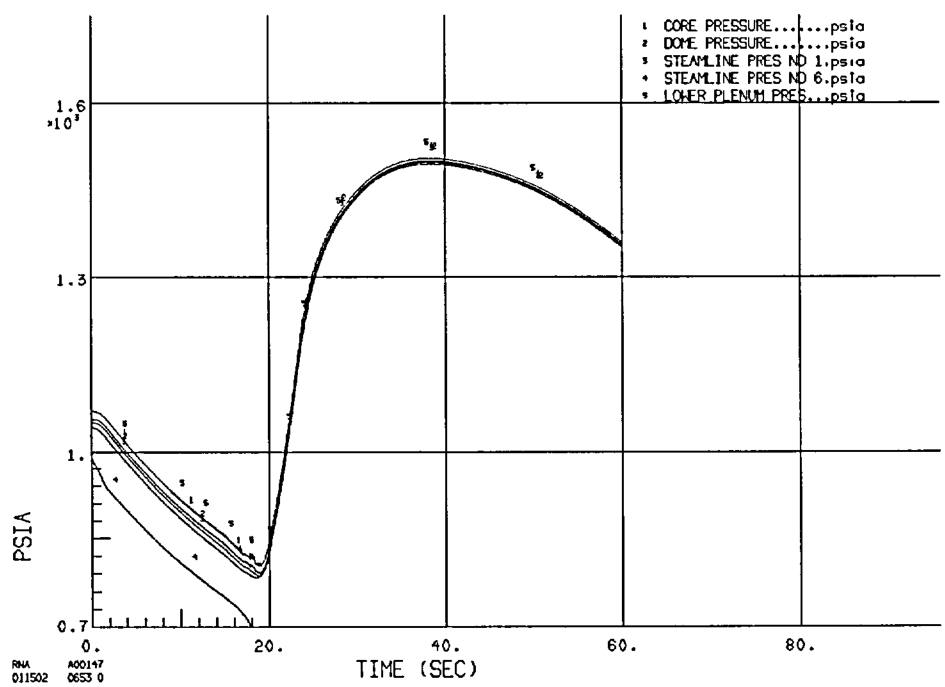
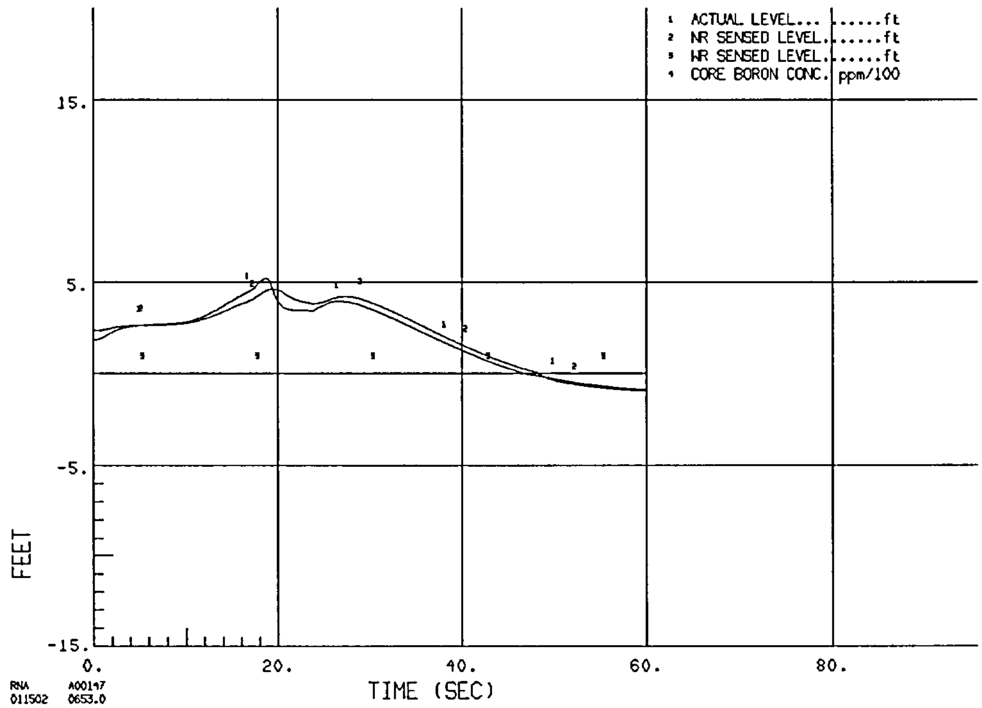
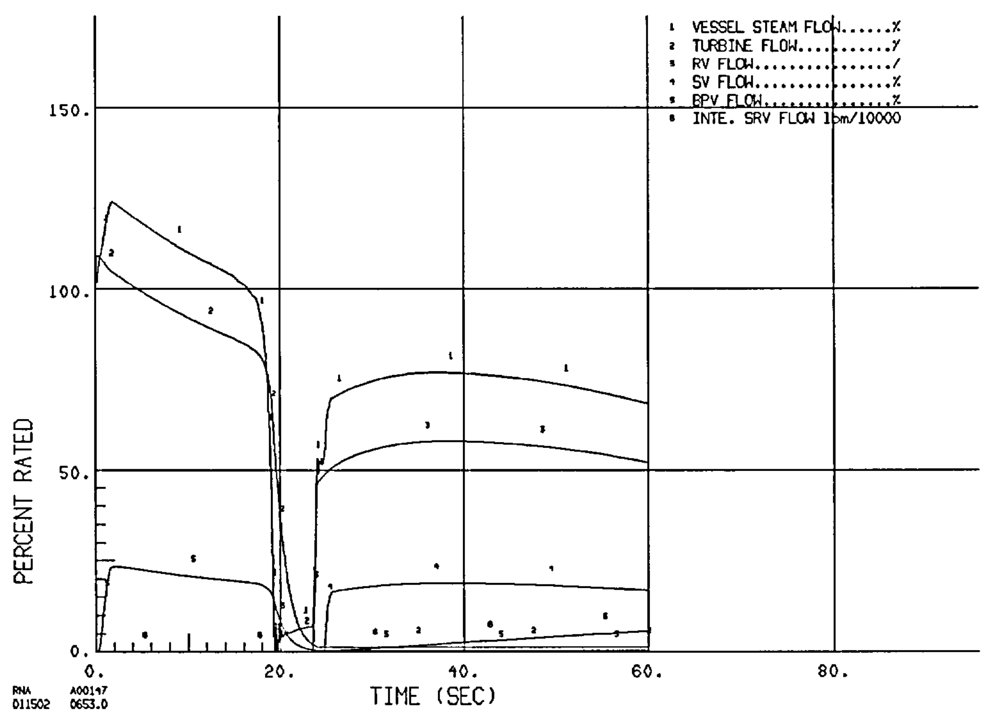


Figure 9-1 Limiting ATWS Event (PRFO at BOC) (Continued)





10.0 OTHER EVALUATIONS

10.1 HIGH ENERGY LINE BREAK

Because the TPO uprate system operating temperatures and pressures change only slightly, there is no significant change in High Energy Line Break (HELB) mass and energy releases. The FW lines, near the pump discharge, increase $< 2°F$ and < 5 psi. The recirculation lines decrease $< 1°F$ and increase < 1 psi due to the slightly higher core pressure drop. Vessel dome pressure and other portions of the RCPB remain at current operating pressure or lower. Therefore, the consequences of any postulated HELB would not significantly change. The postulated break locations remain the same because the piping configuration does not change due to the TPO uprate.

The HELB analysis evaluation was made for all systems evaluated in the UFSAR. Ten specific break locations were identified that govern all HELBs outside the containment for the purposes of determining design basis sub-compartment pressure and temperature profiles. For all of the governing HELBs, the current design basis analyses remain bounding for the TPO uprate conditions. A brief description of systems with governing HELBs follows.

10.1.1 Steam Line Breaks

The critical parameter affecting the high energy steam line break analysis is the system pressure. The PNPS design basis analyses for MS, HPCI, and RCIC steam line breaks use the reactor vessel dome pressure to calculate blowdown. Because there is no nominal reactor vessel dome pressure increase for the TPO, the current design basis analyses for high energy steam line breaks remain bounding for the TPO uprate conditions.

10.1.2 Liquid Line Breaks

10.1.2.1 Feedwater Line Breaks

The failure of an FW line is less severe than the failure of an MS line. Therefore, the current design basis analysis remains bounding for the TPO uprate conditions.

10.1.2.2 ECCS Line Breaks

Steam line breaks in the HPCI pump/turbine room and the MS tunnel are the limiting breaks for structural design and equipment qualification. As discussed in Section 10.1.1, the current design basis analyses for high energy steam line breaks remain bounding for the TPO uprate conditions.

The other ECCS lines are normally isolated from the reactor vessel, and a failure of one of these lines would result in a non-limiting break inside drywell, which would be bounded by other line breaks. Because these lines are normally isolated, the TPO uprate does not affect their line break analyses for breaks outside drywell.

10.1.2.3 RCIC System Line Breaks

Steam line breaks in the RCIC pump/turbine room are the limiting breaks for structural design and equipment qualification. As discussed in Section 10.1.1, the current design basis analyses for high energy steam line breaks remain bounding for the TPO uprate conditions.

10.1.2.4 RWCU System Line Breaks

The RWCU system line breaks are the limiting breaks for structural design and equipment qualification in several areas of the plant. A detailed review of the design basis calculations for a RWCU line break was performed to evaluate the effect of TPO uprate conditions. This review concluded that the current design basis analyses remain bounding for the TPO uprate conditions.

10.1.2.5 CRD System Line Breaks

The CRD pipe rupture analysis is not affected by the TPO uprate.

10.1.2.6 Building Heating Line Breaks

Building heating lines are not connected to the reactor-turbine primary loop. Therefore, building heating lines are not affected.

10.1.2.7 Pipe Whip and Jet Impingement

Because there is no change in the nominal vessel dome pressure, pipe whip and jet impingement loads do not significantly change. Existing calculations supporting the dispositions of potential targets of pipe whip and jet impingement from postulated HELBs have been reviewed and determined to be adequate for the safe shutdown effects in the TPO RTP conditions. Existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the TPO uprate conditions.

10.1.2.8 Internal Flooding from HELB

The effects of flooding due to a postulated HELB are not increased by the TPO uprate. Minor increases in pressure and temperature of high-energy lines remain below design values. In addition, operational modes for the systems that contain high-energy lines are not affected by the TPO uprate. The plant internal flooding analysis and safe shutdown analysis are not affected.

10.2 MODERATE ENERGY LINE BREAK

A Moderate Energy Line Break (MELB) break analysis is not within PNPS licensing basis, and is not required for the TPO uprate.

10.3 ENVIRONMENTAL QUALIFICATION

Safety-related components must be qualified for the environment in which they operate. The TPO 1.5% increase in power level increases the radiation levels experienced by equipment during normal operation and accident conditions. Because the TPO uprate does not increase the nominal vessel dome pressure, there is a very small effect on pressure and temperature conditions experienced by equipment during normal operation and accident conditions. The resulting environmental conditions are bounded by the existing environmental parameters specified for use in the environmental qualification program.

10.3.1 Electrical Equipment

Power uprate issues related to environmental qualification of electrical equipment are currently under review. Any increases in environmental parameters as a result of power uprate will be reviewed and incorporated into Specification E-536 and applicable Environmental Qualification (EQ) documents.

The safety-related electrical equipment is being reviewed to ensure that the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate. Conservatisms in accordance with IEEE 323 were originally applied to the environmental parameters, and minimal change is anticipated for the TPO uprate.

10.3.1.1 Inside Containment

EQ for safety-related electrical equipment located inside the containment is based on a Main Steam Line Break Accident (MSLBA) and/or DBA-LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. The current accident conditions for temperature and pressure are based on analyses initiated from ~102% CLTP. Normal temperatures may increase slightly (< 2°F) near the FW and reactor recirculation lines, but is not expected to affect area temperature profiles for EQ. The current radiation levels under normal plant conditions also increase slightly. The current plant environmental envelope for maximum accident radiation levels from MSLBA and/or DBA-LOCA is based on the CLTP. Any changes resulting from the TPO uprate will be evaluated as part of the routine design change process for core reload analysis and licensing, which will be completed prior to operating at the TPO uprate condition.

10.3.1.2 Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from an MSL break in the steam tunnel, or other HELBs (i.e., pipe breaks outside containment (PBOCs)), whichever is limiting for each area. Some of the HELB (PBOC) pressure and temperature profiles increase by a small amount due to the TPO uprate conditions. However, there is expected to be adequate margin in the qualification envelopes to accommodate the small changes. Maximum accident radiation levels used for qualification of equipment outside containment are based on the CLTP. Any changes resulting from the TPO uprate will be evaluated as part of the routine design change process for core reload analysis and licensing, which will be completed prior to operating at the uprated TPO condition.

Reevaluation of equipment qualification for the TPO uprate may identify some equipment potentially affected by TPO conditions. The qualification of this equipment will be resolved by reanalysis, by refined radiation calculations (location-specific), by slightly reduced qualified life, or by performing additional tests/analyses to support qualification.

As stated in Section 4.1, the containment loads for the TPO uprate are bounded by previous analyses. The effects of increased fluid induced loads on safety-related components are described in Section 3.5. Increased nozzle loads and component support loads due to the revised operating conditions were evaluated in the piping assessments in Section 3.5. These increased loads are insignificant, and become negligible when combined with the dynamic loads; except for loads associated with the SRVs, which currently are being reanalyzed; ensuring the mechanical components and component supports are adequately designed for the TPO uprate conditions.

10.4 TESTING

The TPO uprate power ascension is based on the guidelines from TLTR Section L.2. Required pre-operational tests will be performed.

In preparation for operation at TPO uprated conditions, routine measurements of reactor and system pressures, flows, and select major rotating equipment vibration are taken near 95% and 100% of CLTP, and at 100% of TPO RTP. The measurements will be taken along the same rod pattern line used for the increase to TPO RTP. Core power from the APRMs is re-scaled to the TPO RTP before exceeding the CLTP and any necessary adjustments will be made to the APRM alarm and trip settings.

The turbine pressure controller setpoint will be readjusted at ≤ 95% of CLTP and held constant. The setpoint is reduced so the reactor dome pressure is the same at TPO RTP as for the CLTP. Adjustment of the pressure setpoint before taking the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor and the turbine control valves.

Demonstration of acceptable fuel thermal margin will be performed prior to and during power ascension to the TPO RTP at each steady-state heat balance point defined above. Fuel thermal margin will be projected to the TPO RTP point after the measurements taken at 95% and 100% of CLTP to show the estimated margin. The thermal margin will be confirmed by the measurements taken at full TPO RTP conditions. The demonstration of core and fuel conditions will be performed with the methods currently used at the plant.

Performance of the pressure and FW/level control systems will be recorded at each steady-state point defined above. The checks will utilize the methods and criteria described in the original startup testing of these systems to demonstrate acceptable operational capability. Water level changes of ±3 inches and pressure setpoint step changes of 3 psi will be used. If necessary, adjustments will be made to the controllers and actuator elements.

Steam separator and dryer performance will be evaluated as part of the TPO startup testing by measuring the MSL moisture content. The evaluation will be conducted at TPO RTP at the core flow and radial power distribution conditions achieved. Testing during the current operating cycle will establish the moisture carry-over fraction at 100% CLTP for the various core flow and power distribution cases tested. Following the TPO startup testing, testing will continue during the next operating cycle (Cycle 14) to further evaluate MSL moisture content at other core flow and power distribution conditions attained.

The increase in power for the TPO uprate (1.5%) is sufficiently small that large transient tests are not necessary. High power testing performed during initial startup demonstrated the adequacy of the safety and protection systems for such large transients. Operational occurrences have shown the unit response is clearly bounded by the safety analyses for these events.

[Redacted]

10.5 Operator Training And Human Factors

No additional training (apart from normal training for plant changes) is required to operate the plant in the TPO uprate condition. For TPO uprate conditions, operator response to transient, accident and special events are not affected. Operator actions for maintaining safe shutdown, core cooling, containment cooling, etc., do not change for the TPO uprate. Minor changes to the power/flow map, flow-referenced setpoint, and the like, will be communicated through normal operator training. Simulator changes and validation for the TPO uprate will be performed in accordance with PNPS procedures.

10.6 Plant Life

Two degradation mechanisms may be influenced by the TPO uprate: (1) Irradiation Assisted Stress Corrosion Cracking (IASCC) and (2) FAC. The increase in irradiation of the core internal components influences IASCC. The increase in irradiation of the core internal components influences IASCC. The increase in steam and FW flow rate influence FAC. However, the sensitivity to a 1.5% change is small and various programs are currently implemented to monitor the aging of plant components, including Equipment Qualification, FAC, and Inservice Inspection. Equipment qualification is addressed in Section 10.3, and FAC is addressed in Section 3.5. These programs address the degradation mechanisms and do not change for the TPO uprate. The core internals see a slight increase in fluence, but the inspection strategy used at PNPS based on the BWRVIP is sufficient to address the increase. The Maintenance Rule also provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

The longevity of most equipment is not affected by the TPO uprate because there is no significant change in the operating conditions. No additional maintenance, inspection, testing, or surveillance procedures are required.

10.7 NRC AND INDUSTRY COMMUNICATIONS

NRC and industry communications are discussed in the TLTR, Section B.4. Per the TLTR, a plant-specific review of NRC and industry communications is not needed for a TPO uprate.

10.8 EMERGENCY OPERATING PROCEDURES

The Emergency Operating Procedure (EOP) action thresholds are plant unique and will be addressed using standard procedure updating processes. It is expected that a TPO uprate of 1.5% will have a small effect on the operator action thresholds and to the EOPs in general.

11.0 REFERENCES

1 GE Nuclear Energy, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," (TLTR), Licensing Topical Report NEDC-32938P, Class III (Proprietary), July 2000.

2 GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1), Licensing Topical Reports NEDC-32424P-A, Class III (Proprietary), February 1999; and NEDO-32424, Class I (Non-proprietary), April 1995.

3 GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2), Licensing Topical Reports NEDC-32523P-A, Class III (Proprietary), February 2000; NEDC-32523P-A, Supplement 1 Volume I, February 1999; and Supplement 1 Volume II, April 1999.

4 NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002.

5 Topical Report CENPD-397-P, Rev 1, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," January 2000.

6 NRC SER, Safety Evaluation by the Office of Nuclear Reactor Regulation Subject: Acceptance For Referencing of CENPD-397-P, Revision –01-P, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," (TAC No. MA6452), March 20, 2000.

7 GE Nuclear Energy, "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P, Class III (Proprietary), Revision 1, July 2001.

8 GE Nuclear Energy, "Final Response to Request for Additional Information – GE Nuclear Energy Licensing Topical Report NEDC-32938P RAI#'s 18 & 19," MFN 01-053, dated October 5, 2001.