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Enclosure 6 to L-MT-08-036

ALION-CAL-MNGP-4370-03, "MNGP EPU - TSC Internal Dose," Rev 0

The calculation has been superseded.
See L-MT-10-072 for the revised results.

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The calculation has been superseded.
See L-MT-10-072 for the revised results.

ALION
SCIENCE AND TECHNOLOGY

DESIGN CALCULATION AND ANALYSIS COVER PAGE

Calculation No: ALION-CAL-MNGP-4370-03	Revision: 0	Page 1 of 20

Calculation Title: MNGP EPU – TSC Internal Dose

Project No: 264-4370

Project Name: Monticello Nuclear Generating Plant Extended Power Uprate (EPU) Project

Client: Nuclear Management Company

Document Purpose/Summary: The purpose of this calculation is to determine the internal cloud dose contribution to the technical support center (TSC) operators at the Monticello Nuclear Generating Plant (MNGP) site following a design basis loss of cooling accident (LOCA). The analysis is performed based on plant operation at 102% of the Extended Power Uprate (EPU) power level of 2004 MWth using the Alternative Source Term (AST) in accordance with the guidance provided by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.183 (July 2000) and as allowed by 10 CFR 50.67. All calculated doses are shown to be below the regulatory limits for each specified locations.

All software used in the preparation of this calculation meets QAP 3.5, *Use of Computer Software and Error Reporting* requirements.

Preparer Signature: Not Required Date:

DESIGN VERIFICATION METHOD	QA APPLICABILITY LEVEL
☒ Design Review	☒ Nuclear Safety Related
☐ Alternative Calculation	☐ Quality Significant
☐ Qualification Testing	☐ Nuclear Non-Safety Related

Professional Engineer Approval (if required) Not Required (Signature) Date

	Printed/Typed Name	Signature	Date
Prepared By:	R. E. Anderson	RE Anderson	1/07/08
Reviewed By:	D. M. Hindera	J M. Cajigas for DMH	01/07/08
Approved By:	J. M. Cajigas	J M. Cajigas	01/07/08

Form 3.4.1
Revision 3
Effective Date: 2/28/07

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[redacted]

Table 1-1 Computer Codes Used For EPU

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Nominal Reactor Heat Balance	ISCOR	09	(3)	NEDE-24011P Rev. 0 SER
Reactor Core and Fuel Performance	TGBLA PANACEA ISCOR	06 11 09	Y(2) Y(2) (3)	NEDE-30130-P-A NEDE-30130-P-A NEDE-24011P Rev. 0 SER
Thermal Hydraulic Stability	ODYSY TRACG OPRM	05 04 01	Y N(16) Y	NEDC-32992P-A, Class III, July 2001 NEDO-32465-A, Class I, August 1996
RPV Fluence	TGBLA DORTG	06 01	Y(2) Y	See notes 13 and 14
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	09 07 02	(3) (4) (15)	NEDE-24011P Rev. 0 SER NEDE-20566-P-A NEDE-32176P, Rev. 2, Dec. 1999 NEDC-32177P, Rev. 2, Jan 2000 NRC TAC No M90270, Sep 1994
Transient Analysis	PANACEA ISCOR ODYN SAFER	11 09 10 04	Y (3) Y (6)	NEDE-30130-P-A (5) NEDE-24011P Rev. 0 SER NEDO-24154-A NEDC-32424P-A, NEDC-32523P-A, (9), (10) (11)
Anticipated Transient Without Scram	ODYN STEMP PANACEA	10 04 11	Y (7) Y	NEDE-24154P-A Supp. 1, Vol. 4 NEDE-30130-P-A
Containment System Response	SHEX M3CPT LAMB	06 05 08	Y Y (4)	(8) NEDO-10320, Apr. 1971 NEDE-20566-P-A September 1986
Appendix R Fire Protection	GESTR SAFER SHEX	08 04 06	(6) (6) Y	NEDE-23785-1-PA, Rev. 1 (9) (10) (11) (8)
Reactor Recirculation System	BILBO	04V	NA	(1) NEDE-23504, February 1977
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03A	Y Y Y (3) Y	NEDO-20566A NEDE-23785-1-PA, Rev. 1 (6) (9) (10) (11) NEDE-24011P Rev. 0 SER NEDC-32084P (12)
Station Blackout (SBO)	SHEX	06	Y	(8)
Fission Product Inventory	ORIGEN	2.1	N	Isotope Generation and Depletion Code
Plant Life	CHECWORKS™	2.1	N	Industry Standard

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Add "and 7.2a"

Task	Computer Code*	Version or Revision	NRC Approved	Comments
High Energy Line Break (HELB) Subcompartment Evaluation	GOTHIC	7.1	N	

Add new line:
Task: Containment Accident Pressure and Multiple Spurious Operations analyses
Code: GOTHIC
Version: 7.2b
NRC Approved: N

* The application of these codes to the EPU analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the EPU programs.

(1) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GEH for "Level-2" application and is part of GEH's standard design process. Also, the application of this code has been used in previous power uprate submittals.

(2) Letter, S.A. Richards (USNRC) to G. A. Watford (GEH), "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II – Implementing Improved GE Steady State Methods (TAC No. MA6481)," November 10, 1999.

(3) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GEH) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and LOCA applications is consistent with the approved models and methods.

(4) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566P-A and NEDO-20566A), but no approving SER exists for the use of LAMB in the evaluation of reactor internal pressure differences or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.

(5) The physics code PANACEA provides inputs to the transient code ODYN. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of TGBLA Version 06 and PANACEA Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GEH) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady State Methods," (TAC NO. MA6481), November 10, 1999.

(6) The ECCS-LOCA codes are not explicitly approved for Transient or Appendix R usage. The staff concluded that SAFER is qualified as a code for best estimate modeling of loss-of-coolant accidents and loss of inventory events via the approval letter and evaluation for NEDE-23785P, Revision 1, Volume II. (Letter, C.O. Thomas (See NRC) to J.F. Quirk (GEH), "Review of NEDE-23785-1 (P), "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volumes I and II," August 29, 1983). In addition, the use of SAFER in the analysis of long term Loss-of-Feedwater (LOFW) events is specified in the

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approved LTRs for power uprate: "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 and "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, February 2000. The Appendix R events are similar to the loss of FW and small break LOCA events.

(7) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(8) The NRC approved the application of the methodology in the SHEX code to containment response applications in the CLTR, (Reference 1, Section 4.1). The NRC approval of SHEX for containment analysis applications at Monticello is described in USNRC, Issuance of Amendment responding to Monticello Nuclear Generating Plant, License Amendment Request dated June 2, 2004, Revised Analysis of Long-Term Containment Response and Overpressure Required for Adequate NPSH for Low Pressure ECCS Pumps, (TAC No. MB7185), Amendment 139 to DPR-22.

(9) Letter, J.F. Klapproth (GEH) 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.

(10) Letter, S.A. Richards (NRC) to J.F. Klapproth, "General Electric Nuclear Energy (GENE) Topical Reports GENE (NEDC)-32950P and GENE (NEDC)-32084P Acceptability Review," May 24, 2000.

(11) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(12) The NRC approved the TASC-03A code by letter from S. A. Richards, NRC, to J. F. Klapproth, GE Nuclear Energy, Subject: "Review of NEDC-32084P, TASC-03A, A Computer Code for Transient Analysis of a Single Fuel Channel," TAC NO. MB0564, March 13, 2002. The acceptance version has not yet been published.

(13) CCC-543, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," Radiation Shielding Information Center (RSIC), January 1994.

(14) Letter, H. N. Berkow (USNRC) to G. B. Stramback (GEH), "Final Safety Evaluation Regarding Removal of Methodology Limitations for NEDC-32983P-A, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (TAC No. MC3788)," November 17, 2005.

(15) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(16) TRACG02 has been approved in NEDO-32465-A by the US NRC for the stability DIVOM analysis. The CLTP stability analysis is based on TRACG04, which has been shown to provide essentially the same or more conservative results in DIVOM applications as the previous version, TRACG02.

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[redacted]

the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release. As to each group of criteria, there is a statement of NSP's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a list of references to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-34, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-40 and draft GDC-42. There is no draft GDC directly associated with current GDC-34.

The TBS is described in Monticello USAR Section 11.4, "Main Turbine Bypass System." Section 4.0 in the NRC's Safety Evaluation Report for License Amendment 102, September 16, 1998 summarizes the review and acceptance methodology and conclusions regarding the integrity of the MSIV leakage collection path at Monticello.

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The TBS is included in the discussion of the license renewal evaluation for the Main Steam System. That discussion can be found in NUREG-1865, Section 2.3.4.4. Management of aging effects on the Main Steam System is documented in NUREG-1865, Section 3.4.2.3.4.

Technical Evaluation

The Turbine Steam Bypass System provides a means of accommodating excess steam generated during normal plant maneuvers and transients. The turbine bypass valves are currently rated for a total steam flow capacity of not less than 13.3% of the current rated reactor steam flow, or ~0.97 Mlb/hr. Each of two bypass valves is designed to pass a steam flow of ~485,000 lbm/hr and this does not change at EPU RTP. At EPU conditions, rated reactor steam flow is 8.34 Mlb/hr, resulting in a bypass capacity of ~~~11.6 %~~ of EPU rated steam flow.

The Turbine Steam Bypass System is a normal operating system that is used to bypass excessive steam flow. The absolute flow capacity (mass flow rate) of the bypass system is unchanged. The bypass flow capacity is included in some AOO evaluations (Section 2.8.5).

11.5%

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SRXB RAI No. 2.8.3-11

Provide the following information relevant to ATWS-stability: (1) turbine bypass capacity; (2) percent of feedwater (FW) flow that is driven by electric or steam turbine pumps; (3) location of the extraction steam that feeds the feedwater heaters, (4) location of the extraction steam that feeds the FW steam- driven pumps (if any); (5) FW sparger elevation with respect to top of active fuel; (6) location of the SLC injection point in the vessel.

NSPM Response

1) The absolute bypass flow remains at 967,440 lbm/hr as stated in PUSAR Section 2.4.1.2. The bypass capacity is ~~11.6%~~ 11.5% of EPU rated reactor steam flow of 8,335,000 lbm/hr.

2) 100% of Feedwater flow is driven by electric motor driven pumps. MNGP does not have steam turbine pumps.

3) The extraction points for feedwater heaters are shown below.

FW Heater	Extraction Point
11A/B	12th stage- LP turbine
12A/B	10th stage- LP turbine
13A/B	8th stage- LP turbine
14A/B	7th stage- LP Turbine
15A/B	6th stage-HP Turbine exhaust

4) Not Applicable, See response to 2) above.

5) The center line of the FW sparger is at vessel elevation 466 inches (instrument level – 11.5") and the top of active fuel is at 351.5 inches (instrument level – 126 inches) for a difference of 114.5 inches. Instrument 0 corresponds to vessel elevation 477.5". See attached drawing.

6) Standby Liquid Control injects through the core plate DP nozzle which is 6 ft-3 inches above the inside bottom of the vessel. See attached Figure 9 from the MNGP Operations Manual.

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- Incore Guide Tube
- Control Rod Guide Tube
- Fuel Assembly, Top Guide, and Core Plate
- Guide Rods
- Shroud Head Bolts
- RPV Top Head Spare Instrument Nozzle
- RPV Top Head Vent Nozzle
- RPV Head Spray ~~Pipe and Head Spray~~ Nozzle
- Core Spray Piping (internal to RPV)

The results of the vibration evaluation show that continuous operation at a reactor power of 2004 MWt and 105% of rated core flow does not result in any detrimental effects on the safety-related reactor internal components. See Table 2.2-3.

In order to apply the vibration criteria, a structural dynamic analysis was performed to relate peak stresses to measured strains or displacements at sensor locations. Finite element models for each component were developed to calculate the natural frequencies and mode shapes for these components. The locations and magnitudes of the peak stress intensity, including the effects of stress concentration factors, are identified. The acceptance criteria for each mode are then determined by calculating the modal strains and displacements at sensor locations by normalizing the peak stress intensity to 10,000 psi. The percent criteria for each significant natural mode are then determined by obtaining a ratio of the measured modal response to the calculated acceptance criteria for that mode. The percent contributions of the various modes are absolute summed for conservatism. The component is deemed acceptable if the modal sum is less than 100%. A value of less than 100% confirms that the maximum vibration stress is less than 10,000 psi and therefore no fatigue usage is accumulated by the component due to FIV.

During EPU, the components in the upper zone (plenum) of the reactor, such as the moisture separators and dryer, are mostly affected by the increased steam flow. Components near the shroud head, such as Shroud Head Bolts and Guide Rods, are mainly affected by the increase of feedwater flow. Components in the core region and components such as the Core Spray line are primarily affected by the core flow. Components in the annulus region such as the jet pump are primarily affected by the recirculation pump drive flow and core flow. Maximum licensed core flow at Monticello remains unchanged as a result of EPU. Hence there is no change in the vibrations of core flow dependent components from OLTP. Maximum recirculation drive flow increases negligibly (approximately 1.7%) due to increases in core differential pressure.

The steam dryer and steam separators are non safety-related components. Failure of a dryer component does not represent a safety concern, but can result in a large economic effect.

A proprietary evaluation has been performed to characterize dryer stress at EPU conditions considering dynamic loading conditions. This evaluation is provided as enclosures 11 (proprietary) and 12 (non-proprietary). It concludes that the Monticello steam dryer is structurally adequate for operation at EPU conditions.

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Analysis	Natural Frequency (Hz)	Parameter	Units	EPU/OLTP Change	1.02×EPU Value	Evaluation
RPV Top Head Vent Nozzle	188	Steam Velocity	ft/sec	Insignificant Increase	Acceptable	No FIV concern - The natural frequency is 188 Hz and steam velocity is essentially zero near nozzle.
RPV Head Spray ~~Pipe and Head Spray~~ Nozzle	88	Steam Velocity	ft/sec	Insignificant Increase	Acceptable	No FIV concern – The natural frequency is 88 Hz and steam velocity is essentially zero near nozzle.
Core Spray Piping	19	Vortex Shedding Frequency	Hz	No Change	Acceptable	No FIV concern.

SRXB RAI No. 2.8.3-3

Provide a short description of how the Stability Mitigation Actions (e.g. immediate water level reduction and early boron injection) are implemented at MNGP. Does operation at CPPU conditions require modification of any operator instructions?

NSPM Response

The stability mitigation actions are implemented in accordance with the Emergency Operating Procedures. These include C.5-1100, "RPV Control" and C.5-2007, "Failure to Scram." A copy of C.5-2007 follows this RAI response.

A short description of the key EOP actions follows. It is assumed that the basic condition is a core instability event with power oscillations that is not accompanied by complicating factors such as a commensurate loss of RPV inventory or RPV water level indication or a condition that causes a large drywell pressure increase.

Level Control

- Reduce RPV water level to -33" (Prompt level reduction, ADS will be inhibited)
- If necessary to reduce power further, allow RPV water level to decrease to x, where x ≥-126"(Top of Active Fuel) and x < -33"

Power Control (These actions are done concurrently with Level Control.)

- Run back recirculation flow and then trip the recirculation pumps
- Actuate ATWS
- If peak to peak power oscillations are above the large oscillation threshold of 25%, start Standby Liquid Control (manual boron injection)

(performed using EOP support procedure, C.5-3101, "Alternate Rod Insertion")

Pressure Control (These actions are done concurrently with Level Control)

- Open MSIVs and stabilize reactor pressure via the turbine bypass valves
- Use the SRVs or other pressure control methods if necessary

Assuming the level decrease and boron injection are successful in reducing power to below 3% or terminating the torus heatup (SRVs closed and drywell pressure below the scram setpoint), the following actions would occur.

- Maintain RPV water level between a control band of x and -149 inches (Minimum Steam Cooling RPV Water Level) using only preferred ATWS injection systems
- Inject the Hot Shutdown Boron Weight

EMCB RAI No. 27

In Table 2.2-9 of the PUSAR, some of the locations are shown with "--". Please explain what is meant by this designation.

NSPM RESPONSE

The '--' designation is used in the Location column under the EPU portion of the Table and means not applicable for Items 8 and 16. For Items 6a, 6b, 7, 12 14, and 15, the '--' designation means the specified locations of the stresses for the components in question are not known.

EMCB RAI No. 28

In Section 2.2.2 of the PUSAR, it is stated that, "The effects of FIV induced stresses at EPU conditions on safety-related thermowells in the MS and FW system and the sample probe in the FW system were evaluated" and indicates that they remain acceptable under EPU conditions (see page 2-28 of the PUSAR). However, Enclosure 8, page 5 of 9 states, "Replace or remove the thermowells in main steam piping to insure appropriate margin for flow induced vibration." Provide a quantitative summary of the evaluation that supports the acceptability of the thermowells and sample probes in the MS, FW and related piping systems. Identify nonconforming component(s) and provide description of their modification(s).

NSPM RESPONSE

A quantitative summary of the evaluation results are provided in the table below:

Component	CLTP			EPU		
	Zero-to-peak stress (psi)	fs/fn	Reduced Velocity	Zero-to-peak stress (psi)	fs/fn	Reduced Velocity
FW Thermowells	2683	0.57	1.30	4536	0.67	1.53
FW Sample Probes	1627	0.31	0.70	2332	0.36	0.82
MS Thermowells	1308	0.70	2.32	2809	0.82	2.73

The stress results were compared to the 13,600 psi endurance limit for all the materials of the probes and thermowells. At EPU conditions, all of the stress values are below this endurance limit and thus the thermowells and sample probes are structurally adequate. However, it is desired to reduce the ratio of the vortex shedding frequency to the natural frequency of the MS thermowells (TE 2-127A & B) to the CLTP value to minimize the potential of the system jumping into resonance. Reducing the length of the thermowells

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by 10% will accomplish this goal. Currently two options are being evaluated; either replace the thermowells with shorter ones or remove them altogether. Final resolution of this issue is now scheduled for the 2014 refuel outage.

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EMCB RAI No. 29

Page 2-59 of the PUSAR states that:

"The temperatures, accident radiation level, and the normal radiation level increase due to EPU. These effects are not considered to have an adverse effect on the functional capability of nonmetallic components in the mechanical equipment both inside and outside containment."

Please provide a justification that the radiation due to the EPU is not higher than the radiation damage threshold of the non-metallic parts of the resilient seated check valves, hydraulic snubbers and flex joint bellows affected by the EPU.

NSPM RESPONSE

The predicted dose increase due to EPU operation was determined for all plant general areas. The prediction is that the dose will increase slightly. MNGP will perform plant radiation surveys during power ascension testing and at EPU (power operation and post shutdown) to confirm predicted radiation dose rates.

MNGP has active and formal programs in place to properly manage the slight increase in radiation expected for EPU. The subject components are procured and designed for the applicable service environments in accordance the requirements of the Quality Assurance Program. This program includes requirements to assure that plant equipment is suitable for the intended service, and is of acceptable quality consistent with their effect on safety.

The MNGP Check Valve Program closely monitors valve reliability. The program monitors check valve maintenance history and check valve failures. The check valves with non-metallic seals are included in the program. Valves with non-metallic seats receive regular maintenance including inspection and bench testing. The valves are functionally evaluated during maintenance and replaced if necessary. The O-rings and seals are typically replaced regardless of condition. This program has provided reliable check valve performance to date at Monticello, and the slight increase in radiation due to EPU is not expected to have an adverse effect on continued reliability.

Like the check valve program, the MNGP Snubber Program closely monitors snubber reliability. The program monitors maintenance history and snubber failures. The in-

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to locations in the Monticello USAR where there is subject matter relating to the intent of that particular criteria.

While Monticello is not generally licensed to the current GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can usually be made. For the current GDC listed in the Regulatory Evaluation above, with the exception of current GDC-44, the Monticello comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Monticello USAR Appendix E: draft GDC-4, draft GDC-40 and draft GDC-42. There is no draft GDC directly associated with current GDC-44.

The condensate and feedwater system is described in Monticello USAR Section 11.8, "Condensate and Reactor Feedwater Systems." The condensate demineralizer system is described in Monticello USAR Section 11.7, "Condensate Demineralizer System."

In addition to the evaluations described in the Monticello USAR, Monticello's systems and components were evaluated for License Renewal. Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Monticello Nuclear Generating Plant License Renewal Safety Evaluation Report (SER), NUREG-1865, dated October 2006 (Reference 5). The license renewal evaluation associated with the condensate and feedwater system is documented in NUREG-1865, Section 2.3.4.2. The management of the effects of aging on the condensate and feedwater system is documented in NUREG-1865, Section 3.4.2.3.2.

Technical Evaluation

The FW and Condensate systems provide the source of makeup water to the reactor to support normal plant operation. The increase in power level increases the feedwater requirements of the reactor.

The FW and Condensate systems are designed to provide a reliable supply of FW at the temperature, pressure, quality, and flow rate as required by the reactor. The FW and Condensate systems do not perform a system level safety-related function; however, their performance has an effect on plant availability and capability to operate at the EPU conditions.

For EPU, the FW and Condensate systems will meet their performance requirements with modifications to the following non-safety related equipment for increased capacity:

- FW pumps and motors
- Condensate pumps and motors
- Moisture separator drain tank discharge piping (improve sub-cooling to reduce two phase flow)

For life cycle management (i.e., existing equipment will operate within their ratings at the EPU conditions), modifications are anticipated for the following non-safety related equipment:

- FW heaters 13A, 13B, 14A, 14B, 15A, and 15B
- Drain coolers ~~11 and~~ 12

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ENCLOSURE 8

Table 8-2

EPU Phase I Modifications ~~Planned for 2009 (primarily RFO24)~~	
Modification	**Description**
1AR Transformer Replacement	Replace the existing 1AR transformer due to aging concerns. This is a life cycle management (LCM) modification and is not related to EPU
Power Range Neutron Monitoring System (PRNM) (Included for completeness only. NRC approval has been requested under a separate, prior submittal.)	Replace the existing GE analog system with a GE digital system. This is an LCM modification that includes appropriate design considerations to allow implementation of EPU.
GEZIP	Replace the existing zinc injection (GEZIP) skid with a new passive injection skid. This is an LCM modification.
Piping Requalification	Revise documentation to incorporate revised pressure and temperature ratings for specific piping systems affected by EPU. Modify supports as required by the analyses. Main Steam Relief Valve (MSRV) actuator upgrades due to obsolescence issues. Reweld the Main Steam lead drain pipe connection to the Navy nipple.
HP Turbine Replacement	Replace the existing High Pressure (HP) turbine steam path with a new rotor and diaphragms to accommodate increased steam flow under EPU conditions.
LP Turbine Modifications	Replacement of several diaphragm sets and one set of buckets in each low pressure (LP) turbine to accommodate increased steam flow under EPU conditions. Replacement of selected casing bolts. Evaluate and replace the extraction steam expansion joints.
Isophase Bus Cooling	Replace the existing isophase cooling skid with a new one sized for increased EPU heat loads. Add a new redundant isophase cooling skid to increase reliability.

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ENCLOSURE 8

EPU Phase I Modifications ~~Planned for 2009 (primarily RFO24)~~	
Modification	**Description**
Electronic Pressure Regulator	Replace or respan the electronic pressure regulator to accommodate the increased main steam line pressure drop.
Cross Around Relief Valve (CARV) Replacement	Replace the existing CARVs and associated discharge piping to provide increased relieving capacity under EPU conditions.
Torus Attached Piping	Revise the documentation to incorporate new analyses for EPU conditions. Modify some existing supports to maintain stress limits under EPU conditions.
Main Steam Flow Transmitters	Respan or replace the main steam flow transmitters to accommodate increased flows under EPU conditions.
Feedwater Flow Transmitters and Pressure Control Instrumentation	Respan or replace the FW flow transmitters to maintain functionality with increased flow under EPU conditions. Respan or replace the existing Feedwater pressure control instrumentation to maintain functionality with increased flows and pressure drops under EPU conditions.
Feedwater Regulating Valves	Adjust the stroke of the feedwater regulating valves to provide additional range for operation at increased flow during Phase I of EPU.
Reactor Feedwater Pump Motor	Rerate the reactor feedwater pump motor to allow operation at increased flow during Phase I of EPU.
Feedwater Heater Drain and Dump Valve Replacement.	Replace the drain and dump valves on the feedwater heaters and drain coolers due to obsolescence issues. This is an LCM modification that will consider EPU conditions to enhance margin.
Inboard Main Steam Isolation Valve (MSIV) Solenoid Valve Replacement	Replace the solenoid valves on the inboard MSIVs to increase the margin between maximum containment pressure and minimum nitrogen supply pressure.
~~11 &~~ 12 Drain Cooler and Feedwater Heater Rerate	Rerate the ~~11 and~~ 12 drain coolers and feedwater heaters for EPU conditions.

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ENCLOSURE 8

Table 8-3

EPU Phase II Modifications ~~Planned for 2011 (primarily RFO25)~~	
Modification	**Description**
Reactor Feed Pump Replacement	Replace the existing reactor feedwater pumps with new pumps sized for EPU conditions. Replace the existing 4KV motors with new 13.8KV motors sized for EPU conditions. Upgrade the minimum flow piping and valves as necessary for the new pumps.
~~Reactor Feed Pump Discharge Check Valve Replacement~~	~~Replace the existing reactor feed pump discharge check valves due to obsolescence issues and to maintain flow margin under EPU conditions. This is an LCM modification.~~
Feedwater Regulating Valve Replacement	Replace the existing feedwater regulating valves with new ones sized for operation under EPU conditions.
Condensate Pump Upgrades	Replace the existing condensate pump internals with new assemblies sized for increased EPU flow rates. Replace the existing 4KV motors with a new 13.8KV motors sized for EPU operating conditions. Upgrade the minimum flow piping and valves as necessary for the new pumps. Raise hotwell water level to mitigate potential flashing and vortexing issues.
Condensate Demineralizer Replacement	Replace the existing condensate demineralizer vessels with new vessels to accommodate increased flow under EPU conditions. Replace the existing control panel with a new digital control panel.
Condensate Flow Transmitters	Respan or replace the condensate flow transmitters to accommodate increased flows under EPU conditions.

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ENCLOSURE 8

EPU Phase II Modifications ~~Planned for 2011 (primarily RFO25)~~	
Modification	**Description**
New 13.8KV Bus Installation	This modification is an LCM modification to increase margin in the on site distribution system. Install a new 13.8 KV bus to supply the new FW and Condensate pump motors and new 13.8 KV reactor recirculation motor-generator (M-G) set motors or adjustable speed drives pending evaluation. Replace the existing 1R and 2R transformers with new transformers, to feed two 4.8KV and two 13.8KV busses. Install new 13.8/480V transformer to feed 131 and 141 motor control centers.
Replace the Recirculation M-G Set Motors	Replace the existing recirculation M-G set motors with new 13.8KV motors or adjustable speed drives pending the results of an evaluation. This is an LCM upgrade.
Generator Rewind	Rewind the existing main generator stator and rotor to provide increased capacity required for EPU operation.
~~Generator Hydrogen Coolers~~	~~Replace the generator hydrogen coolers to provide additional capacity for EPU operation.~~
Main Exciter Replacement	Replace the existing main generator exciter with a new one to maintain operating margin under EPU conditions. This is an LCM modification.
Feedwater Heater Replacement	Replace the existing 13, 14 and 15 feedwater heaters with new ones sized for EPU conditions.
Drain Cooler Replacement/Reanalysis	Replace, re-analyze or modify the existing ~~11 and~~ 12 drain coolers to maintain margin under EPU conditions.
Moisture Separator Drain Tank Cooling	Provide condensate injection into the moisture separator drain tanks discharge piping to increase sub-cooling under EPU conditions. This will stabilize flow and eliminate control issues for the drain valves.

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SRXB RAI No. 2.8.3-4

What is the current status of LTSS Option III implementation? When will it be armed in the plant?

NSPM Response

~~The OPRM-based Option III LTS equipment was installed in the plant as part of the PRNMS modification at MNGP. Both OPRM trip outputs will be disabled during the OPRM monitoring and evaluation period. The period extends from the startup following PRNM system installation to 90 days of steady-state operation after reaching full power. The monitoring period is described in Section 5.1.2 of Enclosure 1 of the MNGP PRNM licensing amendment request dated February 6, 2008 and in NSPM letter dated November 6, 2008. It is currently scheduled to be armed on August 31, 2009.~~

The OPRM arming (not bypassed) requirements are included in Section 3.3.1.1 of the MNGP Technical Specifications. For additional information, please see the SER for PRNMS dated January 30, 2009 (TAC No. MD8064).

The OPRM-based Option III LTSS equipment was installed in the plant as part of the PRNMS modification at MNGP. This equipment was turned over to plant operations in September 2009. The monitoring and evaluation period is complete.

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Enclosure 2

<u>2) CPTB – COMPONENT PERFORMANCE AND TESTING BRANCH</u>

<u>NRC Question:</u>

1. In response to Generic Letter (GL) 89-10,"Safety-Related Motor-Operated Valve Testing and Surveillance," and 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," the licensee should have in place approved programs for design-basis review, testing and surveillance for safety-related MOVs. Provide an evaluation of the EPU impact on these programs.

quantify changes to the functional and design requirements for those MOVs included within the scope of the Monticello GL 89-10/96-05 program. The following changes in system pressures were identified:

<u>NMC Response:</u>

All Extended Power Uprate (EPU) Task Reports for the various systems were reviewed to ~~determine increases in system pressure caused by EPU.~~

- The Reactor Water Cleanup (RWCU) tie to Feedwater results in a 12.1 psig increase. This increase is applied as an increase to the RWCU Isolation valves.
- An increase of slightly less than 5 psig occurs for Residual Heat Removal Service Water (RHRSW) due to EPU. There are no RHRSW valves in the GL 89-10 Motor-Operated Valve (MOV) Program.
- An increase in the Containment Response pressures for both wetwell (+1.5 psig) and drywell (+4.1 psig) peak pressures. This increase is applied to applicable MOV system calculations.

Insert A

All of the functional calculations and each of the valve scenarios have been reviewed for changes due to these increases.

A review of GL 89-10 MOVs for effects due to EPU conditions was performed based on changes to environmental temperatures associated with High Energy Line Break (HELB) and post-LOCA Reactor Building Heatup.

Changes identified from the above review were used as inputs to the Monticello MOV (MMOV) database, which computes required and available thrusts and compares the field test results to compute various operating margins.

MOVs that are powered by DC were evaluated using the Boiling Water Reactor Owner's Group (BWROG) DC MOV Methodology. The results from the DC BWROG methodology were input into the MMOV database and margins were obtained, as before.

Following the review of MOV Functional calculations, the following adjustments were made to line pressure and differential pressure for the affected valves and scenarios:

- MO-1426 and MO-4229, MO-4230 (Reactor Building Closed Cooling Water (RBCCW) Drywell Isolation) isolate against maximum drywell pressure (+4.6 psig from previous analysis). The MMOV database pressure for RBCCW valves was increased by 10 psig for both differential pressure and line pressure.
- MO-1986 and MO-1987 (RHR Suction Isolation) isolate against maximum wetwell pressure (+1.5 psig from previous analysis). The MMOV database

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pressure for these RHR valves was increased by 10 psig for both differential pressure and line pressure.

- MO-2100, MO-2101, and MO-2102 (Reactor core isolation cooling (RCIC) Pump Suction Isolation) open (MO-2100, MO-2101) or isolate (MO-2102) against maximum wetwell pressure (+1.5 psig from previous analysis). The MMOV database pressure for these RCIC valves was increased by 10 psig for both differential pressure and line pressure.
- MO-2096 (RCIC Barometric Condenser Isolation) isolates against maximum wetwell pressure (+1.5 psig from previous analysis). The MMOV database pressure for this RCIC valve was increased by 10 psig for both differential pressure and line pressure.
- MO-2061 and MO-2062, MO-2063 (high pressure coolant injection (HPCI) Suction Isolation) open (MO-2061, MO-2062) or isolate (MO-2063) against maximum wetwell pressure (+1.5 psig from previous analysis). The MMOV database pressure for these HPCI valves was increased by 10 psig for both differential pressure and line pressure.
- MO-2397 and MO-2398 (RWCU Containment Isolation) isolate against maximum drywell pressure (+4.1 psig from previous analysis). The MMOV database pressure for RWCU valves was increased by 10 psig for both differential pressure and line pressure.

MOVs have been evaluated using peak temperature values that in many cases are higher than the equivalent current functional value. The values used are bounding and are used for analytical convenience. As noted above, the values used for pressure changes (both differential pressure and line pressure) are chosen to be significantly above the changes due to EPU.

After making the temperature changes and applying the BWROG DC Motor Performance Methodology (as applicable), a margin report was obtained to determine the margin differences reported for each valve.

The results of the MOV Margin Report were reviewed to determine which valves in the GL 89-10 MOV population had margins which were reduced to, or below zero percent due to changes determined above.

Conclusion

All MOVs with the following exception will perform their safety-related functions under EPU conditions:

> MO-2021 requires field adjustment to the torque switch to reduce the output thrust below the maximum actuator capability at degraded voltage conditions.

This is the valve identified in PUSAR Section 2.2.4, MO-2021 (RHR Containment Spray Isolation).

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Insert A

The following functional / design requirements as a result of the EPU project (including those changes described above) were recalculated for each of the following valves:

- RBCCW System MOV Functional Analysis
- Core Spray MOV Functional Analysis
- RHR System MOV Functional Analysis
- REC MOV Functional Analysis
- MS MOV Functional Analysis
- RWCU MOV Functional Analysis
- HPCI MOV Functional Analysis
- RCIC MOV Functional Analysis

Additionally, a review of the GL 89-10 MOVs for the effects due to EPU conditions was performed based on the changes to the environmental temperatures associated with High Energy Line Break (HELB) and post-LOCA Reactor Building Heatup. These changes were calculated and documented in a plant calculation addressing MOV Environmental Temperatures.

The changes to the above MOV functional analyses and MOV environmental temperatures were used as inputs into the weak link analyses and MOV component calculations. The weak link analyses determined the thrust limited component of the valve to ensure that, under design basis temperature / pressure conditions, damage to the valve can be precluded. This weak link value was used as an input (in combination with the system / pressure / temperature conditions) to perform the MOV component calculations.

The MOV Component calculations were completed to evaluate the performance of those GL 89-10 MOVs under the EPU system conditions. Once these component calculations were completed, the in-plant switch configuration from the latest diagnostic test was compared against the newly established performance requirements to determine a functional margin for all GL 89-10 valves. These margins were reviewed to determine which valves in the GL 89-10 MOV population had margins which were reduced to, or below zero percent, due to changes described above.

<u>Conclusion</u>
All MOVs will perform their safety-related functions under EPU conditions with the following exceptions:

- MO-2009, 12 RHR Torus Cooling Injection Valve
- MO-2014, 11 LPCI Inboard Injection Valve
- MO-2015, 12 LPCI Inboard Injection Valve
- MO-2020, 11 Containment Spray Outboard Valve
- MO-2021, 12 Containment Spray Outboard Valve
- MO-2023, 12 Containment Spray Inboard Valve
- MO-2034, HPCI Steam Line Isolation Inboard Valve
- MO-2035, HPCI Steam Line Isolation Outboard Valve
- MO-2061, HPCI Torus Suction Inboard Isolation Valve
- MO-2062, HPCI Torus Suction Outboard Isolation Valve

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The above valves have been evaluated to ensure that they remain operable until their switches can be adjusted either prior to or during the next refueling outage. All valves have switch adjustments scheduled. The specific requirements for adjustment are contained within each valve's respective performance analysis (i.e. component calculation) performed under EPU conditions.

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Enclosure 2

NRC Question

2. Provide review results of each safety-related systems and safety-related valves (including safety/relief valve setpoints) that are affected by EPU, and maximum changes in flow rate, pressure, and fluid/ambient temperature. The licensee states that a field adjustment to a torque switch setting was identified for one MOV. The licensee should identify this valve and associated system, and provide the evaluation that resulted in the required adjustment.

See response provided in RAI 1 above.

NMC Response:

As clarified by conference call with the staff on May 20, 2008, the following PUSAR sections (Monticello EPU License Amendment Request (LAR) Enclosure 5) contain evaluations for specific systems regarding changes, if any, due to new EPU conditions for flow rate, pressure, and temperature:

- ADS (Automatic Depressurization System) PUSAR, Section 2.8.4.2 notes that:

  NMC has evaluated the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The evaluation indicates that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded.

- RHR (Residual Heat Removal System) PUSAR, Section 2.8.4.4 notes that:

  NMC has evaluated the effects of the proposed EPU on the RHR system. The evaluation indicates that the RHR system will maintain its ability to cool the Reactor Coolant System (RCS) following shutdown and provide decay heat removal.

  In addition, the Low Pressure Coolant Injection (LPCI) mode of RHR is discussed in PUSAR Section 2.8.5.6.2.

- SLCS (Standby Liquid Control System) PUSAR, Section 2.8.4.5 notes that:

  NMC has evaluated the effects of the proposed EPU on the SLCS. The evaluation indicates that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU.

- HPCI (High Pressure Coolant Injection System) PUSAR, Section 2.8.5.6.2 notes that:

  Because the maximum normal operating pressure and the safety relief valve (SRV) setpoints do not change for EPU, the HPCI system performance requirements do not change.

- Core Spray (CS) System, PUSAR, Section 2.8.5.6.2 notes that:

  The slight change in the system operating condition due to EPU for a postulated LOCA does not affect the hardware capabilities of the CS system. The generic core spray distribution assessment provided in General Electric (GE) ELTR2, Section 3.3, continues to be valid for the EPU as described in GE CLTR, Section 4.2.3.

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Enclosure 2

~~As described in PUSAR Section 2.2.4, MO-2021 (RHR Containment Spray Isolation) requires a field adjustment to the torque switch. Under EPU conditions, the ambient accident temperature increases to the point that the actuator available output is decreased slightly (-0.65 percent) below the current torque switch setting. This valve is also identified in response to Question 1 of this enclosure.~~

NRC Question

3. Describe activities and lessons learned programs that are dedicated to the enhancement of MOVs and AOVs performance/design basis review, and testing programs.

NMC Response:

Insert B

MOV Program

~~The purpose of the MMOV Program is to ensure that all safety-related MOVs are selected, set and maintained so that the MOVs will operate under design basis conditions.~~

AOV Program

The purpose of the Monticello Air Operated Valve Program is to ensure that air operated valves determined to be active safety related, safety significant, or critical are selected, set, tested, and maintained so that the Air Operated Valves (AOVs) will operate under normal, abnormal, or emergency operating design basis conditions. Furthermore, the AOV Program will ensure continued AOV reliability for the life of the plant.

Additionally, NMC fleet procedures require operating experience to be evaluated and incorporated for the corresponding programs.

The MOV and AOV programs are governed by site and fleet procedures which address:

- MOV and AOV Program Requirements
- MOV and AOV Program Engineering Standards
- Grouping Of MOVs For Selection Of Test Frequency
- MOV Margin Improvement
- MOV Program Design Methodology
- MOV Margin Analysis and Periodic Verification
- MOV Diagnostic Test Preparations and Evaluation

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Insert B

The purpose of the Monticello Motor-Operated Valve (MMOV) Program is to ensure that all Safety-Related (SR) Motor-Operated Valves (MOV) meet the following:

- The functional and design requirements for an MOV meet the applicable U.S. Nuclear Regulatory Commission (U.S. NRC) regulatory requirements, code requirements, and service conditions
- The valve design features and operating conditions that can affect MOV operation are evaluated
- The required stem thrust to operate the valve is calculated
- The valve assembly's rated and survivable thrusts are calculated
- The motor operator design features that affect MOV operation are evaluated
- The operator output thrust and torque are calculated
- The compatibility of the operator and valve are evaluated
- Appropriate control, protection, and alarm logic and the set points for limit switches are evaluated
- Appropriate baseline and periodic verification testing is performed to ensure that the switch set points are within the design tolerances and that the stem thrust is maintained within the desired limits.

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Technical Evaluation

Containment Isolation

Motor-Operated and Air Operated Valves

For the majority of the GL 89-10 MOVs and safety-related air operated valves (AOVs), the EPU operating conditions are bounded by the existing design inputs used to establish thrust/torque margins for these valves. For the remaining valves, the EPU operating conditions result in small changes to design inputs, such as differential pressure and maximum ambient temperature, depending on the valve-operating scenario. These changes have been identified for each affected valve. A bounding engineering evaluation was performed that enveloped the EPU changes to the design inputs. The evaluations determined that sufficient design margin exists such that all valve actuator capabilities remain within the acceptance criteria for safe operation without causing spurious trips or valve damage. Therefore, the MOVs and AOVs remain capable of performing their design basis functions at EPU conditions without modification. ~~A field adjustment to a torque switch setting was identified for one MOV.~~ Field adjustments are required to switch settings for ten MOVs. This change will be made prior to implementing EPU.

EPU does not introduce any changes to the plant specific analysis or compliance methodology for GL-95-07, Pressure Locking and Thermal Binding of Safety Related Power-Operated Valves. EPU does not result in additional valves that are susceptible to pressure locking and thermal binding, and the existing set of potentially susceptible valves continue to perform their associated safety functions.

Conclusion

NSPM has evaluated the effects of the proposed EPU on safety-related valves. The evaluation addressed the effects of the proposed EPU on its MOV programs related to GL 89-10 and GL 95-07. The evaluation indicates that safety-related valves will continue to meet the requirements of 10 CFR 50.55a(f) and the current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to safety-related valves.

2.2.5 Seismic & Dynamic Qualification of Mechanical & Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section.

The NRC's acceptance criteria are based on (1) GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and

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The steam line HELB events in the Monticello licensing basis were evaluated [[[redacted]]].

2.2.1.2 Liquid Line Breaks

Operation at EPU conditions requires an increase in the MS and FW flow rates, which results in a slight increase in downcomer subcooling. This increase in subcooling may lead to increased break flow rates for liquid line breaks. The mass and energy releases for HELBs in the RWCU, FW, Condensate, CRD, Standby Liquid Control, and Zinc Injection System (GEZIP) systems and instrument and sample lines may be affected by EPU and were re-evaluated at EPU conditions. [[[redacted]]] evaluations of liquid line breaks have been performed at EPU conditions. The results of the Monticello evaluation of liquid line breaks are provided in Table 2.2-1.

RWCU Line Breaks

Monticello has undertaken a complete re-analysis of HELB's in the reactor building updating break and crack analysis originally performed in response to the guidelines of the letter from A. Giambusso, Deputy Director for Reactor Projects, to NSP, Consequences of Piping Failures Outside of the Containment Structure, December 18, 1972 (Giambusso Letter), ISEQ # RAC00159, MO2943-0075, now incorporated as Appendix B to Branch Technical Position SPLB 3-1 in NUREG-0800 Standard Review Plan Section 3.6.1. During the 6.3 percent rerate in 1996, only one new case was reanalyzed at CLTP for the RWCU system – a break in the system suction piping at the outboard isolation valve. For this reason a detailed comparison of CLTP and EPU results for HELBs in the RWCU system is not possible. ~~For the break location that was analyzed during rerate, new mass and energy release calculations considered additional blowdown sources that had not been considered in the previous 1996 analysis. This resulted in an increase in integrated mass release of about 90% and an increase in integrated energy release of 63 percent.~~

The Reactor Building pressure, temperature and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1.

Feedwater System Line Breaks

The mass and energy releases for double-ended breaks and critical cracks in the FW lines were re-analyzed at EPU conditions. At EPU peak break and crack mass releases, flow rates increase ~~by up to 23 percent~~ due to assumed increased pump discharge pressure and changes in fluid enthalpy based on the EPU BOP heat balance. Integrated mass flow does not change significantly due to conservative assumptions that basically drain the hotwells. ~~Based on the small changes in fluid enthalpy, integrated energy release increases for breaks by about four percent.~~ The Reactor Building and turbine building pressure, temperature, flooding, and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1.

Note: No change has been made to blacked out information. Information redacted to preserve integrity of proprietary information.

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Condensate System Line Breaks

The mass and energy releases for double-ended breaks and critical cracks in the high energy portions of condensate lines were re-analyzed at EPU conditions. At EPU peak mass releases flow rates increase ~~by up to 28 percent~~ due to conservative assumptions of increased pump discharge pressure and changes in fluid enthalpy based on the EPU heat balance. Integrated mass flow for breaks does not change significantly due to conservative assumptions that basically drain the hotwells. Based on the small changes in fluid enthalpy, integrated energy release for breaks is essentially unchanged. ~~Based on increased flow rates and energy releases assumed for critical cracks results show increases in integrated mass release of 20 percent and on integrated energy releases up to 4 percent[1].~~ The Turbine Building pressure, temperature and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1.

CRD System Line Breaks

Within the CRD system, only a small portion of the CRD return line to the RWCU return line is high-energy piping. Due to its location break or crack in this 3" piping is indistinguishable from a break or crack in the 3" RWCU piping in this area. A separate Mass and Energy Calculation for the CRD system is not necessary.

Standby Liquid Control System Breaks

Breaks or cracks in SLCS involve an isolated section of piping between two check valves in SLCS. The results of this break do not create harsh environmental conditions for temperature, pressure or liquid flooding level. It is a fixed volume and the current analysis assumed the entire contents of this volume are vented into the room in 1 second. The enthalpy of the release was assumed constant at 536.8 BTU/lbm. This remains bounding with respect to reactor coolant conditions at EPU.

GEZIP (Zinc Injection)

The portions of the GEZIP system with postulated HELBs are small piping (1 ½") located near the Feedwater and Condensate lines in the Turbine Building. Due to the proximity to the postulated Feedwater and Condensate Systems HELBs, any adverse effects from a GEZIP HELB would be enveloped by a postulated HELB on one of the Feedwater or Condensate lines. Because pathways to safe shutdown have been demonstrated for the Feedwater and Condensate postulated HELBs, these same pathways would exist for the much smaller GEZIP HELB. The Feedwater and Condensate system HELBs will remain bounding at EPU conditions.

~~[1] One scenario evaluated had a long isolation time of 8 hours and resulted in significantly larger release of energy over a longer time, however, this case is not limiting due to its small mass flow rate. This scenario was omitted from the evaluation of EPU effects because it is a methods change and not an EPU effect.~~

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[redacted]

with EPRI TR-112657, piping welds identified as Category "A" are considered resistant to IGSCC, and as such are assigned a low failure potential provided no other damage mechanisms are present. Examination criteria for these welds will be in accordance with the RI-ISI process.

2.2.2.1 Pipe Whip and Jet Impingement

Pipe whip and jet impingement loads resulting from high energy pipe breaks are directly proportional to system pressure. Because EPU conditions do not result in an increase in the pressure considered in the high-energy piping evaluations, there is no increased pipe whip or jet impingement loads on HELB targets or pipe whip restraints. ~~Additionally, a review of pipe stress calculations determined that the FW temperature increases associated with EPU conditions will not result in pipe stress levels above the thresholds required for postulating HELBs, except at locations already evaluated for breaks. As a result, EPU conditions do not result in new HELB locations, nor affect existing HELB evaluations of pipe whip restraints and jet targets.~~

Installation of new condensate and feedwater pumps with associated piping modifications will include an evaluation of HELB target impact as part of the planned modification.

Main Steam and Associated Piping System Evaluation

The CLTR, Section 3.5.1, requires a plant specific evaluation for the MS and FW piping because the MS and FW piping and associated branch piping up to the first anchor or support experience an increase in flow, pressure and/or temperature due to EPU, resulting in an increase in stress

The MS piping system and associated branch piping (inside containment) were evaluated for compliance with the ASME Section III, Division I, 1977 Edition with Addenda up to and including Winter 1978 Piping Code stress criteria, including the effects of EPU on piping stresses, piping supports including the associated building structure, piping interfaces with the RPV nozzles, containment penetrations, flanges, and valves.

Because the MS piping pressures and temperatures are not affected by EPU, there is no effect on the analyses for these parameters. Seismic inertia loads, seismic building displacement loads, and SRV discharge loads are not affected by EPU, thus, there is no effect on the analyses for these load cases. The increase in MS flow results in increased forces from the turbine stop valve closure (TSVC) transient. The turbine stop valve closure loads bound the MSIV closure loads because the MSIV closure time is significantly longer than the stop valve closure time (CLTR). Due to the magnitude of the TSVC transient load increase, the transient event was reanalyzed. The main steam piping was then reanalyzed using this revised load definition.

Pipe Stresses

The results of the Main Steam system piping analysis indicate that piping load changes do not result in load limits being exceeded for the MS system and attached branch piping (SRVDL, Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), RPV Vent, and MSIV Drain) or for the RPV nozzles. The design analyses demonstrates that the calculated stresses meet ASME Section III, Division I, 1977 Edition with Addenda up to and including Winter 1978 Piping Code allowable limits to justify operation at EPU conditions except for one small bore

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[redacted]

branch line that did not meet displacement criteria. Additional detailed analysis will be performed to qualify this line or the piping modified prior to operation at EPU conditions. No new postulated break locations were identified.

Pipe Supports

The MS piping (inside containment) was evaluated for the effects of flow increase on the piping snubbers, hangers and struts. A review of the increase in MS flow associated with EPU indicates that piping load changes do not result in any load limit being exceeded. Similarly, the drywell steel was evaluated for changes in loading and determined to be adequate for EPU.

Feedwater and Associated Piping System Evaluation

The RCPB portion of FW was evaluated for changes in temperature, pressure and flow. ~~The FW CLTP analyzed temperatures and pressures envelope the EPU conditions.~~ Therefore FW inside containment and associated attached lines outside containment including HPCI, RCIC and Reactor Water Cleanup are acceptable for EPU conditions. FW between the second and third isolation valves outside containment and associated attached lines, HPCI, RCIC and RWCU are not part of the RCPB boundary; however, they were evaluated with the RCPB because they are addressed in the same analytical model as the RCPB FW piping.

The FW analyzed temperatures and pressures for EPU conditions resulted in a small temperature increase for a portion of the FW piping, pressure was not changed. The design temperature on this portion of the FW piping was increased from 400°F to 410°F to bound the temperature change.

Main Steam Isolation Valves

The inboard MSIVs have been evaluated, as discussed in Section 4.7 of ELTR2, Supplement 1. The evaluation covers both the effects of the changes to the structural capability of the inboard MSIV to meet pressure boundary requirements, and the potential effects of EPU-related changes to the safety functions of the inboard MSIVs. The generic evaluation from ELTR2 is based on (1) a 20% thermal power increase, (2) an increased operating dome pressure to 1095 psia, (3) a reactor temperature increase to 556°F, and (4) steam and FW increases of about 24%. The evaluation from ELTR2 is confirmed applicable to Monticello inboard MSIVs. An increase in flow rate assists inboard MSIV closure, which results in a slightly faster inboard MSIV closure time. The Monticello inboard MSIV has design features that ensure the MSIV closure time is not reduced below the stroke time limit. The closing time of the inboard MSIVs is controlled by the design of the hydraulic control valves and the function of the hydraulic damper or dashpot. Prior to EPU implementation, the hydraulic control valve of the inboard MSIV will be adjusted for the required closing time. The solenoid valves for the inboard MSIVs will be replaced with valves designed to function with a lower differential pressure to atmosphere. The inboard MSIVs are designed for 1250 psig at 575°F. Hence, the pressure integrity of the valves is not affected by operation at EPU conditions. The inboard MSIV is designed to close at normal closing time of 3 to 10 seconds at 200% of original design steam flow. The flow restrictor cross-sectional area controls the maximum steam flowrate (choke flow). The flow restrictors for EPU are not being changed or modified, so the maximum flow rate will not change for EPU. Therefore, the normal closing time of 3 to 10 seconds is not affected for EPU. Therefore, the Monticello EPU is bounded by conclusions of the evaluation in Section 4.7 of ELTR2, and the inboard MSIVs are acceptable for EPU operation.

The outboard MSIVs are a double disc gate valve type with air/spring actuator to close.

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[redacted]

The EPU conditions for the outboard MSIV analysis is 1040 psia dome pressure with steam at saturated conditions. The steam flow associated with this condition is 2.133 Mlbm/hr per outboard MSIV. These conditions provide the most limiting conditions of operation for EPU. The outboard MSIVs are designed for 1250 psig at 575°F. Hence, the pressure integrity of the valves is not affected by operation at EPU conditions.

The outboard MSIVs are designed to close with a maximum differential pressure of 1000 psid. The maximum differential pressure for outboard MSIV closure is an outside steam line break. The differential pressure associated with this break for EPU operation is below 1000 psia and thus valve closure is assured for EPU.

Therefore, the outboard MSIVs are acceptable for EPU operation.

Feedwater Evaluation

The FW system outside containment from the isolation valve to the motor operated valve (MOV) downstream of high pressure heaters was evaluated for compliance with the ANSI-B31.1-1977, including Winter 1978 Addenda Power Piping Code stress criteria, and for the effects of deadweight, pressure, seismic and thermal expansion displacements on the piping snubbers, hangers, and struts. Piping interfaces with penetrations, flanges, equipment nozzles, and valves were also evaluated. ~~The remaining FW piping from the MOVs to the condensate pumps will be modified as a result of the replacement of the feedwater and condensate pumps, and will be qualified for full EPU operation as part of the modification. The current piping and associated components are adequate for operation within the capability of the existing feedwater and condensate pumps.~~

Pipe Stresses

A review of the small increases in pressure, temperature and flow associated with EPU indicates that the EPU temperature, pressure and flow conditions are ~~bounded by the existing analyses.~~ modified under EPU conditions. The ~~original~~ verified design analyses have sufficient design margin between calculated stresses and ANSI-B31.1-1977, including Winter 1978 Addenda Code allowable limits to justify operation at EPU conditions.

Therefore, EPU does not have an adverse effect on the FW piping design. ~~No new postulated pipe break locations were identified.~~

Pipe Supports

The FW system was evaluated for the effects of seismic, deadweight and thermal expansion displacements on the piping snubbers, hangers, and struts. A review of the increases in temperature and FW flow associated with EPU indicates that the EPU conditions ~~are bounded by the existing analyses.~~ were reevaluated and determined to be satisfactory.

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[redacted]

Table 2.2-1 Liquid Line Breaks

	EPU Analysis Results[1] Compared to Current EQ Analysis			
Location	**Mass and Energy Release**	**Room Flooding**	**Room Pressure**	**Room Temperature**
Reactor Building		Increase ≤ 2.15 ft	Increase ≤ 1.25 psi	Increase ≤ 86.5 °F
Turbine Building		Increase ≤ 4.35 ft	Increase ≤ 0.25 psi	Increase ≤ 102.5 °F
RWCU System (affects Reactor Building)	Peak mass flow rate increases.			
FW system (affects Reactor Building and Turbine Building)	Peak mass flow rate increases by up to 23%. Total mass release does not change significantly. Integrated energy release increases by about 4%.			
Condensate system (affects Turbine Building)	Peak mass flow rate increases by up to 28%. Total mass release increases by up to 20%. Integrated energy release increases by about 4%.[2]			

Notes:

1. As discussed in Section 2.2.1, EPU Analysis Results include the composite effects of EPU and changes in methodology.
2. One scenario evaluated had a long isolation time of 8 hours and resulted in significantly larger release of energy over a longer time, however, this case is not limiting due to its small mass flow rate. This scenario was omitted from the evaluation of EPU impacts because it is a methods change and not an EPU effect.

Replace this table with the applicable portions of Table 26-1 - Turbine Building HELB Results and Table 26-2 - Reactor Building HELB Results. These tables are provided in L-MT-12-114, Item 26.

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Table 2.2-2a BOP Piping FW, Extraction Steam, FW Heater Drains & Vents, Condensate and MSR Drains

System	FW[1]	Extraction Steam	Heater Drains & Vents	Condensate[2]	MSR Drain Lines
Maximum pipe stress increase from:					
Temperature expansion	0%	11%	72%	N/A	19%
Pressure	0%	64%	Note 3	N/A	Note 3
Fluid Transients	N/A	N/A	N/A	N/A	N/A
Maximum pipe support loading increase (due to thermal expansion loading):	0%	11%	72%	N/A	19%

Notes:

1. FW section from containment flued head penetration to first isolation valve. The remaining BOP FW will be analyzed and qualified for EPU in conjunction with the FW and Condensate pump modifications prior to operation at EPU conditions.
2. The condensate piping from the FW pumps to the condensate pumps will be analyzed and qualified for EPU in conjunction with the FW and Condensate pump modifications prior to operation at EPU conditions.
3. Calculations for existing pressure stresses were not available. New calculations were prepared to qualify these lines.

Replace this table with the applicable portions of Table 11-2 - Feedwater/Condensate Loading Results and Table 11-3 - Design Pressure and Temperature Comparison of CLTP and EPU. These tables are provided in L-MT-12-114, Item 11.

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This table is replaced with the data provided in L-MT-09-044, Enclosure 1, EMCB RAI No. 13, Table 1 as revised by L-MT-12-114, Item 26.

Table 2.2-2b BOP Piping
CS, RCIC, HPCI and RHR (Outside Containment)

System	CS	RHR Shutdown Cooling	RHR injection	RCIC injection[1]	HPCI injection[1]
Maximum pipe stress increase due to:					
Temperature expansion	8.9%	0%	8.9%	8.9%	8.9%
Pressure	0%	0%	0%	0%	0%
Fluid Transients	N/A	N/A	N/A	N/A	N/A
Maximum pipe support loading increase (due to thermal expansion loading):	8.9%	8.9%	8.9%	8.9%	8.9%

Notes:

1. Portion connecting to Torus or Torus header.

Table 2.2-2c BOP Piping
Main Steam System (Outside Containment)

Maximum pipe stress at EPU due to:	
Temperature expansion	No change
Pressure	No change
Fluid Transients	38% (Note 1)
Maximum pipe support loading:	
EPU (due to thermal expansion loading)	No change
EPU (due to fluid transient loading)	38% (Note 1)

Notes: 1. The fluid transient load and stress increases shown above are based on a 17.5% flow increase. A refined analysis was performed to evaluate the TSVC loads resulting in much lower loads and stresses.

Replace this table with the applicable portions of Table 11-1 - Main Steam Loading Results.

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EMCB RAI No. 3 (a)

PUSAR Section 2.2.1 states that corrective actions are underway to perform HELB analysis upgrades at Monticello due to changes in pipe break methodology.

Explain why corrective actions are in place to upgrade the Monticello pipe break methodology.

NSPM RESPONSE

PUSAR Section 2.2.1 states:

Technical Evaluation
No changes to the implementation of the existing criteria for defining pipe break and crack locations and configurations are being made for EPU . . .

Changes in ~~Methods of~~ Analysis Inputs

The results provided for HELB events affected by EPU, specifically, the liquid line breaks in the Feedwater, Condensate, and RWCU systems show much larger changes than would be expected due to the small changes in pump discharge pressures and small enthalpy changes as a result of EPU. The results are driven by conservative changes in analysis ~~methods~~ inputs resulting from corrective actions underway to perform HELB analysis upgrades at Monticello."

The criteria used to determine high energy lines has not changed with EPU, see RAI 2 above. The changes NSPM referred to are covered in corrective action program action request AR01131913, HELB Program Documentation Deficiencies, which documents a summary of issues being addressed. The most significant changes are related to the assumptions used in determining mass and energy releases from postulated breaks and upgrade of the computer code from GOTHIC version 4.0. The EPU liquid break calculation inputs have been upgraded to consider:

1. Double-ended break flow to include flow from both ends of postulated breaks
2. System depletion to include mass and energy that exists in system piping and pressure vessels
3. A conservative change in assumption for isolation valve stroke time from ASME Section XI Limiting Stroke time to the value listed as the maximum valve operating time in the USAR. If break detection logic exists, valve stroke is initiated when the logic detects the break.
4. A conservative change for flow reduction assumptions with valve closure. CLTP analysis assumed flow was reduced proportional to isolation valve percent closed position. The EPU analysis assumed 100% break flow until isolation valve was fully closed.

A conservative addition of delay time to account for Emergency Diesel Generator starting was assumed. The valve stroke was assumed to begin when the delay time expires.

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5. The liquid mass from fire protection sprinkler systems postulated to actuate from HELB events was included
6. Upgrade computer code from GOTHIC version 4.0 to GOTHIC version 7.1 or later versions

The assumption changes noted above are based on recommendations from site self assessments. These changes will bring the HELB program into closer alignment with industry standards and correct identified deficiencies. The failure to consider fire protection sprinkler system actuation for appropriate HELBs resulted in the issuance of LER 2008-001, Non-Conservative HELB Analysis discovered during EPU, and is documented under AR1125675.

Re-analysis of all HELB breaks and an evaluation of affected EQ components have been completed; formal updating of EQ program documents are the only actions remaining. These actions are being performed coincident with EQ program updates required by EPU.

EMCB RAI No. 3(b)

Verify whether the Monticello pipe break methodology upgrade is based on SRP Section 3.6.2, MEB 3-1 criteria. If not, provide supporting justification.

NSPM RESPONSE

As noted above in the response to Part a) of this question, there is no change to the pipe break methodology at Monticello. The changes involve a re-analysis of breaks using more conservative assumptions of mass and energy release.

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EMCB RAI No. 5

Page 2-23 states that:

> "During the 6.3 percent rerate in 1996, only one new case was reanalyzed at CLTP for the RWCU system - a break in the system suction piping at the outboard isolation valve. For this reason a detailed comparison of CLTP and EPU results for HELBs in the RWCU system is not possible."

The statement that, "For this reason a detailed comparison of CLTP and EPU results for HELBs in the RWCU system is not possible" is not clear.
Please provide clarification.

NSPM RESPONSE

The CLTP analysis of RWCU HELBs evaluated the terminal end break and crack case at the inlet to the RWCU heat exchanger. The evaluation used the mass and energy release rates for a break just outboard of the outboard isolation valve. These were considered the bounding cases and other cases were not run. For EPU, ~~eight~~ multiple HELB locations, covering all possible breaks and cracks, were evaluated.
The response to RAI 3.a above explains changes in assumptions used in evaluation of the EPU HELB cases. As noted on PUSAR page 2-21:

> Because of these changes in methodology, a comparison of the results between EPU and CLTP conditions shows a significantly larger change than would normally be expected based on the small changes in process fluid temperatures and enthalpy resulting from EPU based on previous industry experience. Monticello has chosen not to perform a full re-analysis of these specific liquid line HELBs at CLTP conditions because it was determined that our effort should be focused on completing the corrective actions using bounding conditions. Thus, a detailed breakdown of the magnitude of the change is caused by EPU versus the change resulting from the changes in methods and correction of errors is not provided.

A comparison of the results between EPU and CLTP conditions was not done since it would have required the creation of an additional 12 calculations to define CLTP conditions with the new assumptions included. This significant effort was not warranted as the bounding analysis completed for EPU have addressed the desired CLTP analysis improvements. Results of a comparison between the single CLTP RWCU HELB case and the similar EPU HELB case is discussed in RAI 6 below. Re-analysis of all HELB breaks and an evaluation of affected EQ components have been completed; formal updating of EQ program documents are the only actions remaining. These actions are being performed coincident with EQ program updates required by EPU.

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EMCB RAI No. 6 (a)

The same paragraph on page 3-23, as above, in reference to the reactor water cleanup (RWCU), continues as follows:

> "For the break location that was analyzed during Rerate, new mass and energy release calculations considered additional blowdown sources that had not been considered in the previous 1996 analysis. This resulted in an increase in integrated mass release of about 90% and an increase in integrated energy release of 63 percent."

Confirm that the 90% and 63% increases are referring to the proposed EPU.

NSPM RESPONSE

~~The 90% and 63% increases are not referring to the proposed EPU. It is referring to the change in assumptions as noted in response to RAI 3 above rather than system operating condition changes resulting from EPU.~~

~~If the CLTP HELB cases were run using similar assumptions, the changes in mass and energy releases would be minor as a result of EPU.~~

Replace this text with the applicable portions of Table 26-2 - Reactor Building HELB Results. This table is provided in L-MT-12-114, Item 26.

As noted on PUSAR page 2-21:

> A review of the results from several recent EPU submittals concluded that, in most cases, environmental conditions are bounded by previous analyses, confirming that EPU produces relatively minor effects.

EMCB RAI RAI No. 6(b)

Please explain how the effects of the increased mass and energy release have been evaluated, include evaluations of pipe whip restraints and jet targets.

NSPM RESPONSE

Changes in mass and energy were evaluated for impacts on HELBs using the GOTHIC code. This allowed a determination of time histories for all plant areas to evaluate effects on temperature, pressure and flooding. Differential pressures between plant areas verified acceptable margins for structures such as block walls. The effects of changes to temperature, pressure and flooding have been evaluated for impact on the environmental qualification (EQ) of equipment. Upgrades to EQ files to document this evaluation ~~are in progress~~.

have been completed.

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RWCU pipe whip, jet impingement and safe shut down analyses following postulated pipe breaks or cracks are provided in USAR Appendix I. The RWCU high energy lines are located in the RWCU compartment, steam chase; MG set room, and the North West side of elevations 962' and 935' of the reactor building. There ~~are no~~ is one postulated breaks in ~~the MG set room and~~ the reactor building elevation~~s 962' and~~ 935' based on ~~seismic analysis.~~ HELB criteria. There are no pipe whip targets for the RWCU piping in the steam chase.

The safe shutdown evaluation for the RWCU compartment in Appendix I does not rely on pipe whip restraints or jet impingement shields to protect any equipment or structures. The effects of pipe whip and jet impingement in this area do not result in the loss of components required to mitigate the break and shut down the reactor. Therefore there is no impact on RWCU pipe whip and jet impingement due to EPU.

EMCB RAI No. 7

Page 2-37 states that: "The combination of stresses was evaluated to meet the requirements of the pipe break criteria. Based on these criteria, no new postulated pipe break locations were identified." For systems affected by the EPU, specifically steam (all EPU affected steam lines) and FW lines (including condensate), provide a pipe break analysis summary table (that includes the main steam increased turbine stop valve (TSV) closure transient loads in the analysis) which compares values at EPU and CLTP conditions and shows code equation stresses and CUFs compared to break limit for stresses and CUFs. Include pipe break locations and types selected for CLTP and EPU. Include lines inside and outside containment.

NSPM RESPONSE

Systems that have piping meeting the MNGP design basis criteria for classification as "High Energy" include Main Steam, Condensate, Feedwater, Residual Heat Removal (RHR), Core Spray (CS), High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Reactor Water Cleanup (RWCU), Off Gas, Control Rod Drive (CRD), Zinc Oxide Injection (GEZIP), and Standby Liquid Control (SLC). The parameters used for stress analysis in the high energy portions of these systems are unchanged due to EPU except in the Main Steam, Condensate, Feedwater, and GEZIP systems.

The Main Steam system analysis results including TSV closure loads are provided in the table below. The stress result for the Main Steam location with the maximum HELB break postulation equation result is also included in the table. The stress at that location does not meet (is less than) the current design basis criteria to require a postulated break. Hence, there is no Main Steam break outside containment postulated based on stress criteria. Other postulated break locations are based on configuration (e.g., terminal ends) which is not changed by EPU. Note that in the current design

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basis, specific HELB locations are not postulated inside containment. The current design basis does not include fatigue analysis of the Main Steam piping. Due to the revised analysis of the turbine stop valve closure loads, comparison to pre-EPU values is not meaningful.

The Main Steam evaluation results shown below are performed for the EPU pressure, temperature and flow parameters, including the TSV closure loads.

Main Steam Outside Containment - Maximum EPU Results (Highest Interaction Ratio):

Insert A

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

The maximum Feedwater system operating temperature is 397.7°F at EPU conditions for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment. This value is bounded by the original analysis temperature of 400°F. The design pressure for this portion of the Feedwater system is unchanged by EPU. ~~Therefore this piping is unaffected by EPU relative to HELB postulation.~~

based on the turbine heat balance

The feedwater piping and condensate piping from the condensate pump suction to the containment isolation valves will be re-analyzed during the Feedwater and Condensate pump and heater replacement modification process. High Energy Line Breaks and pipe whip restraints in the high energy portion of this piping will be evaluated at that time. GEZIP connections to the portion of the Feedwater system will be analyzed as part of the modification process. Details of the modifications to this piping are not yet finalized. The design will maintain stresses in the condensate and FW piping within code allowable limits of ANSI-B31.1-1977, including Winter 1978 Addenda and the requirements of USAR Chapter 12 including USAR Appendix I. Confirmation that the modifications are complete and meet the code allowables will be provided to the NRC. The FW and condensate system modifications are scheduled for completion ~~during RFO25 in 2011~~.

in 2013.

EMCB RAI No. 8

Enclosure 5, PUSAR Section 2.2.1.2, Liquid Line Breaks, on page 2-23 states that:

"The mass and energy releases for HELBs in the RWCU, FW, Condensate, CRD, Standby Liquid Control, and Zinc Injection (GEZIP) systems and instrument and sample

Following startup after installation of the new turbine and new FW heaters, the FW temperature increased by approximately 5°F for a portion of the FW piping, which was no longer bounded by the design temperature of 400°F for EPU operating conditions. Therefore, the affected FW piping design temperature was increased to 410°F and piping analyses were reperformed to account for the FW temperature change. All piping continues to meet code allowables.

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Insert A

Maximum Pipe Stresses (Outside Containment)

Load Combination	Service Level	Node	Stress (psi)	Allowable (psi)	Interaction Ratio
P + DW	A	TURD	7650	15000	0.51
TH Range	A	TURB	16618	22500	0.74
P + DW + TSV	B	TURC	12288	18000	0.68
P + DW + OBE*	B	X7A	14289	18000	0.79
DW+SRSS(TSV, SSE)*	D	X7A	21026	26325	0.80
HELB TH	N/A	TURB	16618	18000	0.92
HELB DW+TH+OBE	N/A	TURD	32631	30000	1.09**

*Excluding seismic category II pipe between Stop Valves and Turbine
**Indicates a HELB at this location, this load combination is used only to evaluate the need to assume a HELB and is not required to have an Interaction Ratio <1 to meet USAR requirements.

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EMCB RAI No. 12

Page 2-36 of the PUSAR states that, "The effects of the EPU conditions have been evaluated for the following piping [BOP] systems:" A list of piping systems follows this statement. On page 2-37 of the PUSAR, it is stated that, "These piping systems have been evaluated using the process defined in Appendix K of ELTR1 and found to meet the appropriate code criteria for the EPU conditions," when in fact evaluations of many of these systems, including RHR and MS, has not been completed, as shown by the submitted EPU LAR, see PUSAR Table 2.2-2d. In addition, Enclosure 8, Table 8-2 states that EPU planned modifications include, "Revise documentation to incorporate revised pressure and temperature ratings for specific piping systems affected by EPU. Modify supports as required by the analyses."

EMCB RAI No. 12(a)

The above PUSAR statements are not consistent. Please clarify the apparent inconsistency.

EMCB RAI No. 12(b)

The proposed EPU LAR indicates that some EPU evaluations have not been completed for the staff to review. The acceptability of the proposed EPU LAR will be determined based upon the results of the LAR evaluation reviews that are performed by the staff in accordance with the policies and procedures set forth in LIC-101, "License Amendment Review Procedures." Please provide a schedule of completion of these analyses and submittal of your evaluation results which shows that piping and pipe supports meet code allowable. Also, submit a schedule of completion for EPU required piping and pipe support modifications.

NSPM RESPONSE

Response to Part a

The referenced statement on page 2-37 indicating that pipe systems meet code requirements is intended to apply to piping stresses. Later on the same page, under the heading of "Pipe Supports," the structures listed on Table 2.2-2d are discussed. Based on the ELTR1 Appendix K methodology, the components listed were found to exceed code limits. Further, more detailed analysis may resolve some of these issues; others may require modification. This is consistent with the referenced statement from EPU LAR Enclosure 8 which indicates "supports" being modified as required by analysis.

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All piping and pipe support analyses have been completed.

Response to Part b

All piping and support evaluations required in ELTR1 have been completed using the methodology of Appendix K or by a more detailed analytical method.

~~Completion of piping support detailed analysis and/or modifications for items listed in Table 2.2-2d was scheduled for the 2009 outage RF024. The current status of work shown on PUSAR Table 2.2-2d is provided below:~~

~~**Table 2.2-2d Piping Components Requiring Further Reconciliation**~~

Item	System	Current Status
1	Main Steam (Outside Containment)	Refined analysis is complete, all piping components and supports meet code allowables.
2	Feedwater and Condensate (from condensate pump to the feedwater MO valves downstream of the HP Heaters), due to pending pump changes	Replacement of feedwater heaters, condensate and feedwater pumps will result in nozzle changes that will impact piping layout and analysis. Final design of these components is still in progress and is scheduled for completion in the 2011 refueling outage. NSPM will complete piping analysis and modifications as noted in response to RAI 7.
3	Torus Attached Piping	Refined analysis is complete, all modifications are complete with exception of one support, TWH-143, which will be completed on-line prior to implementation. Confirmation that modification of support TWH-143 is complete will be provided to the NRC prior to implementation of the EPU license amendment request.
4	RHR (BOP Condensate Service Water Lines)	Refined analysis is complete, all piping components and supports meet code allowables.
5	Cross Around Piping	Replacement of CARVs and CARV discharge piping during the RFO impacted this analysis. Prompt evaluations of field changes from this work are complete and all piping and supports meet code allowables.

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Table 2.2-2d is no longer required as piping analyses have been completed. The results from the piping analyses indicate that code limits are not exceeded.

Item	System	Current Status
~~6~~	~~CARV Discharge Piping~~	~~Replacement of CARVs and CARV discharge piping during the RFO impacted this analysis. Prompt evaluations of field changes from this work are complete, all piping and supports meet code allowables.~~

EMCB RAI No. 13

a) Provide a list of systems (inside and outside containment) for which temperature, pressure, flow and mechanical loads have been increased due to EPU. Please include OLTP and EPU values.

b) Provide a brief summary that shows the EPU maximum code equation stresses compared to CLTP for the affected systems. For MS, FW and condensate see RAI 17, below. Include fatigue evaluation CUFs, where applicable. It is noted, that although the tables in Section 2.2 of the PUSAR include, for some BOP systems, the percentage increases for pipe stresses and pipe support loads, varying from 9 to 72 percent increases, due to temperature or pressure increases, these percentages are not indications that piping and pipe supports meet code equation allowable values, without providing maximum resulted values compared to code allowable.

NSPM RESPONSE

The system temperature, pressure, and flow changes due to EPU that are not bounded by the parameters used in the existing stress analyses are shown in Table 1, below.

The maximum code equation stresses for Main Steam at EPU conditions are summarized in Table 2, below. The maximum code equation stresses for BOP systems are summarized in Table 3, below. The current design basis does not include fatigue analysis of the Main Steam piping.

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Table 1
MNGP EPU Piping Analysis Input Parameter Changes

Item	Parameter	OLPT	CLTP Value	EPU Value
Inside Containment				
1	**Main Steam**			
	Flow (Lbm/hr)	6.78E+6	7.262E+6	~~8.524~~E+06 8.534
2	**Feedwater,** from outboard containment isolation valves (FW-91-1 and FW-91-2) to RPV			
	Flow (Lbm/hr)	6.83E+6	7.313E+06	~~8.575~~E+06 8.452
3	**Core Spray** (CS)			
	Temperature (˚F)	180	196.7	212
Outside Containment				
1	**Main Steam**, upstream of TSV			
	Flow (Lbm/hr)	6.78E+6	7.262E+06	~~8.524~~E+06 8.534
2	**Feedwater,** From MO-1614/1615 to FW-91-1/FW-91-2			
	Flow (Lbm/hr)	6.75E+6	7.235E+6	~~8.497~~E+06 8.362
	From pumps to MO-1614 and MO-1615			
	Temperature (˚F)	400	400	~~Note 1~~ 410
	Pressure (psig)	1550	1550	~~Note 1~~ 1550
	Flow (Lbm/hr)	6.75E+6	7.235E+6	~~8.497~~E+06 8.362
3	**Condensate**, from Condensate pump suction to Feedwater pump			
	Temperature (˚F)	302	310	~~Note 1~~ 321
	Pressure (psig)	434	434	~~Note 1~~ 450
	Flow (Lbm/hr)	6.75E+6	7.235E+6	~~8.497~~E+06 8.362
4	**Torus Attached Piping** (CS, HPCI, RCIC, Note 2)			
	Temperature (˚F)	180	196.7	212

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Table 1
MNGP EPU Piping Analysis Input Parameter Changes

Item	Parameter	OLPT	CLTP Value	EPU Value
5	**Emergency Service Water**, ECCS Pump Room Ventilation Units (V-AC-4/5) Outlet Lines			
	Temperature (˚F)	120	120	122
6	**Extraction Steam**			
	Operating Temperature (˚F)			
	To Heater E-11	177	183	~~186~~ 183
	To Heater E-12	236	246	~~253~~ 249
	To Heater E-13	313	315	~~323~~ 318
	To Heater E-14	344	348	~~358~~ 356
	To Heater E-15	386	396	~~407~~ 400
	Design pressure (psig)			
	To Heater E-11	-8	-7	~~-6~~ 50
	To Heater E-12	8	13	~~16.8~~ 50
	To Heater E-13	66	68	~~79~~ 92
	To Heater E-14	111	117	~~136.4~~ 158
	To Heater E-15	197	220	~~254~~ 269
	Flow (Mlbm/hr)			
	To Heater E-11	0.404	0.592	~~0.700~~ 0.681
	To Heater E-12	0.371	0.423	~~0.490~~ 0.478
	To Heater E-13	0.443	0.444	~~0.525~~ 0.511
	To Heater E-14	0.806	0.893	~~1.164~~ 1.134
	To Heater E-15	0.388	0.443	~~0.548~~ 0.531
7	**Heater Drains**			
	Operating Temperature (˚F)			
	From Heater E-11	173	180	~~183~~ 104
	From Heater E-12	236	243	~~250~~ 187
	From Heater E-13	241	248	254
	From Heater E-14	315	318	~~327~~ 325
	From Heater E-15	343	349	~~359~~ 358

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Table 1
MNGP EPU Piping Analysis Input Parameter Changes

Item	Parameter	OLPT	CLTP Value	EPU Value
	Design pressure (psig)			
	From Heater E-11	-8	-7	~~-6.6~~ -6.3
	From Heater E-12	7	12	~~15~~ 19
	From Heater E-13	54	64	~~74~~ 87
	From Heater E-14	96	110	~~128~~ 149
	From Heater E-15	184	215	~~238~~ 269
	Flow (Mlbm/hr)			
	From Heater E-11	2.52	2.80	~~3.43~~ 3.34
	From Heater E-12	2.04	2.20	~~2.73~~ 2.65
	From Heater E-13	1.67	1.78	~~2.24~~ 2.18
	From Heater E-14	1.22	1.34	~~1.71~~ 1.67
	From Heater E-15	0.39	0.44	~~0.55~~ 0.53
8	**Service Water**			
	Inlet Temperature (°F)	85	90	90
9	**Cross Around**			
	Temperature (°F)	387	393	~~407~~ 410
	Pressure (psig)	197	214	~~254~~ 269
	Flow (Mlbm/hr)	6.33	6.75	~~7.91~~ 7.73
10	**Cross Around Relief Valve Inlet**			
	Temperature (°F)	381	389	~~403~~ 410
	Pressure (psig)	182	204	~~242~~ 269
	Flow	5.66	6.05	~~7.03~~ 6.87
11	**Moisture Separator Drain**			
	Temperature (°F)	383	392	~~403~~ 410
	Pressure (psig)	202	204	~~242~~ 269
	Flow (Mlbm/hr)	0.6728	0.7011	~~0.877~~ 0.853

Note:
1. Due to the planned extensive piping modification to the Condensate and Feedwater systems, this piping is analyzed for EPU condition changes as part of the modification process (Reference response to RAI 7).
2. Torus attached RHR piping is currently analyzed at a temperature higher than the peak torus temperature and is therefore not changed by EPU.

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EMCB RAI No. 17

Steam flow and feedwater flow will increase as a result of the CPPU implementation. The load due to the TSV fast closure transient is used in the design of the MS piping system. Page 2-31 states that "Due to the magnitude of the TSVC transient load increase [at EPU], the transient event was reanalyzed. The main steam piping was then reanalyzed using this revised load definition."

a) Provide a quantitative summary of the MS and associated piping system evaluation (inside and outside containment), including pipe supports, that contains the increased loading associated with the TSV closure transient at EPU conditions, along with a comparison to the code allowable limits. For piping, include maximum stresses and data at critical locations (i.e. nozzles, penetrations, etc), including fatigue evaluation CUFs, where applicable. For pipe supports, state the method of evaluation for EPU conditions and confirm that the supports on affected piping systems have been evaluated and shown to remain structurally adequate to perform their intended design functions. For non-conforming piping and pipe supports, provide a summary of the modifications required to ensure that piping and pipe supports are structurally adequate to perform their intended design functions and the schedule for completion of these modifications.

b) For FW and condensate, please respond as in part (a) of this RAI.

NSPM RESPONSE

Response to Part a
The Main Steam system piping analysis results, including TSV closure loads are summarized below. The piping system was evaluated (by re-analysis versus scaling) using requirements from the existing code of record. The supports in the Main Steam piping remain adequate to perform their intended design functions. ~~An updated status for PUSAR Table 2.2-2d is provided in response to RAI 12, Part b above.~~ There are no non-conforming pipes or supports requiring modifications on the main steam system.

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Maximum Support Loads
MS Relief Valve Discharge Line Support RV25A-H1 (spring hanger)

Load Condition	Service Level	Node	Max Load lb	Allowable lb	IR Max/Allow	Min Load lb	Allowable lb	IR Allow/Min
DW+TH+ SRSS(TSV,SRV,OBE)	B	285	1341	1344	0.998	1162	780	0.671

Deleted per Item 11

~~**Main Steam Outside Containment**~~
~~**Maximum EPU Results (Highest Interaction Ratio):**~~

~~**Maximum Pipe Stresses**~~

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

~~**Maximum Turbine Loads**~~

Load Combination	Service Level	Node	Mx ft-lb	Allowable ft-lb	Ratio Mx/Allow	Mz ft-lb	Allowable ft-lb	Ratio Mz/Allow
DW	B	*	32244	413000	0.078	171446	722000	0.237
DW + TH	B	*	271321	413000	0.657	302310	722000	0.419

~~*Note: Loads from all turbine nodes were combined~~

~~**Maximum Support Loads**~~
~~**Main Steam Line Support PS-16, Node 283**~~

Load Condition	Service Level	Component	Max Load lb	Allowable lb	IR Max/Allow
DW+TH+SRSS(TSV,SRV,OBE)	B	Anchor bolt	20026	20731	0.966

Response to Part b

The maximum Feedwater system operating temperature is 397.7°F at EPU conditions for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment. This value is bounded by the original analysis temperature of 400°F. The design pressure for this portion of the Feedwater system is unchanged by EPU. Therefore this piping is unaffected by EPU relative to HELB postulation. The current design basis for Feedwater piping analysis does not include fluid transient analysis. The stress analyses for the Feedwater piping from the outboard

Item 26

Following startup after installation of the new turbine and new FW heaters, the FW temperature increased by approximately 5°F, which was no longer bounded by the design temperature of 400°F. Therefore, the FW design temperature was increased to 410°F and piping analyses were reperformed to account for the FW temperature change. All piping continues to meet code allowables.

containment isolation valve to the containment and inside containment are therefore unaffected by EPU.

The feedwater piping and condensate piping from the condensate pump suction to the containment isolation valves will be re-analyzed during the Feedwater and Condensate system modifications (reference response to RAI 7).

EMCB RAI No. 18

In accordance with Section 2.2.2 of the PUSAR, the main steam and associated piping system structural evaluation was performed to justify the operation of these systems at EPU conditions. This evaluation showed that one small bore branch line did not meet the displacement criteria. PUSAR further states that, "Additional detailed analysis will be performed to qualify this line or the piping modified prior to operation at EPU conditions."

a) Provide identification of the small bore branch line (size, system, location, function).

b) Describe the required displacement limits and their bases.

c) Since this piping analysis, with potential piping and or support modifications, is required for EPU, please discuss the reasoning for not including this information in your application. Also, indicate when necessary modifications, as needed, will be completed.

NSPM RESPONSE

a) The branch line is a 1 inch instrument sensing line located inside the primary containment. The line connects one of the differential pressure sensing ports on the D steam line flow restrictor to a containment instrument piping penetration. This line is used for flow sensing in main steam line D and serves a safety related input function to the high flow Group 1 Containment Isolation logic that will automatically isolate the MSIV's in the event of a main steam line break.

b) A differential displacement of 1/16 inch for branch connection points was used as screening criteria in the piping analysis. Those in excess of 1/16 inch were noted as outliers needing further evaluation. The basis for the 1/16 inch criteria is:

1. The 1/16 inch displacement produces an insignificant stress in the branch line which is typically supported by a standard deadweight span (span length from run pipe nozzle connection to first support on the branch).

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SBPB RAI 2.5-1

Table 2.2-1, "Liquid Line Breaks," indicates the flooding elevation within the reactor building and turbine building will increase. Section 2.2.1.2 of the PUSAR states that the reactor building and turbine building pressure, temperature, flooding and relative humidity profiles at EPU conditions were evaluated for the effect on equipment qualification as discussed in Sections 2.2.5 and 2.3.1. Explain the criteria used to evaluate the effect of increased flooding elevations on safe shutdown equipment and the basis for the acceptance criterion for flooding elevation.

NSPM RESPONSE

~~MNGP chose not to perform its flooding analysis using an equipment list detailing those components required for safe shutdown and core cooling. Instead,~~ MNGP ~~assumed~~ uses the criteria that if at least one division of an engineered safety system ~~was~~ is demonstrated to be operable during ~~an internal~~ a HELB flooding event, then the requirement for safe shutdown and core cooling ~~was~~ is met. This evaluation is provided in USAR Appendix I Section I.5. ~~Additional detail is provided in Enclosure 17 (Task Report T1004) to the EPU License Amendment Request L-MT-08-052 (Accession No. ML083230111).~~

~~An increase in submergence level for four valves in the reactor building steam chase is discussed in our response to EEEB Draft RAI 3 submitted in NSPM letter L-MT-09-045.~~ L-MT-12-114, Item 27, Table 27-3 provides a comparison of Reactor building HELB water levels at CLTP and EPU conditions. These levels were used for the evaluation of EPU conditions for EQ component compliance. The conclusion is that there is no adverse impact on the safety functions of the associated components and systems.

The EPU HELB analysis identified a deficiency in the existing HELB analysis in the turbine building. The existing analyses failed to consider the actuation of the fire sprinklers in the condenser bay and the resultant flooding on the lower 4kV equipment room. A flood barrier had previously been installed which assured current operability and past operability was evaluated and submitted to the NRC as LER 2008-01 in L-MT-08-019 (Accession No. ML080910155). The current design does not require change for operation under EPU conditions. No other turbine building components required for safe shutdown under HELB are adversely impacted by flooding.

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NRC RAI No. 4

In Note 4 of Section 3.4.3, the licensee stated that the Rockbestos cables could be submerged and these cables are qualified for 18 hours in a submerged condition. The staff requests the licensee to provide the basis for the 18 hour criteria for qualification, and also confirm that these cables are not submerged more than 18 hours and that these cables will perform their safety function during and post submergence.

NSPM Response

The Rockbestos cables used in this application are not submerged more than 1 hour and will perform their safety function during and post submergence.

The event for which the predicted worst-case 1.7 foot submergence level in Reactor Building Volume 32 is based upon is the "RWCU-B-30-R0" case, all other HELB events affecting this location have less than 6-inches of submergence. The peak submergence level in Volume 32 is reached at 70 seconds, and the level reduces to 0-inches at 1861 seconds. The cables are therefore predicted to be submerged for less than 1 hour. The qualification of the cables is based on Rockbestos Reports QR-5804/5805. The 18-hour (minimum of QR-5804 testing) soak following the simulated LOCA testing was conducted at room temperature with tap water. The cable testing was compliant with IEEE Standard 383-1974 and included mandrel re-bending.

The specific values for submergence and duration of submergence reported in the RAI response above have changed. However, these submergence values have been reduced for EPU, making the values for level and soak time reported above following a HELB event bounding and conservative. See L-MT-12-144, Enclosure 1, Item 27, Table 27-3 for revised HELB submergence values.

Item 27

NRC RAI No. 7

In Note 3 of Section 3.4.4, the licensee stated that 1) the ITT Royal cable is only installed in local instrument panels in RB Volumes 14, 18, 19, 22, and 33 and that these cables are qualified for peak temperature of 211°F, 2) the EPU HELB temperatures in RB Volumes 14, 18, 19, 22, and 33 are 174.7°F, 209.7°F, 209.3°F, 184°F, and 211.7°F, respectively, and 3) using engineering judgment that the HELB event would not exceed the test level of 211°F as only a 0.7°F difference between the test level and the predicted values exists. The staff requests the licensee to confirm that these cables are not exposed to any temperature rise or hot spots in the panels. If these cables are exposed to hot spots in the panels, then the staff requests the licensee to provide the qualification basis for these cables.

NSPM Response

The cables are not exposed to any temperature rise or hot spot conditions in the panels. The instrument panels discussed are not enclosed cabinets but rather the original General Electric plant instrument racks on which various pressure transmitters, switches, or other system instrument monitoring devices are located. ~~As indicated by the data presented in Note 3 of Section 3.4.4 of T1004, Revision 1, the ITT Royal cable is qualified with margin (1.3°F minimum) for use in Reactor Building Volumes 14, 18, 19, and 22.~~ The ITT Royal cable has been identified as original plant equipment and is environmentally qualified in accordance with the DOR Guidelines.

~~The predicted peak EPU HELB temperature in Reactor Building Volume 33 exceeds the test temperature of 211°F by only 0.7°F. The ITT Royal cable is only used on pressure switches PS-5-12A/B on instrument rack C-55 (25-5) in Reactor Building Volume 33. Plant drawing NX-7829-31-1 provides the layout of the instrument rack which shows that only non-heat producing measuring instruments are mounted on the rack (see picture below). The drawing also specifies the use of individual conduit routings to the local junction box for each device. PS-5-12A/B have normally closed contacts and serve to energize relays 5A-K4A and 5A-K4B, respectively. These relays are General Electric type HFA and are the only load on PS-5-12A/B pressure switches (and thus the ITT Royal cable). As such, internal cable heating is insignificant.~~

ITT Royal cable has been evaluated to the revised EPU temperatures for the reactor building discussed in L-MT-12-114, Item 27, tables 27-1 and 27-4. These cables remain qualified for EPU conditions.

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NRC RAI No. 13(c)

The licensee stated the following:

"As a result of accepting twice the thermal aging for the Viton as represented by the 30 day exposure at 210°F for specimen # 3, the limiting life component has changed from the Viton to either "M" or non "M" style internal switch for all ASCO pressure switches."

The staff does not understand the intent of this discussion. The staff requests the licensee to provide a detailed explanation of the above statement.

NSPM Response

The enclosed statement on Page A2-7 of Attachment A2 was intended to indicate the qualified life change basis for EQ File 98-006, ASCO Pressure Switches and not be a part of the formal EQ File revision. The attachment was principally provided to support the interim post-accident operating time evaluation of the ASCO pressure switches under EPU conditions. It was recognized during the assessment of EPU conditions that there was a need to incorporate changes to normal plant ambient temperature conditions. The normal plant ambient temperature condition has increased for most plant areas in which ASCO pressure switches are located. The +5°F or +10°F designators within the enclosed text on Page A2-7 of Attachment A2 indicate the normal ambient temperature increases.

As part of the preliminary EQ file revision under the EPU project, it was discovered that specimen number 3 in the ASCO test (ASCO Report AQR-101083) included Viton seals that were aged for the indicated 30 days at 210°F. This represents a departure from the current EQ File of record which only credited 8.5 days of aging for the Viton material, but is a legitimate conclusion reached from further review of the ASCO testing. The current EQ file basis was conservative in choosing to use the Viton aging of 8.5 days.

The "doubling" of the aging time only applies to the ASCO pressure switches with the "M" style micro switch. In these cases, the limiting component shifted to the 15 days of aging on the "M" style micro switch versus the "newly accepted" 30 days of Viton aging. The aging for the non-M style micro switches is 30 days of aging for either the Viton or micro switch. Regardless, in all cases, the life limiting component switched from the Viton to the micro switch sub-component. The use of the wording "twice the thermal aging" or "doubling of the aging" was somewhat an unclear phrase to describe the effect of shifting the life limiting component from the Viton to the micro switch. Again, the wording within the box on Page A2-7 of Attachment 2 is not intended and will not be included as part of the formal EQ file revision. The conclusions and re-calculated lives and post-accident operability discussions otherwise presented in Attachment A2 stand correct based on the data presented therein.

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Independent of the EPU project, ASCO pressure switches PS-4664 through PS-4672 ~~were being replaced in Refueling Outage 24~~ as a precaution to ensure they would not exceed their qualified life based on the current EQ file analysis basis.

have been replaced

NRC RAI No. 13(d)

Based on our review of the licensee's ASCO pressure switch evaluation, the staff understands that the licensee has used thermal aging to qualify for the loss-of-coolant accident (LOCA) and/or Post-LOCA conditions. The staff requests the licensee to provide a detailed explanation and justification for using this methodology for qualifying these switches.

NSPM Response

The complete analysis of ASCO pressure switches required for HELB mitigation included utilization of the ASCO accident testing of 10 hours at 210°F of steam testing. To minimize the volume of the EPU Task Report, only those pages from the current EQ file being revised or amended in the interim for post-LOCA evaluation under EPU conditions were included.

In the case of EQ File CA-98-006 for ASCO pressure switches; there is an additional appendix that evaluates HELB accident conditions. The ASCO pressure switches required for HELB mitigation have operating times less than and bounded by the 10 hours of accident testing conducted by ASCO (as reviewed in Table 3.4.4-1 of Task Report T1004, Revision 1).

For long-term post-LOCA evaluation, thermal aging alone was used to envelop the qualified life plus post-LOCA operating time periods. Under a design basis LOCA, the Reactor Building is not a harsh steam environment, but is heated gradually. As a bounding location, the torus compartment, given its significant heat source (suppression pool), is the area of the Reactor Building most affected by post-LOCA heat-up conditions. Figure 3.4.6-1 of the task report shows a composite of the post-LOCA heat-up conditions of the Reactor Building. This figure is dominated by the post-LOCA conditions for the torus compartment for either EPU or CLTP plant conditions. The curve shown reveals the general slowly changing nature of post-LOCA heat-up conditions in the Reactor Building.

The interim evaluation for EPU conditions for the ASCO pressure switches as provided in Attachment A2 focused mainly on post-LOCA heat-up conditions of the torus compartment, but also provided details of other Reactor Building post-LOCA heat-up conditions as well. For EQ purposes, post-LOCA and HELB events are not concurrent. The analysis presented results in a reduction of the qualified life by the amount of expected degradation imparted to the equipment under LOCA operation. In all cases,

ENCLOSURE 3

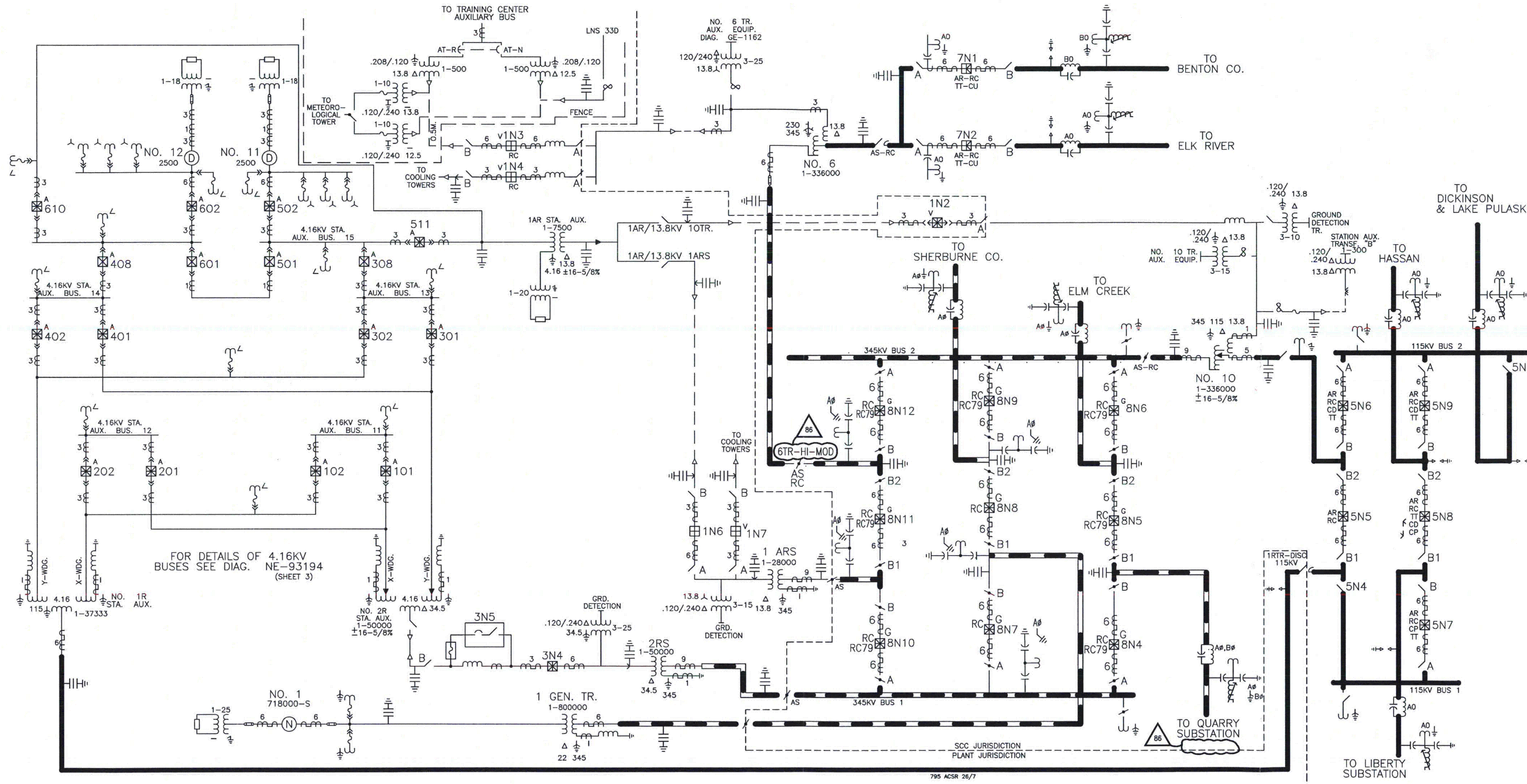
ADDITIONAL SUPPLEMENTAL MATERIALS

Item 1 – Drawing NH-178635 – USAR Section 15 figure update
– Email dated September 24, 2015 from Vikram Godbole of MISO
Item 17 – EOP flow chart C.5-2007, Revision 17

7 pages follow

NH-178635

REVISIONS
83 EC16880
DWN: TJF 03-14-11 CHK: JAZ 3-21-11 APPR: AJD 3-21-11
84 EC11018,12332 15425
DWN: TJF 05-11-11 CHK: DDW 6-9-11 APPR: AJD 6-9-11
85 EC16881
DWN: BAT 6-27-11 CHK: WGN 6-27-11 APPR: AJD 6-27-11
86 EC17014
DWN: JJP 3-6-12 CHK: PEB 3-6-12 APPR: SCP 3-6-12



This map/document is to assist employees in the performance of their jobs. Your personal safety is provided for by using safety practices, procedures and equipment as described in safety training programs, manuals and SPAR's.

MONTI CAD DWG 'A'

SUBSTATION 1-LINE DIAGRAM

MONTICELLO NUCLEAR GENERATING PLANT

Xcel Energy

SCALE: NONE

REV 86

NH-178635

Item 1

Email Addresses have been redacted to protect personally identifiable information.

From: Vikram Godbole [VGodbole@misoenergy.org]
Sent: Monday, September 24, 2012 5:06 PM
To: 'Cindy Hammarlund (MP)'[redacted] Josephine Daggett; Alan Burbach; J Mayhan; Mike Riewers; Dan Burns; Aaron VanderVorst
Cc: Karan Joshi; Jesse M. Phillips; Eric Laverty; Jerome Fohey; Jeanna Furnish
Subject: DPP 1,2, 6 West Area Restudy evaluation for projects with existing GIAs

All,

MISO has finished the evaluation of DPP 1,2, and 6 West area projects with Permanent GIAs. Please read this email bottoms to for a quick refresher on what this is about.

Please find attached 3 attachments that include the following:

1. Stability Analysis
2. NRIS analysis
3. Per project summary results.

<u>High Level Summary:</u>

1. Most of the projects were not impacted at all, as noted in individual project tabs in the "DPP 1 2 6 per project Result Summary.xlsx" spreadsheet.
2. There were a few projects where the network upgrades identified in earlier GIAs dropped off and are not required anymore.
3. There were a couple of projects for which a few new constraints have showed up and MISO will work with the affected TOs to get those resolved.
4. For most of the projects the NRIS levels have gone up without any new network upgrades. For a couple of them there may be network upgrades required to go to 100% NRIS levels.

<u>Next Steps:</u>

1. MISO will setup a in-person meeting at the MISO St Paul office to discuss the results with the Interconnection Customers. Please provide your availability for October 9th from 9AM-Noon Central time.
2. MISO will share the cases and all associated files by the end of this week.

Email Addresses have been redacted to protect personally identifiable information.

If you have any general questions about your specific project, please do not hesitate to call me. If you have questions on how the analysis was performed, please contact Karan Joshi and Jesse Phillips because they will provide you with a much better and an accurate answer. ☺

Regards,

Vikram Godbole

Manager, Transmission Facilities Planning
MISO
317-249-5376 O | 317-753-6966 C | 317-249-5358 F

-----Original Message-----
From: Vikram Godbole
Sent: Tuesday, September 04, 2012 1:18 PM
To: 'Cindy Hammarlund (MP)'; [redacted]
Cc: Karan Joshi; Jesse M. Phillips; Eric Laverty
Subject: DPP 1,2, 6 West Area Restudy evaluation for projects with existing GIAs

All

As you all are aware, many Generation Interconnection projects, that were studied under the old GIP, either withdrew or did not true up to the Queue Reform 3 M2 milestones by the June 28th,2012 transition deadline. Many of these withdrawn projects were either higher queued or queued in the same group as your individual projects. MISO has discussed the transition analysis a few times at the IPTF meetings in the 2nd quarter of 2012, specifically the June 26th, 2012 IPTF meeting. During the transition analysis discussion, it was pointed out that MISO would analyze all the projects with an executed IAs first to determine if any restudy was warranted. Any changes emanating from that analysis would become the higher queued study assumptions for the August 2012 DPP Studies. The details of the transition analysis can be found by clicking on the following link

https://www.misoenergy.org/_layouts/MISO/ECM/Redirect.aspx?ID=133157 .

The information below is pertinent to only those projects that were part of the old DPP 1-6 cycles that eventually executed IA. Any other projects that did not have a GIA but trued up with the M2 milestones are part of the DPP August 2012 study. Therefore, if you don't have a project with an executed IA that was part of any of the DPP cycles, you can ignore the rest of the email.

According to the transition plan, MISO was supposed to communicate to the Interconnection Customers if a restudy of their project was required. Although MISO has not conclusively made the restudy determination yet, we still feel it is important to let the Interconnection Customers know about the status of each potentially impacted projects.

1. MISO has finished the Group 5 analysis (post withdrawal)
    a. This has changed the study assumptions that were modeled for your project when it was first studied.
    b. MISO has incorporated the latest changes in its preliminary analysis that is being run currently

2. Removed all withdrawn projects from DPP 1-6 cycles that were base case assumptions in your project's studies.
    a. This will reduce the list of higher queued assumptions under each of your GIAs
    b. We have completed the ERIS analysis for the DPP 1, 2 and 6 West area projects. Based on a very cursory look, there is no evidence at this point that demonstrates any additional upgrade requirement. If anything changes, we will notify when we officially post and share the results with the affected Interconnection Customers.
    c. NRIS and Stability analysis are underway and should be done in the next 10 days.

3. Once we have all the results ready, we will setup a conference call to discuss the impact on each project. If the group prefers an in-person meeting in St Paul, that can be accommodated too.

Stay tuned and we will send another email update sometime early next week. Once again, I would like to thank each and every one of you for being patient through the entire Queue Reform Implementation process.

The list of impacted projects to follow soon.

Regards,
Vikram

Excel Engineering, Inc.
5267 Program Ave • Second Floor • Saint Paul, MN • 55112 • Bus: (763) 571-5008 • Fax: (763) 571- 6002

Providing Engineering Solutions to Help You Achieve Your Business Objectives

September 19, 2012

Karan Joshi
Midwest Independent System Operator (MISO)
Transmission Access Planning

Vikram Godbole
Midwest Independent System Operator (MISO)
Transmission Access Planning

RE: DPP Permanent GIA Stability Analysis

Overview:
Excel Engineering performed stability analysis on the DPP Projects with Permanent Generator Interconnection Agreements prior to the August 2012 DPP Cycle. The generators which were included in this analysis are shown below in Table 1.1. The purpose of the study was to identify any possible adverse impacts these project may have on local or regional system stability.

Table 1.1 Permanent GIA DPP Projects

Project	MW		Project	MW
G667	13.0		G968	100.0
G685	20.0		G971	20.0
G741	8.0		H007	41.0
G746	200.0		H061	39.9
G788	49.0		H062	39.0
G873	20.0		H067	40.0
G875	80.0		H078	121.0
G929	60.8		H092	60.0
G930	60.0			

The power flow model used for this analysis was the MTEP 2015 Shoulder Peak model, "Post_Restudy_TAP12_2015SH_08_22_2012.sav", which was developed for the DPP analysis. A stability snapshot was developed for this model using the MTEP 2017 package dynamic response data.

Approximately four local disturbances, two single line to ground and two three-phase disturbances, were developed and analyzed for each of the generators. Additionally, 49 standard and 345 kV regional faults were analyzed. Fault definitions and results are available upon request.

The results indicated that there are no adverse impacts to system stability due to the addition of the 17 proposed generators which were analyzed. The results did indicate some possible model issues with the Mackinac Straits back-to-back DC model. This will be addressed prior to the DPP analysis.

Let us know if you have any comments or questions.

Craig Thingvold, P.E.
Excel Engineering, Inc.

MN Registration # 49049

LaShel Marvig, P.E.
Excel Engineering, Inc.

MN Registration # 42891

PROJECT INFORMATION
Project Nu G929
Orignal DP DPP 1 Nov 2008
Interconne XEL

PROJECT NOT IMPACTED

This information is from an attachment to the Email.

THIS PAGE IS AN OVERSIZED DRAWING OR FIGURE,

THAT CAN BE VIEWED AT THE RECORD TITLED: MONTICELLO NUCLEAR GENERATING PLANT EMERGENCY OPERATING PROCEDURE DRAWING NO. C.5-2007 REVISION 17

WITHIN THIS PACKAGE...

D-01X