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U. S. Nuclear Regulatory Commission
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Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License
License No. DPR-22

Monticello Extended Power Uprate: Response to NRC Probabilistic Risk Assessment (PRA) Licensing Branch Request for Additional Information (RAI) dated April 29, 2009 (TAC No. MD9990)

References:

1. NSPM letter to NRC, License Amendment Request: Extended Power Uprate (L-MT-08-052) dated November 5, 2008 (Accession No. ML083230111)
2. Email P. Tam (NRC) to G. Salamon and K. Pointer (NSPM) dated March 18, 2009, "Monticello-Draft RAI from the PRA Branch re. Proposed EPU Amendment (TAC MD9990)"
3. Email P. Tam (NRC) to G. Salamon and K. Pointer (NSPM) dated April 29, 2009, "Monticello-Revised RAI from the PRA Branch re. Proposed EPU Amendment (TAC MD9990)"

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), requested an amendment to the Monticello Nuclear Generating Plant (MNGP) Renewed Operating License (OL) and Technical Specifications (TS) to increase the maximum authorized power level from 1775 megawatts thermal (MWt) to 2004 MWt in Reference 1.

On March 18, 2009, the U.S. Nuclear Regulatory Commission (NRC) Probabilistic Risk Assessment (PRA) Licensing Branch provided twelve Requests for Additional Information (RAIs) in Reference 2. Following a conference call with NSPM on April 17, 2009, the PRA Licensing Branch provided revised the RAIs in Reference 3. Enclosure 1 provides the NSPM response to these RAIs. In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Minnesota Official.

Summary of Commitments

This letter makes no new commitments and does not change any existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on May 29, 2009.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT

NSPM RESPONSE TO PRA LICENSING BRANCH RAIs
DATED APRIL 29, 2009

NRC RAI No. 1

The NRC staff's evaluation of the individual plant examination of external events (IPEEE) report specifically specified that additional analysis is necessary to identify if single pump success is adequate for the Service Water System. The licensee's response dated 2/4/09, NRC Review Item (7), indicates that the current internal events Monticello Nuclear Generating Plant (MNGP) PRA Model of Record assumes that a single Service Water pump is adequate to successfully accommodate post transient cooling requirements. Describe how MNGP confirms that the single Service Water pump assumption modeled in the PRA is adequate for post-extended power uprate (EPU) requirements.

NSPM RESPONSE

The only loads modeled in the current PRA that have service water dependency include CRDH Pumps (via the Reactor Building Closed Cooling Water System), Feedwater Pumps, Mechanical Vacuum Pump seal water cooling, and alternate water makeup to the condenser hotwell. In addition, two of three instrument air compressors have a PRA dependency but this load demand will be eliminated when new air compressors are placed in service prior to EPU operation.

It is important to note that the cooling for the safety related loads is not supplied by the Service Water (SW) System. Cooling for safety related systems is provided by other systems that are separate and independent of the SW System. For instance, Emergency Diesel Generator cooling is accomplished by the EDG Emergency Service Water System (EDG-ESW). Motor and room cooling for the ECCS systems is accomplished by the Emergency Service Water (ESW) System. Emergency decay heat removal cooling is provided by the RHR Service Water System (RHRSW).

Table 1 indicates the before and after service water demand for PRA-dependent loads for both CLTP and EPU. The net change to the PRA dependent loads is not significant with respect to the capacity of a single SW pump.

Table 1 - PRA Dependent Loads

Load	**Demand CLTP**	**Demand EPU**	**Net Change**
Reactor Feedwater Pump Motors (2)	300 gpm	270 gpm	-30 gpm
Reactor Feed Pump Lube Oil Coolers (2)	10 gpm*	52 gpm	+ 42 gpm
CRDH Pumps (2) (RBCCW load shown)	4 gpm	4 gpm	
Instrument Air Compressors (2)	20 gpm*	0 gpm	- 20 gpm
Mechanical Vacuum Pump	50 gpm	50 gpm	
Hotwell Makeup**	354 gpm	400 gpm	+46 gpm
Net Change			+ 38 gpm
* Estimate ** Based on LOFW event demand at EPU adjusted for decay heat.			

From Table 1 above the change in post transient loads with PRA dependency is not significant. In practice, operators will vary the load demand on service water to maintain equipment temperatures. Even if the actual PRA load demand was tripled over the design demand, the SW pump capacity margin would continue to be substantial at EPU conditions as the minimum capacity of a single SW pump is 8000 gpm. The 8000 gpm value is derived from the pump design point. In practice, 10% additional flow (8800 gpm total) is available per the Hydraulic Institute Standards.

Table 2 below indicates the before and after service water loads for non-PRA loads that are changing due to EPU. As with the PRA loads, the net change to the non-PRA loads is not significant with respect to the capacity of a single SW pump. The EPU values are based on estimates for in-progress modifications, but are not expected to change significantly.

Table 2 - Non-PRA Dependent Loads

Load	**Demand CLTP**	**Demand EPU**
Isophase Bus	45 gpm	80 gpm
Condensate Pump Motors (2) (thrust bearing only)	NA	50 gpm*
Main Generator Hydrogen coolers (E-8A, B, C, & D)	1205 gpm	1500 gpm*
Stator Winding Liquid Coolers (E-10A & B)	700 gpm	893 gpm*
Net Change		573 gpm*
*Estimate		

For post-transient operations at CLTP and at EPU with a single SW pump available, existing procedures require operators to remove unnecessary equipment from service and to direct service water flow to essential equipment if needed. The need for a manual reduction in system load, if any, is dependent on the hydraulic conditions of the SW system at the time of the event. The post scram procedure requires operators to monitor the temperature of equipment served by the SW System. The large SW loads are non-safety related and are non-PRA dependent (e.g. Generator Hydrogen Coolers, Stator Winding Coolers, Turbine Lube Oil Coolers). These loads will be either automatically reduced (Generator Hydrogen Coolers) or can be reduced by simple valve manipulations per procedure without affecting PRA dependent loads. If necessary, these actions, which are not dependent on changes due to EPU, can reduce the non-PRA loads to a fraction of the 8800 gpm single SW pump capacity.

Given the above, the change in the service water system loads at EPU conditions is not significant with respect to supplying sufficient service water flow to the PRA dependent loads during post-transient conditions, and the post-transient cooling requirements continue to be met by a single service water pump with adequate margin.

NRC RAI No. 2

EPU Safety Analysis Report (SAR), Section C.3, discusses assessments for the 2003 Monticello PRA model against the American Society of Mechanical Engineers (ASME) standard and NRC draft Regulatory Guide DG-1122 performed by Applied Reliability Engineering Inc, (ARE), in early 2004. This section does not provide information on findings and comments related to this assessment. Please identify and discuss the dispositions of any open items identified as a result of the 2004 ARE assessment, how they were subsequently addressed in the MNGP PRA, and provide justification for those open-items that affect the EPU.

NSPM RESPONSE

All open items identified in the 2004 Applied Reliability Engineering (ARE) Self Assessment of the 2003 version of the Monticello PRA model have been addressed and incorporated into the current model utilized for the EPU risk assessment, with the following exceptions.

An open item related to Human Reliability Analysis (HRA) element in NEI 00-02 recommended that a sensitivity study be re-performed to identify any changes to the list of key pre-initiator operator actions identified in the Individual Plant Examination (IPE). If any were found, it was recommended that the HRA analysis be re-performed using more rigorous HRA approach, to reduce conservatism. The EPU implementation will have no impact on pre-initiator Human Error Probability (HEP) values; therefore, even if values were modified for some pre-initiator HEPs, these values would apply unchanged to both the pre-EPU and post-EPU risk quantification, and so this open item has no potential impact on the conclusions of the EPU assessment.

An open item recommended verifying data used to generate some initiating event frequencies accounting for plant unavailability. It is recognized that the elimination of non-operational time may result in moderate increases in calculated initiating event frequencies. Like the above item, any changes in initiating event frequencies to reflect unavailability time, would apply equally to pre and post-EPU quantification, and thus would have no impact on the conclusions of the EPU assessment.

An open item recommended considering performance of Bayesian updating for some additional events. Again, if this data enhancement was performed, it would apply equally to pre and post-EPU quantifications. No impact on the conclusion of the EPU assessment would result.

Several recommendations were made to improve model documentation, conduct sensitivity studies and perform uncertainty analysis to meet enhanced capabilities set forth in the ASME standard. These enhancements were intentionally deferred to be accomplished in preparation for Monticello's upcoming formal Regulatory Guide 1.200 Peer Review, and would not result in any significant impact on the results of the EPU risk assessment.

In conclusion, all open items from the ARE self-assessment have been incorporated into the PRA model or have been determined to have no significant impact on the EPU risk assessment.

NRC RAI No. 3

EPU SAR, Section 4.1.1 and Section 4.5 states no significant impact on internal flooding initiator frequencies are postulated due to the EPU. Since higher flow rates can contribute to changes in initiator events for floods, please provide a more thorough justification for your conclusion on how EPU flow rates will not affect internal flooding initiator frequencies.

NSPM RESPONSE

The piping systems that will experience significantly higher flow rates for the EPU are Main Steam, Feedwater and Condensate. NSPM has evaluated the effect of the proposed EPU on the Flow Accelerated Corrosion (FAC) analysis for the plant. The increased MS and FW flow rates at EPU conditions do not significantly affect the potential for FAC in these systems. The increase in flow rates due to EPU is not expected to increase failure rates in plant piping systems due to FAC. See Section 2.1.6 of the PUSAR for additional information (Enclosure 5 to LMT-08-052).

One may conservatively postulate an increase in piping failure rate due to increased flow due to the EPU, but any such minor perturbation in the piping failure rate is not a key decision making factor in the risk impact of the MNGP EPU. As a sensitivity case, the MNGP EPU risk assessment conservatively postulated a doubling of the Large LOCA initiating event frequency) and the ΔCDF and ΔLERF results remained within RG 1.174 Region III (very small change in risk). Refer to Section 5.7.1 of the MNGP EPU risk assessment. In response to this RAI, that sensitivity study was revised to also conservatively assume a doubling of the HELB frequencies for MS and FW and the quantitative results were non-significantly impacted (ΔCDF and ΔLERF changes in the 2nd decimal place).

NRC RAI No. 4

The NRC staff notes that success criteria changes crediting Control Rod Drive Hydraulic (CRDH) by depressurizing the reactor is unique compared to boiling water reactor (BWR) EPUs previously approved. If depressurization is successful, then the low pressure injection sources would be available, obviating any need for successful CRDH injection. Further, if changes to emergency operating procedures (EOP) are required to implement this change, then this change could potentially complicate operator response to other events. The NRC staff requests responses to the following issues to better understand the impacts of this change. If new operator actions are required, then questions d, e, and f apply.

a. Was depressurization required in order to credit two CRDH pumps pre-EPU? Describe the analyses conducted which determined that depressurization was necessary for EPU conditions.

b. Describe how the action for depressurization has been modeled in the PRA. Explain how the human reliability analysis (HRA) for this action compares to any other actions to depressurize the reactor. Describe how the HRA dependency of this action on other operator actions was assessed.

c. Does the new requirement for reactor depressurization to allow CRDH injection create new sequences and end states? Provide the basis for this conclusion and a summary of any resulting changes to the sequences and end states.

d. Describe the risk significance of the CRDH success criteria change (i.e., Fussell Vesely Importance and Risk Achievement Worth) and contribution of the sequences associated with this change to core damage frequency (CDF) and large early release frequency (LERF).

e. Did changes to CRDH success criteria require changes to EOPs? If yes, describe the changes, operator training and validation methods, and why the changes to plant-specific EOPs remain consistent with BWR EOP guidelines or were otherwise determined to be acceptable. Your response should also discuss any potential negative impacts from the operator inappropriately depressurizing the reactor due to the new procedures.

f. Was a focused peer review performed for the PRA model changes necessary to incorporate the new CRDH success criteria including any new event tree structure and operator actions? If yes, provide the results of that peer review. If not, please justify why a peer review was not judged to be required.

NSPM RESPONSE

No EOP changes and no special or new requirements for operator actions pertain to this PRA success criterion adjustment for the EPU.

In light of this question, MNGP is providing the following clarification of the basis of the CRDH credit for EPU.

An excerpt from Section 4.1.2.8 of Enclosure 15 of L-MT-08-052 is included below.

> CRDH as the only early injection source using 2 CRDH pumps at nominal flow now requires that the RPV be depressurized.

MNGP recognizes that the use of the words *requires* and *nominal* in the context of the EPU PRA above may need further explanation. A revised wording and an explanation are below.

> CRDH as the only early injection source using 2 CRDH pumps at nominal flow is not successful at EPU. If CRDH using 2 pumps is the only high pressure injection source available, RPV level will continue to drop and the EOPs direct initiation of RPV emergency depressurization. If RPV emergency depressurization is successfully initiated, then 2 CRDH pumps alone will be successful to maintain adequate core cooling; if RPV emergency depressurization is not initiated, then RPV level will continue to drop unless another injection source is aligned.

The CRDH credit for early high pressure injection is determined by the thermal-hydraulic conditions that exist for the event sequence. The success criteria are based on the flow delivery requirements and reactor conditions present at the time that the conditions are evaluated. One CRDH pump is running at the outset of the event. Initiation of the second pump to increase nominal flow requires manual operator action. Consequently, credit for two CRDH pumps as an early injection source is a function of the timing for starting a second pump, the flow developed from these pumps, and the EPU reactor conditions. In this case, flow is defined as the nominal flow that would be developed from two CRDH pumps operating in parallel with the reactor at pressure.

Under CLTP conditions, the model shows that reactor conditions are such that credit can be taken for two CRDH pumps as the only early injection source at nominal flow regardless of reactor pressure. Under EPU conditions, the changes in the reactor conditions are such that nominal flow from two CRDH pumps is insufficient to meet the success criteria as an early injection source for the cases where the RPV remains at pressure. See section 4.1.2.2 of Enclosure 15 to L-MT-08-052, "Identification of Risk Implications Due to Extended Power Uprate at Monticello."

Once the RPV is depressurized due to RPV low level conditions *in accordance with the existing EOP depressurization actions,* the reactor pressure decrease results in a CRDH flow increase such that the EPU success criteria are met. Depending on the assumed failures, CRDH may or may not be the only injection source. This scenario does not involve any changes to the EOPs, nor in any way, does it involve a special or new requirement for the operators to manually depressurize the reactor to specifically enable or increase CRDH flow above nominal.

Specific summary responses are provided below to each of the RAI items.

a. As discussed above, this success criterion adjustment for the MNGP EPU PRA does not pertain to any EOP changes or new operator action requirements in the plant. Two CRDH pumps at nominal flow are adequate as an early injection source in the pre-EPU PRA regardless of reactor pressure. Given the increased power level of the EPU, two CRDH pumps at nominal flow is not adequate as an early injection source for the EPU condition at high reactor pressure, but for low reactor pressure scenarios this success criterion is adequate.

   MAAP thermal hydraulic runs using the MNGP plant-specific MAAP model were performed to investigate success criteria changes for the MNGP EPU risk assessment. MAAP runs MNGPEPU5a through MNGPEPU5ix investigate the CRDH success criteria for early injection (see Appendix E of the MNGP EPU risk assessment).

b. As discussed above, this success criterion adjustment for the MNGP EPU PRA does not pertain to any EOP changes or new operator action requirements in the plant. No new operator actions were added to the PRA or revised for this success criterion adjustment.

   The MNGP PRA, like all BWR PRAs, includes operator actions in the accident sequence models for RPV emergency depressurization due to RPV low level conditions in accordance with the existing EOPs. No new RPV emergency depressurization action was added to the PRA model.

c. As discussed above, this success criterion adjustment for the MNGP EPU PRA does not pertain to any EOP changes or new operator action requirements in the plant. This success criterion adjustment for the MNGP EPU PRA does not change any accident sequence structures or accident end states.

d. As discussed above, this success criterion adjustment for the MNGP EPU PRA does not pertain to any EOP changes or new operator action requirements in the plant. As such, this item is not applicable as defined by the RAI.

e. As discussed above, this success criterion adjustment for the MNGP EPU PRA does not pertain to any EOP changes or new operator action requirements in the plant. As such, this item is not applicable as defined by the RAI.

f. As discussed above, this success criterion adjustment for the MNGP EPU PRA does not pertain to any EOP changes or new operator action requirements in the plant. As such, this item is not applicable as defined by the RAI.

NRC RAI No. 5

Describe the calculation for the change in risk as a result of needing one additional safety/relief valve (SRV) to open for anticipated transient without scram (ATWS). Does the change in risk calculated for EPU include a contribution due to the SRV success criteria? Discuss how the change in SRV success criteria has been incorporated into the PRA model and include a discussion of the changes to common cause failure events and their basis.

NSPM RESPONSE

The change in risk calculated for the MNGP EPU does include the change in the SRV success criterion for the RPV overpressure protection function (refer to Sections 4.1.2.5 and 5.1 of the MNGP EPU risk assessment). Further discussion is provided below by comparing the modeling of this success criterion in the MNGP CLTP PRA for ATWS scenarios and the changes made to reflect the EPU.

- MNGP CLTP PRA: The modeling in the MNGP CLTP PRA of SRVs for RPV overpressure protection during ATWS scenarios is summarized as follows.
  - Success criterion: Six of eight SRVs must open to ensure adequate RPV overpressure protection (i.e., failure of 3 SRVs to open will fail this function in the PRA). The PRA models in the system fault tree structures both the random failures of SRVs and the common cause failure of SRVs for this success criterion.
  - Random Failure: The system fault tree structure for failure of RPV overpressure protection during ATWS scenarios includes a 3 of 8 "K/N" Boolean logic gate that models the probability that three SRVs randomly fail to open in response to the ATWS RPV pressure challenge. Eight random "failure to open" basic events (one for each of the eight MNGP SRVs) are input into this logic gate. The random failure probability of an SRV to open is 1.2E-4/demand per SRV.
  - CCF Failure: In addition to the random SRV failure logic described above, the system fault tree structure for failure of RPV overpressure protection during ATWS scenarios includes a basic event for common cause failure (CCF) to open 3 of 8 SRVs. This single CCF basic event covers all combinations of three out of a group of eight. The CCF methodology used is the Alpha methodology. The probability of this CCF event (2.0E-6) is calculated as λ x $\text{Alpha}_{3/8}$ (i.e., 1.2E-4 x 1.7E-2 = 2.0E-6). Lambda (λ) is the random failure rate (1.2E-4/demand) and $\text{Alpha}_{3/8}$ is the CCF parameter for 3 failures in a Common Cause Component Group (CCCG) size of 8 SRVs (1.7E-2). This is a standard industry approach for calculating the probability for a CCF basic event modeling k failures in a group of n components (refer to Appendix A of NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in PRA). The $\text{Alpha}_{3/8}$ value of 1.7E-2 is from the latest Idaho National Laboratory (INL) CCF parameter database and is based on

industry experience from 1980 through 2001 at the time of development of the MNGP 2005 PRA.

- MNGP EPU PRA: PRA changes to reflect the EPU condition are summarized as follows:

  - Success criterion: Based on MNGP EPU Task Report for ATWS, 7 of 8 SRVs are assumed required to open to ensure adequate RPV overpressure protection for EPU (i.e., failure of 2 SRVs to open will fail this function in the PRA).

  - Random Failure: The 3 of 8 "K/N" Boolean logic gate discussed above was revised to be a 2 of 8 "K/N" gate. The random failure probability of an SRV to open was not changed (the EPU does not impact the SRV failure rate).

  - CCF Failure: The probability of the SRV CCF event discussed above was revised to 1.2E-5 (i.e., 1.2E-04 x 0.1 = 1.2E-5). The selection of 0.1 as the CCF factor for this event was conservative, but has a non-significant impact on the MNGP EPU risk assessment results. The INL CCF database provides an $Alpha_{2/8}$ value of 3.5E-2 for 2 failures in a CCCG size of 8 SRVs. Use of either a CCF factor of 0.1 or 3.5E-2 has no significant impact the delta CDF or delta LERF estimates calculated in the MNGP EPU risk assessment (the risk results would not change in the third decimal place).

NRC RAI No. 6

EPU SAR Enclosure 15, Section 5.6 states that the EPU change in power represents a relatively small change to the overall challenge to the containment under severe accident conditions. Please provide additional details which justify this conclusion.

NSPM RESPONSE

Section 5.6 of the MNGP EPU risk assessment is a summary of the Level 2 PRA impacts due to EPU. Section 4.7 of the MNGP EPU risk assessment provides more discussion. The issues related to EPU impacts on containment challenges under severe accident conditions (i.e., post core damage) are summarized below.

- Containment Isolation: Containment isolation is demanded early in an accident scenario before extreme containment conditions manifest. The EPU has no impact on the failure probabilities of containment isolation signals or containment isolation valves.

- Quasi-Static Pressure/Temperature Loading: Primary containment integrity is challenged as the containment pressurizes and temperatures increase. Containment failure can occur in a variety of locations and due to different mechanisms (e.g., high temperature seal failure, structural failure, penetration failure, drywell head lift, etc.). The increased decay heat load of the EPU has no impact on these containment loading profiles, the EPU only impacts the time required to reach the loading challenges. MAAP runs performed for the EPU show that the time to reach the primary containment ultimate failure point (as assessed in the MNGP PRA) is over 40 hrs for both the CLTP and EPU conditions. Changes in such lengthy timings have a negligible impact on human error rates and thus a negligible impact on the calculated risk profile. These challenges apply to loss of containment heat removal scenarios prior to the onset of core damage as well as post-core damage scenarios in which the core debris is maintained in-vessel.

- Containment Dynamic Loading: These challenges include un-mitigated ATWS, LOCA loads and energetic phenomena post core damage (see bullet below). Un-mitigated (inadequate level/power control, SLC failure) ATWS scenarios are modeled in the PRA as leading directly to a containment failure, this is a standard PRA modeling approach and is not changed due to the EPU. EPU LOCA dynamic loads on the containment have been calculated to be within safety and design limits.

- Energetic Phenomena: A variety of severe challenges to the primary containment post core damage have been identified in the MNGP PRA and in industry studies and guidelines. These energetic phenomena may manifest at the time of the onset of core damage, the time of core slump into the lower RPV head, the time of RPV melt-through, or after core debris falls to the drywell floor and migrates. These energetic phenomena include (among others): in-vessel steam explosions, hydrogen deflagration, ex-vessel steam explosions, direct containment heating, core-concrete interaction, and drywell shell melt-through. The likelihood of each of these phenomena, and the required conditions, are based on industry generic studies and are not influenced by initial reactor power level. This is a standard PRA industry practice.

NRC RAI No. 7

EPU SAR Enclosure 15, Section 4.3.1 states "fire PRA results are less impacted by changes in operator actions timings than the internal events PRA results." The re-rate safety evaluation dated September 16, 1998, Section 5.3, states: "The CDF contribution from internal fires increased from 8.34E-6/Year to 8.8E-6/Year. This was attributed solely to the increase in human error rates because the time available to perform various accident mitigating tasks decreases with uprate." Please provide additional justification for the conclusion stating fire PRA results are less impacted by changes in operator actions timings than the internal events PRA results. Please explain the inconsistency between the re-rate fire CDF increase and the EPU fire CDF increase.

NSPM RESPONSE

The re-rate study was conducted in the late 1990's and assessed the risk impact of a 12% power increase from 1670 MWt to 1880 MWt. The actual license submittal requested a 6.3% power increase to 1775 MWt, which is the current operating license limit for MNGP. The level I internal events assessment applied to the re-rate study estimated that Core Damage Frequency (CDF) increased 17.5% from a baseline value of 1.37 E-05/yr to 1.61 E-05/yr. A further assessment of internal fires based on a conservative fire analysis model, estimated that fire risk increased 5.5% from a baseline value of 8.34 E-06/yr to 8.80 E-06/yr. It is noted that the calculated increase due to fire (5.5%) is approximately 1/3rd of the calculated internal events risk increase (17.5%) for the re-rate assessment.

The recent EPU risk assessment, utilizing an *updated* PRA model, was performed to assess the impact of a 13% power increase from the current 1775 MWT to 2004 MWT. This level I internal events assessment applied to the EPU estimated that CDF increased 7.8% from a baseline value of 7.32 E-06/yr to 7.89 E-06/yr. For lack of a current fire PRA model compatible with the existing Level I internal events model, the CDF risk increase due to fires is estimated to be, consistent with the conclusion of the re-rate assessment; approximately 1/3rd of the internal events increase, or 2 to 3% (1/3 of 7.8%).

In each assessment, re-rate and EPU, the major contribution to change in CDF is dominated by a reduction in available reaction time and the corresponding increase in human error probabilities for both the internal events and fire assessments. It is noted that fire risk is impacted less (a factor of approximately 1/3) by changes in human error probabilities brought about by the increased power level, than the impact on internal events risk. Although simplified, when a dominant and common mode failure mechanism (fire) is added to the PRA model, failure mechanisms that exist without fire tend to be less significant in driving results.

NRC RAI No. 8

The NRC staff requests responses for the following information on Low Power and Shutdown PRA

- Explain how the EPU affects the scheduling of outage activities.
- Provide additional information regarding the reliability and availability of equipment used for shutdown conditions.
- Explain how the EPU affects the availability of equipment or instrumentation used for contingency plans.
- Explain how the EPU affects the operation to close containment during loss of shutdown cooling.

NSPM RESPONSE

Scheduling of Outage Activities

A review of the associated procedures for the scheduling of outage activities was performed including the procedure for managing outage risk. There is no significant difference between CLTP and EPU for these activities.

Reliability and Availability of Equipment

The EPU does not have a significant effect on the reliability or availability of equipment used for shutdown conditions or for contingency plans.

Operation to Close Containment During Loss of Shutdown Cooling

The increased decay heat levels resulting from operation at EPU power would, ignoring other factors, decrease the time to reach boiling conditions and subsequently the time to uncover the core following a loss of decay heat removal (DHR). Assuming the primary containment is open for maintenance, there would be correspondingly less time to establish primary containment following a loss of DHR prior to reaching saturation temperature, at which time Technical Specifications requires primary containment to be operable and intact.

The time required and the physical actions to be completed to re-establish primary containment are unaffected by the EPU. These variables (time and required actions) are highly dependent on the existing conditions that are present at the time a loss of shutdown cooling occurs. In fact there are conditions where it would be counterproductive to re-establish primary containment upon a loss of shutdown cooling event. For example, during refueling operations, the reactor water level is flooded well above the level of the reactor vessel, to be equal to the level of the fuel pool. This results in a substantial increase in the amount of time available to recover DHR prior to reaching saturation temperatures. Lowering water level in an effort to re-establish primary containment under these conditions would not be consistent with sound reactor safety practices. These flooded up conditions may exist for a majority of the time during refueling outages. Other variables that have a pronounced effect on the time to boil following loss of DHR include status of the fuel pool gates, location of spent fuel, and initial reactor water/fuel pool water temperature.

Overall, outage risk, including boiling frequency and CDF, is and will continue to be managed by using existing procedures to rigorously balance the effects of equipment availability, water inventory, water temperature as well as decay heat levels such that there will be little or no impact from EPU on risk during shutdown conditions. When decay heat levels are high, risk will be managed by limiting the equipment removed from service and controlling water level and temperature.

NRC RAI No. 9

EPU SAR Enclosure 15, Section 3.3.2, indicates that the changes to EOPs and severe accident management guides as a result of the EPU were not available prior to completion of the PRA evaluation and it was assumed that the procedural changes would have a minor impact on the PRA results. The NRC staff needs to conclude that EOP impacts are minimal. Therefore, please provide a schedule for the development of the final draft of EOPs, and confirm that the PRA results would only be minimally impacted.

NSPM RESPONSE

The EOP and SAMG impacts due to EPU are minimal. All EOP and SAMG impacts have been identified, and the changes are limited to figures. There are no changes to EOP or SAMG actions due to EPU, and the PRA results are only minimally impacted.

The EOP changes will be completed on a schedule that supports completion of all required training prior to EPU implementation. Implementation of the first phase of uprate is planned to be completed within 120 days from NRC approval of the EPU LAR.

NRC RAI No. 10

EPU SAR Enclosure 15 provides a quantitative assessment of the risk impact of the COP credit for low pressure ECCS pump NSPH for ATWS, SBO, and internal fires events. The analysis indicates a CDF for each of these events, but does not provide an LERF metric. Please provide an LERF metric for each of the aforementioned events.

NSPM RESPONSE

The estimated delta LERF (ΔLERF) values for the aforementioned events for the COP sensitivity study are as follows:

Scenario Type	ΔLERF (1/yr)	Comment
ATWS	4.5E-12	This estimate conservatively assumes that ΔLERF is equal to ΔCDF.
Station Blackout (SBO)	5.9E-9	The SBO scenario for the COP sensitivity is a long-term station blackout (ECCS injection assumed to fail at t=4 hrs when AC power recovered and ECCS pumps aligned to pool fail due to inadequate NPSH). Per MAAP thermal hydraulic runs performed for the EPU, the operators have approximately another 2.5 hrs until the onset of core damage. The SBO COP sensitivity conservatively assumed that the operators did not respond after t=4 hrs. Recognizing the EOP direction for alternative RPV injection (e.g., RHRSW crosstie via LPCI), and conservatively assuming ΔLERF is equal to ΔCDF, the 1.7E-7/yr conservative sensitivity estimate in the MNGP EPU Submittal is revised here to 5.9E-9 (based on application of a 3.5E-2 nominal failure probability estimate for alignment of alternate injection external from the containment – the long-term HEP for the alignment is ~4E-3 and the equipment failure rate contribution is approximately ~3E-2).
Internal Fires	5.1E-9	This estimate conservatively assumes that ΔLERF is equal to ΔCDF.