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Identification of Risk Implications Due to Extended Power Uprate at Monticello

IDENTIFICATION OF RISK IMPLICATIONS DUE TO EXTENDED POWER UPRATE AT MONTICELLO

Prepared for
Nuclear Management Company
(NMC)

Prepared by:

ERIN Engineering and Research, Inc.
an SKF Group Company

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EXECUTIVE SUMMARY

The Extended Power Uprate (EPU) project for Monticello has been reviewed to determine the net impact on the Monticello risk profile.

The existing Monticello Probabilistic Risk Assessment (PRA) is based on the current licensed thermal power (CLTP) level of 1775 MWt. Monticello is currently pursuing a 13% increase (i.e., Extended Power Uprate) of the CLTP to 2004 MWt.

The enclosed assessment of the power uprate impacts on risk has been performed relative to the current PRA. The guidelines from the NRC (Regulatory Guide 1.174) are followed to assess the change in risk as characterized by core damage frequency (CDF) and Large Early Release Frequency (LERF) and to determine if the change in risk is anything but very low.

The methodology consists of an examination of the important elements of the Monticello Probabilistic Risk Assessment (PRA) to assess the impact of the following EPU changes on the PRA elements:

- Hardware changes
- Procedural changes
- Set point changes
- Power level change

These changes are interpreted in terms of their PRA model effects, which can then be used to assess whether there are any resulting risk profile changes.

The scope of this report includes the complete risk contribution associated with the Extended Power Uprate at Monticello. Risk impacts due to internal events are assessed using the MNGP Level 1 and Level 2 PRA Model of Record (2005 Monticello PRA ***average maintenance*** model, fault tree ***Risk-T&M.caf***). External events are evaluated

using the analyses of the Monticello Individual Plant Examination of External Events (IPEEE) Submittal. [10] The impacts on shutdown risk contributions are evaluated on a qualitative basis.

All commitments resulting from the MNGP IPE and IPEEE programs have been resolved.

The results of the PRA evaluation are the following:

- Detailed thermal hydraulic analyses of the plant response using the EPU configuration indicate slight reductions in the operator action "allowable" times for some actions.
- The reduced operator action "allowable" times resulted in minor increases in the assessed Human Error Probabilities (HEPs) in the PRA model, specifically in RPV water level control errors during failure to scram sequences.
- Only very small risk increases were identified for the changes associated with the EPU, those associated with: (1) slightly reduced times available for effective operator actions; and (2) minor changes in some functional success criteria in the PRA.
- The risk impact due to the implementation of the Extended Power Uprate is low and acceptable. The risk impact is in the "very small" category (i.e., Region III of the Regulatory Guide 1.174 Guidelines) for CDF and for LERF.

The EPU is estimated to increase the Monticello internal events PRA CDF from the base value of 7.32E-6/yr to 7.89E-6/yr, an increase of 5.67E-7 (7.8%). LERF increases from the base value of 3.64E-7/yr to 3.94E-7/yr, an increase of 3.00E-8/yr (8.2%).

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Section 1
INTRODUCTION

Monticello is currently pursuing an increase in reactor power from the current licensed thermal power of 1775 MWth to 2004 MWth, an Extended Power Uprate (EPU) of 113% CLTP. The purpose of this report is to:

(1) Identify any significant change in risk associated with the Extended Power Uprate (EPU) as measured by the Monticello PRA models;

(2) Provide the basis for the impacts on the risk model associated with EPU

1.1 BACKGROUND

The Monticello PRA is a state-of-the-technology tool developed consistent with current PRA methods and approaches. The MNGP model is developed and quantified using the CAFTA (part of the EPRI R&R Workstation) software.

The Monticello PRA is based on realistic assessments of system capability over the 24 hour mission time of the PRA analysis. Therefore, PRA success criteria may be different than the design basis assumptions used for licensing Monticello. This report examines the risk profile changes from this realistic perspective to identify changes in the risk profile on a best estimate basis that may result from postulated accidents, including severe accidents.

1.2 PRA QUALITY

The quality of the MNGP PRA models used in performing the risk assessment for the MNGP EPU is manifested by the following:

- Sufficient scope and level of detail in PRA
- Active maintenance of the PRA models and inputs

- Comprehensive Critical Reviews

Scope and Level of Detail

The MNGP PRA is of sufficient quality and scope for this application. The MNGP PRA modeling is highly detailed, including a wide variety of initiating events (e.g., transients, internal floods, LOCAs inside and outside containment, support system failure initiators), modeled systems, extensive level of detail, operator actions, and common cause events.

Maintenance of Model, Inputs, Documentation

The MNGP PRA model and documentation has been updated to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data. The current MNGP PRA model at the time of this analysis is 2005 Monticello PRA ***average maintenance*** model (fault tree ***Risk-T&M.caf***). The Level 1 and Level 2 MNGP PRA analyses were originally developed and submitted to the NRC in February 1992 as the Monticello Individual Plant Examination (IPE) Submittal. The MNGP PRA submittal and the subsequent NRC approval are described in Section 14.1 of the MNGP USAR.

Critical Reviews

The Monticello internal events received a formal industry PRA Peer Review in October 1997. All of the "A" and "B" priority comments from the 1997 peer review have been addressed by MNGP and incorporated into the current MNGP PRA model as appropriate.

Summary

In summary, it is found that the Monticello Level 1 and Level 2 PRAs provide the necessary and sufficient scope and level of detail to allow the calculation of CDF and LERF changes due to the Extended Power Uprate (EPU). Refer to Appendix C for further details regarding the quality of the MNGP PRA.

1.3 PRA DEFINITIONS AND ACRONYMS

Definitions

The following PRA terms are used in this study:

CDF – Core Damage Frequency (CDF) is a risk measure for calculating the frequency of a severe core damage event at a nuclear facility. Core damage is the end state of the Level 1 PRA. A core damage event may be defined in the MNGP PRA by one or more of the following:

- Maximum core temperature greater than 2200 degrees Fahrenheit,
- RPV water level at 1/3 core height and decreasing,
- Containment failure induced loss of injection.

CDF is calculated in units of events per year.

With respect to analyzing MAAP thermal hydraulic runs, very short spikes (e.g., seconds or a couple minutes) above 2200F are not automatically declared core damage. The case is typically re-run and re-analyzed carefully.

LERF – Large Early Release Frequency (LERF) is a risk measure for calculating the frequency of an offsite radionuclide release that is HIGH in fission product magnitude and EARLY in release timing. A HIGH magnitude release is defined as a radionuclide release of sufficient magnitude to have the potential to cause early fatalities (e.g., greater than 10% Cesium Iodide contribution to release). An EARLY timing release is defined as the time prior to that where minimal offsite protective measures have been implemented (e.g., less than 6 hours from accident initiation). LERF is calculated in units of events per year.

Initiating Event – Any event that causes/requires a scram/manual shutdown (e.g., Turbine Trip, MSIV Closure) and requires the initiation of mitigation systems to reach a safe and stable state. An initiating event is modeled in the PRA to represent the primary transient event that can lead to a core damage event given failure of adequate mitigation systems (i.e., adequate with respect to the transient in question).

Internal Events – Those initiating events caused by failures internal to the system boundaries. Examples include Turbine Trip, MSIV Closure, Loss of an AC Bus, Loss of Offsite Power, and internal floods.

External Events – Those initiating events caused by failures external to the system boundaries. Examples include fires, seismic events, and tornadoes.

HEP – Human Error Probability (HEP) is the probabilistic estimate that the operating crew fails to perform a specific action (either properly or within the necessary time frame) to support accident mitigation. The HEP is calculated using industry methodologies and considers a number of performance shaping factors such as:

- training of the operating crew,
- availability of adequate procedures,
- time required to perform action
- time available to perform action
- stress level while performing action

HRA – Human Reliability Analysis (HRA) is the systematic process used to evaluate operator actions and quantify human error probabilities.

MAAP – The Modular Accident Analysis Package (MAAP) is an industry recognized thermal hydraulic code used to evaluate design basis and beyond design basis accidents. MAAP can be used to evaluate thermal hydraulic profiles within the primary system (e.g., RPV pressure, boildown timing) prior to core damage. MAAP also can be used to evaluate post core damage phenomena such as RPV breach, containment mitigation, and offsite radionuclide release magnitude and timing.

Level 1 PRA – The Level 1 PRA is the evaluation of accident scenarios that begin with an initiating event and progress to core damage. Core damage is the end state for the Level 1 PRA. The Level 1 PRA focuses on the capability of plant systems to mitigate a core damage event.

Level 2 PRA – The Level 2 PRA is a continuation of the Level 1 PRA evaluation. The Level 2 PRA begins with the accident scenarios that have progressed to core damage and evaluates the potential for offsite radionuclide

releases. Offsite radionuclide release is the end state for the Level 2 PRA. The Level 2 PRA focuses on the capability of plant systems (including containment structures) to prevent a core damage event to result in an offsite release.

RAW – The Risk Achievement Worth (RAW) is the calculated increase in a risk measure (e.g., CDF or LERF) given that a specific system, component, operator action, etc. is assumed to fail (i.e., failure probability of 1.0). RAW is presented as a ratio of the risk measure given the component is failed divided by the risk measure given the component is assigned its base failure probability.

FV – The Fussell-Vesely (FV) importance is a measure of the contribution of a specific system, component, operator action, etc. to the overall risk. F-V is presented as the percentage of the overall risk to which the component failure contributes. In other words, the F-V importance represents the overall decrease in risk if the component is guaranteed to successfully operate as designed (i.e., failure probability of 0.0).

Acronyms

The following acronyms are used in this study:

AC	Alternating Current
ACRS	Advisory Committee on Reactor Safeguards
ARI	Alternate Rod Insertion
ATWS	Anticipated Transient Without Scram
BIIT	Boron Injection Initiation Temperature
BOC	Break Outside Containment
BOP	Balance of Plant
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CLTP	Current Licensed Thermal Power
CPPU	Constant Pressure Power Uprate
CRDH	Control Rod Drive Hydraulics
CS	Core Spray
CST	Condensate Storage Tank
CSW	Condensate Service Water
CTS	Condensate Transfer System
DBA	Design Basis Accident

DC	Direct Current
DFP	Diesel Driven Fire Pump
DHR	Decay Heat Removal
DW	Drywell
ECCS	Emergency Core Cooling System
ED	Emergency Depressurization
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
FIVE	Fire-Induced Vulnerability Evaluation
FV	Fussell-Vesely (risk importance measure)
FW	Feedwater
FWLC	Feedwater Level Control
GE	General Electric
HCLPF	High Confidence Low Probability of Failure
HCTL	Heat Capacity Temperature Limit
HEP	Human Error Probability
HP	High Pressure
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
I&C	Instrumentation and Control
IORV	Inadvertently Opened Relief Valve
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Evaluation of External Events
ISLOCA	Interfacing Systems LOCA
LERF	Large Early Release Frequency
LLOCA	Large LOCA
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
MAAP	Modular Accident Analysis Program
MLOCA	Medium LOCA
MSCWLL	Minimum Steam Cooling Water Level Limit
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MWt	Megawatt (thermal)
NEI	Nuclear Energy Institute
NMC	Nuclear Management Company
NPSH	Net Positive Suction Head

NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OLTP	Original Licensed Thermal Power
OOS	Out Of Service
PCPL	Primary Containment Pressure Limit
PRA	Probabilistic Risk Assessment (alternative term for PSA)
PSA	Probabilistic Safety Assessment (alternative term for PRA)
PSSA	Probabilistic Shutdown Safety Assessment
RAW	Risk Achievement Worth (risk importance measure)
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	RHR Service Water
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Clean-Up
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SDC	Shutdown Cooling
SLOCA	Small LOCA
SMA	Seismic Margins Analysis
SORV	Stuck Open Relief Valve
SRV	Safety Relief Valve
SSC	Systems, Structures, and Components
SV	Safety Valve
TAF	Top of Active Fuel
VB	Vacuum Breaker
MNGP	Monticello Nuclear Generating Plant
WW	Wetwell

1.4 GENERAL ASSUMPTIONS

The Extended Power Uprate (EPU) risk evaluation includes a limited number of general assumptions as follows:

- The plant and procedural changes identified by NMC are assumed to reflect the as-built, as-operated plant after the Extended Power Uprate is fully implemented. The information provided by NMC (as well as the

MNGP EPU GE Task Reports) is used as input to the current Monticello PRA model to evaluate the risk impact of the power uprate. MNGP will uprate to the full EPU power level in two steps over the next few years. The risk analysis documented in this report is performed for a one step increase to the full EPU power level; this analysis bounds the MNGP two step uprate process.

- This analysis is based on all the inputs provided by NMC in support of this assessment. For systems where no hardware or procedural changes have been identified, the risk evaluation is performed assuming no impact as a result of the EPU.
- Replacement of components with enhanced like components does not result in any supportable significant increase in the long-term failure probability for the components.
- The PRA success criteria are different than the success criteria used for design basis accident evaluations. The PRA success criteria assume that systems that can realistically perform a mitigation function (e.g., main condenser or containment venting for decay heat removal) are credited in the PRA model. In addition, the PRA success criteria are based on the availability of a discrete number of systems or trains (e.g., number of pumps for RPV makeup).

Section 2
SCOPE

The scope of this risk assessment for the Extended Power Uprate at Monticello addresses the following plant risk contributors:

- Level 1 Internal Events At-Power (CDF)
- Level 2 Internal Events At-Power (LERF)
- External Events At-Power
  - Seismic Events
  - Internal Fires
  - Other External Events
- Shutdown Assessment

Risk impacts due to internal events are assessed using the MNGP 2005 Monticello PRA ***average maintenance*** model (fault tree ***Risk-T&M.caf***). Level 2 sequences resulting in the Large-Early release category comprise the LERF risk measure. External events are evaluated using the analyses of the Monticello Individual Plant Examination of External Events (IPEEE) Submittal. [10] The impacts on shutdown risk contributions are evaluated on a qualitative basis.

All commitments resulting from the MNGP IPE and IPEEE Programs have been resolved.

As discussed in Section 3, all PRA elements are reviewed to ensure that identified EPU plant, procedural, or training changes that could affect the risk profile are addressed. The information input to this process consisted of preliminary design, procedural, and training information provided by NMC. The final design, analytical calculations, and procedural changes had not been completed prior to this risk assessment.

Section 3
METHODOLOGY

This section of the report addresses the following:

- Analysis approach used in this risk assessment (Section 3.1)
- Identification of principal elements of the risk assessment that may be affected by the Extended Power Uprate and associated plant changes (Section 3.2)
- Plant changes used as input to the risk evaluation process (Section 3.3)
- Scoping assessment (Section 3.4)

3.1 ANALYSIS APPROACH

The approach used to examine risk profile changes is described in the following subsections.

3.1.1 Identify PRA Elements

This task is to identify the key PRA elements to be assessed as part of this analysis for potential impacts associated with plant changes. The identification of the PRA elements uses the NEI PRA Peer Review Guidelines.[4] Section 3.2 summarizes the PRA elements assessed for the Monticello EPU.

3.1.2 Gather Input

The input required for this assessment is the identification of any plant hardware modifications, procedural or operational changes that are to be considered part of the

Extended Power Uprate. This includes changes such as instrument setpoint changes, added equipment, and procedural modifications.

3.1.3 Scoping Evaluation

This task is to perform a scoping evaluation by reviewing the plant input against the key PRA elements. The purpose is to identify those items that require further quantitative analysis and to screen out those items that are judged to have negligible or no impact on plant risk as modeled by the MNGP PRA.

3.1.4 Qualitative Results

The result of this task is a summary which dispositions all the risk assessment elements regarding the effects of the Extended Power Uprate. The disposition consists of three Qualitative Disposition Categories:

Category A: Potential PRA change due to power uprate. PRA modification desirable or necessary

Category B: Minor perturbation, negligible impact on PRA, no PRA changes required

Category C: No change

A short explanation providing the basis for the disposition is provided in Section 4.

3.1.5 Implement and Quantify Required PRA Changes

This task is to identify the specific PRA model changes required to address the EPU, implement them, and quantify the model. The MNGP PRA elements were investigated with the aid of additional deterministic calculations performed in support of this analysis (see Appendix E). Section 4.1 summarizes the review of PRA analysis impacts

associated with the increased power level. These effects and other effects related to plant or procedural changes are identified and documented in Section 4.

3.2 PRA ELEMENTS ASSESSED

The PRA elements to be evaluated and assessed can be derived from a number of sources. The NEI PRA Peer Review Guidelines [4] provide a convenient division into "elements" to be examined.

Each of the major risk assessment elements is examined in this evaluation. Most of the risk assessment elements are anticipated to be unaffected by the Extended Power Uprate. The risk assessment elements addressed in this evaluation for impact due to the EPU (refer to Section 4 for impact evaluation) include the following:

- Initiating Events
- Systemic/Functional Success Criteria, e.g.:
  - RPV Inventory Makeup
  - Heat Load to the Suppression Pool
  - Time to Boildown
  - Blowdown Loads
  - RPV Overpressure Margin
  - SRV Actuations
  - SRV Capacity for ATWS
- Accident Sequence Modeling
- System Modeling
- Failure Data
- Human Reliability Analysis
- Structural Evaluations
- Quantification
- Containment Response (Level 2)

3.3 INPUTS (PLANT CHANGES)

This section summarizes the inputs to the risk evaluation, which include hardware modifications, setpoint changes, procedural and operational changes associated with the Extended Power Uprate.

3.3.1 Hardware Modifications

The hardware modifications associated with the Extended Power Uprate have been identified by NMC as input to this assessment. The hardware modifications to be implemented as part of the power uprate are included in an attachment to the License Amendment Request. At the time this assessment was completed, the onsite AC distribution system modifications for EPU were not finalized. The PRA impact for these modifications, if any, will be evaluated as part of the modification process.

3.3.2 Procedural Changes

Slight adjustments to the MNGP EOPs/SAMGs will be made to be consistent with EPU operating conditions. In almost all respects, the EOPs/SAMGs are expected to remain unchanged because they are symptom-based; however, certain parameter thresholds and graphs are dependent upon power and decay heat levels and will require slight modifications. The specifics of any procedural changes associated with the Extended Power Uprate were not available prior to completion of this PRA evaluation.

Based on the GE EPU Evaluations [14], EOP variables that play a role in the PRA and which may require adjustment for the EPU include:

- Boron Injection Initiation Temperature (BIIT)
- Heat Capacity Temperature Limit (HCTL)
- Primary Containment Pressure Limit (PCPL)

These variables may require adjustment to reflect the change in power level, but will not be adjusted in a manner that involves a change in accident mitigation philosophy. The HCTL and PCPL relate to long-term scenarios, any changes in the scenario timings associated with EPU changes to these curves will be minor (e.g., changes on the order of 10 minutes over accident times greater than 3 hours) and would not significantly impact the human error probabilities in the PRA.

Any EPU related changes to the MNGP EOPs or SAMGs are considered minor perturbations to the already assessed EPG/SAG changes. Therefore, the EOP/SAMG changes as a result of the EPU will not influence the risk profile.

3.3.3 Setpoint Changes

The RPV operating pressure and the operating temperature are not being changed as part of the Extended Power Uprate. Potential setpoint changes for the EPU may include:

- Turbine overspeed
- Turbine first stage pressure steam scram bypass

Changes to the following setpoints are not anticipated for the EPU:

- RPT/ATWS high dome pressure
- RPV level trips/actuations
- MSL low pressure isolation
- MSL high flow trip (lb/hr)
- SRV setpoints

Any EPU related changes to setpoints are not expected to significantly influence the EPU risk profile and are not expected to change the conclusions of this study. Refer to Section 4.1.2.6 regarding postulated minor increase in SORV probability.

3.3.4 Plant Operating Conditions

The key plant operational modifications to be made in support of the EPU are:

- Increase in reactor thermal power from 1775 to 2004 MWt
- Feedwater/Condensate flow (and steam flow) rates will increase by approximately 13% over current licensed thermal power

RPV pressure will remain unchanged for the EPU.

In addition, no significant changes in the operating conditions of the following systems are projected at this time:

- RCIC
- ECCS Systems
- Instrument Air
- Circulating Water
- Service Water
- RBCCW

3.4 SCOPING EVALUATION

The scoping evaluation examines the hardware, procedural, setpoint, and operating condition changes to assess whether there are PRA impacts that need to be considered in addition to the increase in power level. These changes are also examined in Section 4 relative to the PRA elements that may be affected. The scoping evaluation conclusions reached are discussed in the following subsections.

3.4.1 Hardware Changes

The hardware changes required to support the EPU (see Section 3.3.1) were reviewed and determined not to result in new accident types or increased frequency of challenges to plant response. This assessment is based on review of the plant hardware modifications and engineering judgement based on knowledge of the PRA models. The majority of the changes are characterized by either:

- Replacement of components with enhanced like components
- Upgrade of existing components

The MNGP PRA program encompasses an effectively exhaustive list of hazards and accident types (i.e., from simple non-isolation transients to ATWS scenarios to internal fires to seismic events, and numerous others). Sabotage and acts of war are outside the scope of the PRA program. Extensive and unique changes to the plant would have to be implemented to result in new previously unidentified accidents.

Extensive changes to plant equipment have been shown by operating experience to result in an increase in system unavailability or failure rate during the initial testing and break-in period. There may be some short term increase in such events at Monticello but the frequency and duration of such events can not be projected. Nevertheless, it is expected that a steady state condition equivalent to (or potentially better than) current plant performance would result within approximately one year of operation with the new equipment.

3.4.2 Procedure Changes

Changes to the EOPs/SAMGs as a result of the EPU were not available prior to completion of the PRA evaluation. It is assumed that the procedural changes (e.g., modification to HCTL curve) have a minor impact on the PRA results. No changes to

the PRA are identified as a result of potential EOP/SAMG procedural changes. See Section 3.3.2.

3.4.3 Setpoint Changes

None of the planned setpoint changes listed in Section 3.3.3 will result in any quantifiable impact to the PRA. Key setpoints that play a role in the PRA are planned to remain unchanged, such as:

- Main Steam SRV opening and closing setpoints
- RPV Level Setpoints (e.g., high level trips, level actuations)
- RPV pressure setpoint (e.g., RPT/ARI)

3.4.4 Normal Plant Operational Changes

The Feedwater/Condensate flow rates will be increased to support the EPU, but this operational change is not expected to significantly impact component failure rates or initiating event frequencies used in the PRA. However, a sensitivity case is performed (refer to Section 5) that postulates significant increase in LOCA frequency due to increased erosion corrosion rates.

There are no significant systemic configuration changes as part of the EPU as far as additional trains of key equipment required to operate during plant operation.

Section 4
PRA CHANGES RELATED TO EPU CHANGES

Section 3 has examined the plant changes (hardware, procedural, setpoint, and operational) that are part of the Extended Power Uprate (EPU). Section 4 examines these changes to identify MNGP PRA modeling changes necessary to quantify the risk impact of the EPU. This section discusses the following:

- Individual PRA elements potentially affected by EPU (4.1)
- Level 1 PRA (4.2)
- Internal Fires Induced Risk (4.3)
- Seismic Risk (4.4)
- Other External Hazards Risk (4.5)
- Shutdown Risk (4.6)
- Radionuclide Release (Level 2 PRA) (4.7)

4.1 PRA ELEMENTS POTENTIALLY AFFECTED BY POWER UPRATE

A review of the PRA elements has been performed to identify potential effects associated with the Extended Power Uprate. The result of this task is a summary which dispositions all PRA elements regarding the effects of the Extended Power Uprate. The disposition consists of three Qualitative Disposition Categories.

Category A: Potential PRA change due to power uprate. PRA modification desirable or necessary

Category B: Minor perturbation, negligible impact on PRA, no PRA changes required

Category C: No change

Table 4.1-1 summarizes the results from this review. Based on Table 4.1-1, only a small number of the PRA elements are found to be potentially influenced by the power uprate.

The following PRA elements are discussed in Table 4.1-1 to summarize whether they may be affected by the Extended Power Uprate and the associated changes.

- Initiating Events
- Systemic/Functional Success Criteria, e.g.:
  - RPV Inventory Makeup
  - Heat Load to the Suppression Pool
  - Time to Boildown
  - Blowdown Loads
  - RPV Overpressure Margin
  - SRV Actuations
  - SRV Capacity for ATWS
- Accident Sequence Modeling
- System Modeling
- Failure Data
- Human Reliability Analysis
- Structural Evaluations
- Quantification
- Containment Response (Level 2)

4.1.1 Initiating Events

The evaluation has examined whether there may be increases in the frequency of the initiating events or whether there may be new types of initiating events introduced into the risk profile.

The MNGP PRA program encompasses an effectively exhaustive list of hazards and accident types (i.e., from simple non-isolation transients to ATWS scenarios to internal fires to hurricanes to toxic releases to draindown events during refueling activities, and numerous others). Extensive and unique changes to the plant would have to be implemented to result in new previously unidentified accidents; this is not the case for the MNGP EPU.

The MNGP PRA initiating events can be categorized into the following:

- Transients
- LOOP
- LOCAs
- Support System Failures
- Internal Floods
- External Events

Transients

The evaluation of the plant and procedural changes does not result in any new transient initiators, nor is there anticipated any direct significant impact on transient initiator frequencies due to the EPU.

However, a sensitivity quantification is performed that increases the Turbine Trip transient initiator frequency to bound the various changes to the BOP side of the plant (e.g., main turbine modifications).

LOOP

No change in the Loss of Offsite Power initiating event frequency is expected. Currently MNGP has certain operating configurations/conditions that require power reductions to

maintain grid stability or to respond to grid voltage changes. The same or similar conditions and operations will exist for the EPU, and are not expected to have any grid related impact on the LOOP initiating event frequency. The EPU stability analysis did not find significant impacts on grid stability due to the MNGP power uprate.

LOCAs

No significant changes to RPV operating pressure, inspection frequencies, or primary water chemistry are planned in support of the EPU; as such, no significant impact on LOCA frequencies due to the EPU can be postulated. It is anticipated that condensate and feedwater system pressures will be slightly higher due to pump replacement, particularly during system startup conditions. It is expected that this will result in a negligible impact on the frequency of LOCA initiators. However, a sensitivity case is analyzed that doubles the Large LOCA initiator frequency.

Support System Initiators

No significant changes to support systems (e.g., Instrument Air, Service Water) are planned in support of the EPU; as such, no significant impact on support system initiating event frequencies due to the EPU are postulated.

Internal Flood Initiators

No changes to pipe inspection scopes or frequencies are planned in support of the EPU; as such, no significant impact on internal flooding initiator frequencies due to the EPU are postulated.

External Event Initiators

The frequency of external event initiators (e.g., seismic events, extreme winds, fires) is not linked to reactor power or operation; as such, no impact on external event initiator frequencies due to the EPU can be postulated.

4.1.2 Success Criteria

The success criteria for the Monticello PRA are based on realistic evaluations of system capability over the 24 hour mission time of the PRA analysis. These success criteria therefore may be different than the design basis assumptions used for licensing Monticello. This report examines the risk profile changes caused by EPU from a realistic perspective to identify changes in the risk profile that may result from severe accidents on a best estimate basis. The following subsections discuss different aspects of the success criteria as used in the PRA. Appendix E provides the deterministic calculations performed to support assessment of the impacts on success criteria and sequence timing. MNGP EPU task reports were also used to assist in assessing impacts on success criteria.

4.1.2.1 Timing

Shorter times to boildown are likely on an absolute basis due to the increased power levels. The reduction in timings can impact the human error probability calculations, especially for short-term operator actions. See HRA discussion in Section 4.1.6.

4.1.2.2 RPV Inventory Makeup Requirements

The PRA success criteria for RPV makeup remains the same for the post-uprate configuration; the one minor exception is CRDH. Both high pressure (e.g., FW, HPCI, and RCIC) and low pressure (e.g., LPCI, CS, and condensate) injection systems have more than adequate flow margin for the post-uprate configuration.

CRDH remains a viable RPV makeup source at high and low pressures in the post-EPU. CRDH is a success in the CLTP PRA as the sole early injection source for transient and SORV scenarios if a second CRDH pump at nominal flow is initiated in a timely manner, or if enhanced flow actions for one CRDH pump are initiated in a timely manner.[1] The MNGP CLTP PRA also credits CRDH late in accident scenarios when decay heat is less, and in such scenarios only a single CRDH pump at nominal flow is required.

The CRDH success criteria for the EPU condition are relatively unchanged. MNGP EPU MAAP runs MNGPEPU5e - MNGPEPU5i show that enhanced CRDH flow is sufficient for high pressure makeup for transient and SORV scenarios for the EPU condition. Nominal CRDH flow with 2 pumps is also successful as the only injection source for transients and SORV scenarios for the EPU (refer to MNGP EPU MAAP runs MNGPEPU5b and MNGPEPU5d); except for the case in which the RPV remains at pressure (refer to MNGP EPU MAAP runs MNGPEPU5a and MNGPEPU5c).

4.1.2.3 Heat Load to the Pool

Energy to be absorbed by the pool during an isolation event or RPV depressurization increases for the EPU case relative to the CLTP. For non-ATWS scenarios, the RHR heat exchangers, the main condenser, and the containment vent all have capacities that exceed the increase in heat load due to extended power uprating. The heat removal capability margins are sufficiently large such that the changes in power level associated with EPU do not affect the success criteria for these systems.

Although a MNGP "successful vent initiation" MAAP run was not performed in support of this risk assessment, MAAP runs for other BWR plants show that once the containment vent is opened, per the EOPs, containment pressure decreases immediately and rapidly.

[1] Use of CRDH as an RPV injection source is an option identified in the EOPs. Various CRDH alignments can produce different flow rates into the RPV. OPS Manual Section C.5.3204 provide the instruction for use of CRDH as an RPV injection source. The first, and most simple action is to start a second pump. Addition action may be taken to further enhance ("or maximize") CRDH flow; these actions involve operator manipulations in the reactor building to open bypass valves.

The small percentage difference in decay heat level (i.e., CLTP vs. EPU) at the time of EOP vent initiation will not change this performance.

No changes to the above DHR systems to augment their capabilities for the EPU configuration are planned.

4.1.2.4 Blowdown Loads

Dynamic loads would increase slightly because of the increased stored thermal energy. This change would not quantitatively influence the PRA results. The containment analyses for LOCA under EPU conditions indicate that dynamic loads on containment remain acceptable.

4.1.2.5 RPV Overpressure Margin

The RPV dome operating pressure will not be increased as a result of the power uprate. However, the RPV pressure following a failure to scram is expected to increase slightly.

The current MNGP CLTP PRA requires two (2) SRVs to open for initial pressure control during a transient. Based on MAAP runs performed for this EPU risk assessment, this success criterion remains unchanged for the EPU. MNGP EPU MAAP runs MNGPEPU1a and MNGPEPU1a_a show that two SRVs are required for initial RPV overpressure protection during an isolation transient for the EPU configuration to maintain RPV pressure below the ASME service Level C RPV pressure of 1500 psig.

The current MNGP PRA does not require any SRVs for initial RPV overpressure control for LOCA initiators. This success criterion also remains unchanged for the EPU.

The CLTP PRA uses a success criterion of 6 of 8 SRVs required for RPV initial overpressure protection during an ATWS scenario. Based on EPU ATWS analysis, 7 of 8

SRVs are required for the uprated condition for RPV initial overpressure protection during an ATWS scenario.

4.1.2.6 SRV Actuations

Given the power increase of the EPU, one may postulate that the probability of a stuck open relief valve given a transient initiator would increase due to an increase in the number of SRV cycles.

The stuck open relief valve probability following a plant trip and SRV challenge used in the MNGP PRA is 2E-3 for transient events (basic event XVRONESRVC) and 2E-2 for ATWS scenarios (basic event XVR-ATWS-C). The MNGP PRA base stuck open relief valve probabilities may be modified using different approaches to consider the effect of a postulated increase in valve cycles. The following three approaches are considered:

1. The upper bound approach would be to increase the stuck open relief valve probability by a factor equal to the increase in reactor power (i.e., a factor of 1.13 in the case of the MNGP 113% CLTP EPU). This approach assumes that the stuck open relief valve probability is linearly related to the number of SRV cycles, and that the number of cycles is linearly related to the reactor power increase.

2. A less conservative approach to the upper bound approach would be to assume that the stuck open relief valve probability is linearly related to the number of SRV cycles, BUT the number of cycles is not necessarily directly related to the reactor power increase. In this case the postulated increase in SRV cycles due to the EPU would be determined by thermal hydraulic calculations (e.g., MAAP runs).

3. The lower bound approach would be to assume that the stuck open relief valve probability is dominated by the initial cycle and that subsequent cycles have a much lower failure rate. In this approach the base stuck open relief valve probability could be assumed to be insignificantly changed by a postulated increase in the number of SRV cycles.

Approach #1 is used here to modify the MNGP PRA stuck open relief valve probability. Therefore, the MNGP PRA base stuck open relief valve probability given a transient

initiator is increased 13% to 2.26E-3 to represent the EPU configuration, and the probability for ATWS scenarios is likewise increased 13% to 2.26E-2.

4.1.2.7 RPV Emergency Depressurization

The current MNGP PRA requires one SRV for RPV emergency depressurization in transient scenarios. MAAP cases performed in support of this EPU risk assessment (e.g., MNGPEPU1a) show that this success criterion remains unchanged by the EPU.

The CLTP MNGP PRA also assumes that two (2) SRVs are required in those instances when alternative low pressure injection system alignments of FPS crosstie or CSW are used. This success criterion is also assessed as appropriate for the EPU.

4.1.2.8 Success Criteria Summary

The Level 1 and Level 2 MNGP PRAs have developed success criteria for the key safety functions. Tables 4.1-2 through 10 summarize these safety functions and the minimum success criteria under the current power configuration and that required under the Extended Power Uprate configuration. Success criteria are summarized for the following:

- General Transients (Table 4.1-2)
- IORV, Transient w/SORV (Table 4.1-3)
- Small LOCA (Table 4.1-4)
- Medium LOCA (Table 4.1-5)
- Large LOCA (Table 4.1-6)
- ATWS Events (Table 4.1-7)
- Internal Floods (Table 4.1-8)
- ISLOCA, Breaks Outside Containment (Table 4.1-9)
- Level 2 (Table 4.1-10)

The PRA success criteria are affected by the increased boil off rate, the increased heat load to the suppression pool, and the increase in containment pressure and temperatures.

Selected MAAP runs demonstrate the significant margins associated with the installed systems. However, MAAP runs were not performed to verify success criteria for all PRA systems. For example, the high pressure and low pressure ECCS system success criteria is assumed in this assessment to remain the same for the EPU condition as for the CLTP condition based on the task analysis reports performed as part of the EPU program.

The Level 1 PRA success criteria impacts due to the EPU are as follows:

1. 7 of 8 SRVs are required for the EPU condition for RPV initial overpressure protection during an ATWS scenario.
2. CRDH as the only early injection source using 2 CRDH pumps at nominal flow now requires that the RPV be depressurized (the use of enhanced flow CRDH with a single CRDH pump is unchanged for the EPU).

These Level 1 PRA success criteria changes are addressed in the MNGP EPU risk assessment.

No changes in success criteria have been identified with regard to the Level 2 containment evaluation. The slight changes in accident progression timing and decay heat load have only minor or negligible impacts on Level 2 PRA safety functions, such as containment isolation, ex-vessel debris coolability, and challenges to the ultimate containment strength.

This assessment is consistent with GE's generic conclusions on this issue [15]:

> *"..CPPU is not expected to have a major impact on the PRA success criteria."*

4.1.3 Accident Sequence Modeling

The EPU does not change the plant configuration and operation in a manner such that new accident sequences or changes to existing accident scenario progressions result. A slight exception is the reduction in available accident progression timing for some scenarios and the associated impact on operator action HEPs (this aspect is addressed in the Human Reliability Analysis section).

This assessment for MNGP is consistent with GE's generic conclusions on this issue [14]:

> *"The basic BWR configuration, operation and response is unchanged by power uprate. Generic analyses have shown that the same transients are limiting. ... Plant-specific analyses demonstrate that the accident progression is basically unchanged by the uprate."*

4.1.4 System Modeling

The MNGP plant changes associated with the EPU do not result in the need to change any system fault trees to address changes in standby or operational configurations, or the addition of new equipment (refer to failure data discussion below regarding replacement of components with upgraded components).

Changes were made to the CRD and SRV fault tree logic to address the Level 1 PRA success criteria changes for EPU discussed in Section 4.1.2.8.

4.1.5 Failure Rate Data

The majority of the hardware changes in support of the EPU may be characterized as either:

- Replacement of components with enhanced like components
- Upgrade of existing components

Although equipment reliability as reflected in failure rates can be theoretically postulated to behave as a "bathtub" curve (i.e., the beginning and end of life phases being associated with higher failure rates than the steady-state period), no significant impact on the long-term average of initiating event frequencies, or equipment reliability during the 24 hr. PRA mission time due to the replacement/modification of plant components is anticipated, nor is such a quantification supportable at this time. If any degradation were to occur as a result of EPU implementation, existing plant monitoring programs would address any such issues. This assessment is consistent with GE's generic conclusions on this issue [15]:

> *"..CPPU is not expected to have a major effect on component or system reliability, as long as equipment operating limits, conditions, and/or ratings are not exceeded."*

No planned operational modifications as part of the MNGP EPU include operating equipment beyond design ratings. However, sensitivity cases that increase transient initiating event frequencies are quantified in this EPU risk analysis to bound the various changes to the BOP side of the plant (refer to Section 5.7 of this report).

4.1.6 Human Reliability Analysis

The Monticello risk profile, like other plants, is dependent on the operating crew actions for successful accident mitigation. The success of these actions is in turn dependent on a number of performance shaping factors. The performance shaping factor that is principally influenced by the power uprate is the time available within which to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some actions. To quantify the potential impact of this performance shaping factor, deterministic thermal hydraulic calculations using the MAAP computer code are used. Refer to Appendix E for a summary of MAAP cases performed to support the Monticello power uprate.

Discussion of Impact on Human Error Probabilities

The increased power level reduces the time available for some operator actions by small increments. The reduction in the available time is generally small compared with the total time available to detect, diagnose, and perform the actions.

Table 4.1-11 summarizes the assessment of the operator actions explicitly reviewed in support of this analysis (both Level 1 and Level 2 PRA operator actions considered). The operator actions identified for explicit review were selected based on the following criteria:

1. F-V (with respect to CDF) importance measure ≥ 5E-3
2. RAW (with respect to CDF) importance measure ≥2.0
3. F-V (with respect to LERF) importance measure ≥ 5E-3
4. RAW (with respect to LERF) importance measure ≥ 2.0
5. Time critical (≤ 30 min. available) action

These criteria have been used in past EPU risk assessments. If any of the above criteria are met for an operator action the action is maintained for explicit consideration in the EPU risk assessment. Potential HEP changes for operator actions screened out from explicit assessment in this EPU risk assessment will not have a significant impact on the quantitative results. Given that the EPU impacts on the significant HEPs modified for this study results in increasing the plant risk profile by about 7%, the non-significant HEPs if adjusted would be expected to impact the risk profile by a fraction of a percent.

In addition, of all the actions screened from further analysis, only a single action when conservatively increased to an error probability of 1.0 would result in an increase in CDF by ≥ 1E-6 or LERF by ≥ 1E-7. However, this one screened action, OIL-LOSS-Y, (related to failing to observe the need to address low fuel oil in the EDG day tank) has a long (≥ 1 hour) allowable response time such that the HEP would not be significantly

impacted by the EPU (and indeed, the action timing is not directly related to RPV initial power level).

Approximately fifty operator actions were identified for explicit consideration regarding potential timing impacts due to the EPU. MAAP calculations for the MNGP CLTP and EPU configurations were performed to determine changes in allowable operator action timings. The human error probabilities (HEPs) were then re-calculated using the same human reliability analysis (HRA) methods used in the MNGP PRA. [2]

Refer to Appendix D for a summary of the operator action screening performed for this risk assessment.

As can be seen in Table 4.1-11, the changes in timing are estimated to result in changes to some HEPs. The changes in allowable operator action timings are not always directly linear with respect to the EPU power increase (i.e., a 13% power uprate does not always correspond to a 13% reduction in operator action timings):

- Allowable time windows for some actions are not impacted by the power uprate (e.g., timings based on battery life, timings based on internal flood rates, etc.)
- Allowable time windows for LOCAs may be driven more by the inventory loss than the decay heat.
- Allowable time windows for actions related directly to RCS boil-off time during non-LOCA events are also not necessarily linear with respect to the power uprate percentage. It is not uncommon that some actions have reductions many percentage points more than the uprate percentage. This is due to various factors, such as higher initial fuel temperature for the EPU providing more initial sensible heat to the RCS water in the early time frame after a plant trip than the CLTP condition, or more integrated fluid release out SRVs in the early time frame compared to the CLTP condition.

Section 5 summarizes the increase in the CDF and LERF associated with these HEP changes (in addition to other model changes).

The risk importance measures of these actions change slightly for the EPU but do not result in changing their relative significance to the MNGP risk profile. Using the $FV_{CDF} \geq 5E\text{-}3$ and $RAW_{CDF} \geq 2.0$ as the criteria for risk significance of the operator actions, no post-initiator operator action HEP moved up past this risk significance test threshold for the EPU results. As such, no new risk significant operator actions resulted from this analysis.

The EPU SBO procedure will require the operator to manually switch HPCI suction from the torus back to the CST. According to the simulation, torus temperature may reach 170F in the last few minutes of the 4 hr coping period (HPCI operability is challenged at 170F). This action is already included in the EOPs, and it can be easily performed within 3 hours (3 MOVs and one knife switch manipulation, all in the control room), but it is included as a new time critical action given that the 170F temperature may be reached just before the 4 hr coping period for the EPU. However, this issue is not significant with respect to the PRA. The PRA does not use the 170F temperature limit for HPCI, but rather uses more realistic temperature challenge for HPCI (200F in the pool) and already includes an operator action to perform the suction transfer to the CST upon reaching 200F in the pool (the HEP for this action in the PRA, HPI-CSTS-Y, is not changed by the EPU - refer to Table 4.1-11).

No significant changes are to be made to the Control Room for the EPU that would impact the MNGP PRA human reliability analysis (HRA).

4.1.7 Structural Evaluations

This assessment did not identify issues associated with postulated impacts from the EPU on the PRA modeling of structural (e.g., piping, vessel, containment) capacities. This is consistent with GE's generic conclusions [14]:

> *"The RPV is analyzed for power uprate conditions. Transients, accident conditions, increased fluence, and past operating history are considered to recertify the vessel. Plant specific analyses at power uprate conditions demonstrates that containment integrity will be maintained....no significant effect on LOCA probability. Increase in flow rates is addressed by compliance with Generic Letter 89-08, Erosion/Corrosion in Piping..."*

4.1.8 Quantification

No changes in the MNGP PRA quantification process (e.g., truncation limit, etc.) due to the EPU have been identified (nor were any anticipated). Small changes in the quantification results (accident sequence frequencies) were realized as a result of HEP and modeling changes made to reflect the EPU.

4.1.9 Level 2 PRA Analysis

Given the minor change in Level 1 CDF results, minor changes in the Level 2 release frequencies can be anticipated. Such changes are directly attributable to the change in the Turbine Trip initiating event frequency and the minor changes in short term accident sequence timing and the impact on HEPs. (Refer to Section 4.7 for additional discussion). The accident sequence modeling in the Level 2 PRA is not impacted by the EPU.

No modeling or success criteria changes are required in the post core damage Level 2 sequences due to the EPU. The Level 2 functions are either conservatively based or are driven by accident phenomena. Refer to Table 4.1-10.

Fission product inventory in the reactor core is higher as a result of the increase in power due to the EPU. The increase in fission product inventory results in an increase in the total radioactivity available for release given a severe accident. However, this does not impact the definition or quantification of the LERF risk measure used in Regulatory Guide 1.174, and as the basis for this risk assessment. The MNGP PRA release categories are defined

based on the percentage (as a function of EOC inventories) of CsI released to the environment, which is consistent with most, if not all, industry PRAs. MAAP runs were performed for the Medium-Early and Large-Late release sequence types in the MNGP Level 2 PRA to show that these sequence types remain the same release categorizations and do not become LERF as a result of the EPU. Refer to Section 4.7 and Appendix E.

Table 4.1-1

REVIEW OF PRA ELEMENTS FOR POTENTIAL RISK MODEL EFFECTS

PRA Elements	Disposition Category	Basis
Initiating Events	B	No new initiators or increased frequencies of existing initiators are anticipated to result from the MNGP EPU. However, quantitative sensitivity cases that increase the transient and LOCA frequencies are performed as part of this analysis.
Success Criteria	B	There are a number of potential effects that could alter success criteria. These are discussed in the text. They include the following: • Time to boil down • Heat Load to the Pool • Blowdown Loads • RPV Overpressure Margin (number of SRVs/SVs required) • Depressurization (number of SRVs required)
Accident Sequences (Structure, Progression)	C	No changes in the accident sequence structure result from the increase in power rating. The accident progression is slightly modified in timing. These changes are incorporated in the Human Reliability Analysis (HRA).
System Analysis	B	No new system failure modes or significant changes in system failure probabilities due to the EPU.
Data	C	No change to component failure probabilities.
Human Reliability Analysis	A	The change in initial power level in turn results in decreases in the time available for operator actions. See discussion of operator actions in Section 4.1.6.

Table 4.1-1 (Continued)

REVIEW OF PRA ELEMENTS FOR POTENTIAL RISK MODEL EFFECTS

PRA Elements	Disposition Category	Basis
Structural	C	No changes in the structural analyses are identified that would adversely impact the PRA models.
Quantification	B	No changes in PRA quantification process (e.g., truncation limit, flag settings, etc.) due to EPU. However, a small number of changes are identified in the accident sequence quantification results. Individual basic event quantification effects are addressed under HRA.
Level 2	B	Slight changes in accident progression timing result from the increased decay heat. However, the slight changes are negligible compared with the overall timing of the core melt accident progression.

Table 4.1-2

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***GENERAL TRANSIENTS***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	Turbine bypass or 2 of 8 SRVs[9]	Same[9], [10]
Primary System Pressure Control (SRVs reclose)	All SVs/SRVs must reclose	Same (by definition)
High Pressure Injection	1 FW pump & 1 Cond. pump[1] or HPCI or RCIC or CRDH (2 pumps at nominal flow or 1 pump at "enhanced" flow) [3]	Same[11] (except nominal CRDH flow w/2 pumps now requires the RPV to be at reduced pressure to be successful for the EPU)
RP Emergency Depressurization	1 of 8 SRVs (2/8 SRVs required for FPS and CSW injection sources)	Same[12]
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump or 1 Condensate pump[2]	Same[13]
Alternate Injection	1 CRDH pump at nominal flow for late injection[3] or RHRSW A crosstie to LPCI[4] or Condensate Service Water (CSW) Injection[4] or FPS crosstie to LPCI[4]	Same[14]

Table 4.1-2

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***GENERAL TRANSIENTS***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Containment Heat Removal	Main Condenser or 1 RHR Hx Loop[6] or Containment Venting[7]	Same[14]

Notes To Table 4.1-2:

(1) One FW pump injecting, with one condensate pump providing suction, is a success for high pressure injection for a transient. FW operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(2) One condensate pump injecting is a success for low pressure injection for a transient. Operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(3) CRDH injection flow rate at MNGP is sufficiently large that it can be used as a the sole early injection source for non-LOCA and non-ATWS scenarios if a second CRDH pump is started in a timely manner, or the flow of a single pump is enhanced (via CRDH flow enhancement procedures) in a timely manner.

Later in accident sequences, many hours into the event after other injection sources have operated for some time (and have failed for some reason), CRDH is also a success but only requires one pump at nominal flow.

(4) The fire protection system alternate alignment is via LPCI and can provide 1000 gpm to the core when the RPV is at approximately 100 psi. Two (2) SRVs are required in the PRA for this alignment. Requires manual alignment. Any one of the following FPS pumping sources is a success: diesel fire pump, electric fire pump, screen wash fire pump, or pumper truck (longer term option).

Like FPS, Condensate Service Water RPV injection alignment also requires 2 SRVs for success in the PRA. CSW alignment also requires manual actions for alignment.

RHRSW A crosstie to LPCI provides significant flow and only requires a single SRV. Like FPS and CSW alignments, RHRSW crosstie also requires manual actions for alignment.

(5) <Not used.>

(6) 1 RHR pump, 1 RHR heat exchanger and 1 RHRSW pump are required for success.

(7) By design and EOPs, emergency containment venting is a success in the PRA for the containment heat removal function. The PRA credits the hard-pipe, wetwell, and drywell vent paths for containment heat removal.

(8) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgement using conservative margins.

(9) The previous 112% re-rate study (refer to MNGP document II.SMN.96.001) determined that 2 SRVs are required to lift for isolation transients for successful RPV overprotection (to prevent the RPV from exceeding 1500 psi, Service Level C). The MNGP 2005 PRA currently models that 8/8 SRVs must fail to open (basic event XVR8SRVCCN88); the PRA documentation acknowledges this, appropriately stating that 2 SRVs are required but that adjustment to this basic event to make it 7 out of 8 fail to open would not change the already very low probability (which is overwhelmingly dominated by common cause failure, such that the probability of CCF of 7 SRVs to open is the same value as CCF of 8 SRVs to open).

MNGP EPU MAAP runs MNGPEPU1a and MNGPEPU1a_a also show that two SRVs are required for initial RPV overpressure protection during an isolation transient for the EPU configuration. MNGP EPU MAAP run MNGPEPU1ax shows that 1 SRV for the CLTP case is marginal (RPV pressure just below 1500 psi); so, the CLTP assumption requiring two is reasonable.

(10) By plant design the MNGP turbine bypass is sufficient for RPV overpressure protection during a transient with the condenser heat removal path available. (Refer to MNGP EPU transient analysis.)

(11) FW/Condensate, HPCI, and RCIC, by design, have more than enough capacity to provide coolant makeup at the EPU condition for a transient initiator.

Refer to MNGP EPU MAAP runs MNGPEPU5e – MNGPEPU5h that show that "enhanced CRDH" is sufficient for high pressure makeup for transients for the EPU condition. Nominal CRDH flow with 2 pumps is also successful as the only injection source for a transient for the EPU as long as the second pump is started in a timely manner (refer to MNGP EPU MAAP runs MNGPEPU5b and MNGPEPU5d); except for the case in which the RPV remains at pressure (refer to MNGP EPU MAAP runs MNGPEPU5a and MNGPEPU5c).

(12) MAAP run MNGPEPU1a shows that 1 SRV is sufficient for RPV Emergency Depressurization for the EPU configuration for a transient initiator. The EPU risk assessment reasonably assumes the 2 SRV success criterion for use of the alternate low flow LP injection sources in the CLTP PRA remains appropriate for the EPU.

(13) LPCI, Core Spray, and Condensate, by design, have more than enough capacity to provide coolant makeup at the EPU condition. (Also refer to MAAP run MNGPEPU1a) for a transient initiator.

(14) Engineering judgment.

By plant design, the main condenser, RHR system, and emergency containment vent remain successful for the EPU condition. Also refer to MNGPEPU3 MAAP run that shows that 1 loop of SPC is effective for 24 hrs. The PRA credits RHR suppression pool cooling, shutdown cooling, and drywell spray modes.

In addition, the MNGP EPU MAAP runs (e.g., MNGPEPU5e through MNGPEPU5h) that show the lower flow CRDH system injection option is a success as an early injection source for the EPU, supports the reasonable assumption that the alternative alignments remain a success for the EPU.

Table 4.1-3

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***IORV or TRANSIENT w/SORV***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	n/a (addressed by SORV)	Same
Primary System Pressure Control (SRVs reclose)	n/a (SRV stuck-open)	Same (by definition)
High Pressure Injection	1 FW pump & 1 Cond. pump[1] or HPCI or CRDH (2 pumps at nominal flow or 1 pump at "enhanced" flow) [3]	Same[11] (except nominal CRDH flow w/2 pumps now requires the RPV to be at reduced pressure to be successful for the EPU)
RPV Emergency Depressurization	n/a (performed by SORV at t=0)	Same[9]
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump or 1 Condensate pump[2]	Same[10]
Alternate Injection	1 CRDH pump at nominal flow for late injection[3] or RHRSW A crosstie to LPCI[4] or Condensate Service Water (CSW) Injection[4] or FPS crosstie to LPCI[4]	Same[12]
Containment Heat Removal	Main Condenser or 1 RHR Hx Loop[6] or Containment Venting[7]	Same[12]

Notes To Table 4.1-3:

(1) One FW pump injecting, with one condensate pump providing suction, is a success for high pressure injection for a transient w/SORV. FW operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(2) One condensate pump injecting is a success for low pressure injection for a transient w/SORV. Operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(3) CRDH injection flow rate at MNGP is sufficiently large that it can be used as a the sole early injection source for non-LOCA and non-ATWS scenarios if a second CRDH pump is started in a timely manner, or the flow of a single pump is enhanced (via CRDH flow enhancement procedures) in a timely manner.

Later in accident sequences, many hours into the event after other injection sources have operated for some time (and have failed for some reason), CRDH is also a success but only requires one pump at nominal flow.

(4) The fire protection system alternate alignment is via LPCI and can provide 1000 gpm to the core when the RPV is at approximately 100 psi. Two (2) SRVs are required in the PRA for this alignment. Requires manual alignment. Any one of the following FPS pumping sources is a success: diesel fire pump, electric fire pump, screen wash fire pump, or pumper truck (longer term option).

Like FPS, Condensate Service Water RPV injection alignment also requires 2 SRVs for success in the PRA. CSW alignment also requires manual actions for alignment.

RHRSW A crosstie to LPCI provides significant flow and only requires a single SRV. Like FPS and CSW alignments, RHRSW crosstie also requires manual actions for alignment.

(5) <Not used.>

(6) 1 RHR pump, 1 RHR heat exchanger and 1 RHRSW pump are required for success.

(7) By design and EOPs, emergency containment venting is a success in the PRA for the containment heat removal function. The PRA credits the hard-pipe, wetwell, and drywell vent paths for containment heat removal.

(8) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgment using conservative margins.

(9) MAAP run MNGPEPU1a shows that 1 SRV is sufficient for RPV Emergency Depressurization for the EPU configuration for a transient initiator. Thus, the one SORV is considered a success for the RPV emergency depressurization function. The EPU risk assessment reasonably assumes the 2 SRV success criterion for use of the alternate low flow LP injection sources in the CLTP PRA remains appropriate for the EPU.

(10) LPCI, Core Spray, and Condensate, by design, have more than enough capacity to provide coolant makeup at the EPU condition for an SORV scenario.

(11) FW/Condensate, HPCI, and RCIC, by design, have more than enough capacity to provide coolant makeup at the EPU condition for a transient initiator. However, the RCIC system is not credited in the PRA for IORV/SORV scenarios because level will dip below TAF, causing the operators to initiate RPV emergency depressurization per the EOPs.

Refer to MNGP EPU MAAP runs MNGPEPU5e – MNGPEPU5h that show that "enhanced CRDH" is sufficient for high pressure makeup for transients for the EPU condition. Nominal CRDH flow with 2 pumps is also successful as the only injection source for a transient for the EPU as long as the second pump is started in a timely manner (refer to MNGP EPU MAAP runs MNGPEPU5b and MNGPEPU5d); except for the case in which the RPV remains at pressure (refer to MNGP EPU MAAP runs MNGPEPU5a and MNGPEPU5c).

(12) Engineering judgment.

By plant design, the main condenser, RHR system, and emergency containment vent options remain successful for the EPU condition. Also refer to MNGPEPU3 MAAP run that shows that 1 loop of SPC is effective for 24 hrs. The PRA credits RHR suppression pool cooling, shutdown cooling, and drywell spray modes.

In addition, the MNGP EPU MAAP runs (e.g., MNGPEPU5e through MNGPEPU5h) that show the lower flow CRDH system injection option is a success as an early injection source for the EPU, supports the reasonable assumption that the alternative alignments remain a success for the EPU.

Table 4.1-4

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***SMALL LOCA***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[7] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	Not required	Same
Vapor Suppression	Not required	Same
High Pressure Injection	1 FW pump & 1 Cond. pump[1] or HPCI [4]	Same[3]
RPV Emergency Depressurization	1 of 8 SRVs	Same[9]
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump or 1 Condensate pump[2]	Same[6]
Alternate Injection	RHRSW A crosstie to LPCI[5] or FPS crosstie to LPCI[5]	Same[9]
Containment Heat Removal	Main Condenser or 1 RHR Hx Loop or Containment Venting	Same[8]

Notes To Table 4.1-4:

(1) One FW pump injecting, with one condensate pump providing suction, is a success for high pressure injection for a SLOCA scenario. FW operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(2) One condensate pump injecting is a success for low pressure injection for a SLOCA. Operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(3) FW/Condensate and HPCI have more than enough capacity to provide coolant makeup at the EPU condition for a SLOCA scenario. Refer to MNGP EPU MAAP run MNGPEPU3 which shows that HPCI can function as the only injection source for a SLOCA for the EPU condition throughout the PRA 24 hour mission time.

(4) CRDH flow is not sufficient for early or late coolant makeup for LOCA scenarios. This is true for CLTP and for EPU.

(5) FPS crosstie and RHRSW crosstie are the only alternate LP systems of sufficient capacity for a SLOCA. CSW is not of sufficient capacity.

The fire protection system alternate alignment is via LPCI and can provide 1000 gpm to the core when the RPV is at approximately 100 psi. Two (2) SRVs are required in the PRA for this alignment. Requires manual alignment. Any one of the following FPS pumping sources is a success: diesel fire pump, electric fire pump, screen wash fire pump, or pumper truck (longer term option).

RHRSW A crosstie to LPCI provides significant flow and only requires a single SRV. Like FPS, RHRSW crosstie also requires manual actions for alignment.

(6) LPCI, Core Spray, and Condensate have more than enough capacity to provide coolant makeup at the EPU condition for a small LOCA. Refer to MNGP EPU MAAP run MNGPEPU4 which shows the one LPCI train is sufficient for a MLOCA.

(7) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgement using conservative margins.

(8) By plant design, the main condenser, RHR system, and emergency containment vent options remain successful for the EPU condition. Also refer to MNGPEPU3 MAAP run that shows that 1 loop of SPC is effective for 24 hrs. The PRA credits RHR suppression pool cooling, shutdown cooling, and drywell spray modes.

(9) Engineering judgment.

Table 4.1-5

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***MEDIUM LOCA***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	Not required	Same
Vapor Suppression	Not required	Same
High Pressure Injection	HPCI	Same[1]
RPV Emergency Depressurization	1 of 8 SRVs or HPCI initially available[2]	Same[2]
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump [4]	Same[5]
Alternate (Late) Injection	RHRSW A crosstie to LPCI[6] or FPS crosstie to LPCI[6]	Same[9]
Containment Heat Removal	1 RHR Hx Loop	Same[7]

Notes To Table 4.1-5:

(1) Refer to MNGP EPU MAAP run MNGPEPU4 which shows the HPCI is sufficient for a MLOCA for the EPU until the RPV sufficiently depressurizes so that LPCI or CS can then take over.

(2) HPCI operation in combination with the MLOCA will act as the method for RPV depressurization. (refer to MNGP EPU MAAP run MNGPEPU4).

(3) FW is not credited because it assumed that the MLOCA may be in a recirculation loop, thus preventing flow from reaching the core.

(4) Condensate is not credited because it is assumed that the MLOCA will deplete the hotwell before sufficient hotwell makeup can be aligned.

(5) LPCI and Core Spray have more than enough capacity to provide coolant makeup at the EPU condition for a MLOCA. Refer to MNGP EPU MAAP run MNGPEPU4 which shows the one LPCI train is sufficient for a MLOCA.

(6) FPS crosstie and RHRSW crosstie are the only alternate LP systems of sufficient capacity for a MLOCA. CSW is not of sufficient capacity. FPS and RHRSW crossties are only successful for late injection (after another injection source has already operated and failed). They are not successful as the only early injection source due to lack of available time in which to complete the manual alignments.

The fire protection system alternate alignment is via LPCI and can provide 1000 gpm to the core when the RPV is at approximately 100 psi. Requires manual alignment. Any one of the following FPS pumping sources is a success: diesel fire pump, electric fire pump, screen wash fire pump, or pumper truck (longer term option).

Like FPS, RHRSW crosstie also requires manual actions for alignment.

(7) By plant design, the RHR system remains successful for the EPU condition. Also refer to MNGPEPU3 MAAP run that shows that 1 loop of SPC is effective for 24 hrs. The PRA credits RHR suppression pool cooling and drywell spray modes for a MLOCA. The main condenser is not credited because the MSIVs will likely close due to accident signals. Shutdown cooling is also not credited for MLOCAs due to the potential break location in a recirculation loop. Containment venting is conservatively assumed not successful as the sole decay heat removal mechanism for MLOCAs and LLOCAs due to potential NPSH limitations on continued LPCI or CS injection.

(8) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgment using conservative margins.

(9) Engineering judgment.

Table 4.1-6

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***LARGE LOCA***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[6] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	Not required	Same
Vapor Suppression	< 6 WW-DW vacuum breakers stuck open is acceptable[1]	Same[7]
High Pressure Injection	N/A[4]	Same
RPV Emergency Depressurization	Not required	Same
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump	Same[3]
Alternate Injection	RHRSW A crosstie to LPCI[4] or FPS crosstie to LPCI[4]	Same[8]
Containment Heat Removal	1 RHR Hx Loop[5]	Same[8]

Notes To Table 4.1-6:

(1) Six (6) stuck open WW-DW vacuum breakers will lead to sufficient suppression pool bypass to result in containment overpressurization. This condition is assumed to lead to core damage due to loss of potential injection sources.

(2) The LLOCA initiator results in rapid depressurization of the RPV, precluding the use of the FW, HPCI, and RCIC high pressure injection systems. In addition, the CRDH system is of inadequate flow rate to keep up with the inventory loss.

(3) LPCI and Core Spray have more than enough capacity to provide coolant makeup at the EPU condition for Large LOCAs. Refer to MNGP EPU ECCS-LOCA analysis. MNGP MAAP runs MNGPEPU4 and MNGPEPU4ax show that LPCI is successful for LLOCA throughout the 24 hr PRA mission time.

(4) Insufficient time is available during a LLOCA to align FPS or RHRSW crossties for use as the sole early injection source. However, FPS and RHRSW crossties are credited for late injection after another injection source has operated and subsequently failed for some reason.

(5) By plant design, the RHR system remains successful for the EPU condition for containment heat removal. The PRA credits RHR suppression pool cooling and drywell spray modes for a LLOCA. The main condenser is not credited because the MSIVs will likely close due to accident signals. Shutdown cooling is also not credited for LLOCAs due to the potential break location in a recirculation loop. Containment venting is conservatively assumed not successful as the sole decay heat removal mechanism for MLOCAs and LLOCAs due to potential NPSH limitations on continued LPCI or CS injection.

(6) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgment using conservative margins.

(7) No change in the number of VBs for success is made for the EPU (postulating one or two more VBs required to not stick open for the EPU would not significantly change the vapor suppression failure probability).

(8) Engineering judgment.

Table 4.1-7

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***ATWS***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Reactivity Control	ARI[1] or; 1 of 2 SLC trains	Same (ARI and ADS Inhibit by definition)
Primary System Pressure Control (Overpressure)	Turbine bypass or; 6 of 8 SRVs and RPT[2]	Turbine bypass or; 7 of 8 SRVs[10] and RPT[2]
Primary System Pressure Control (SRVs reclose)	Not modeled	Same
High Pressure Injection	1 FW pump & 1 Cond. pump or HPCI	Same[3]
RPV Emergency Depressurization	3 of 8 SRVs	Same[4]
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump	Same[5]
Alternate Injection	N/A[6]	Same
Containment Heat Removal	Main Condenser[7] or 1 RHR Hx Loop[7] or WW/DW Venting[7]	Same[9]

Notes To Table 4.1-7:

(1) Alternate Rod Insertion (ARI) is a successful reactivity control measure only for electrical scram failures.

(2) The Recirculation Pump Trip (RPT) must actuate as designed and trip both recirculation pumps for initial RPV pressure control during an isolation ATWS. If turbine bypass remains available then RPT is not needed for initial pressure control.

(3) By plant design and the EOPs, FW and HPCI are successful for high pressure makeup during an ATWS. This is true for the EPU condition, as well (refer to MNGP EPU MAAP runs MNGPEPU7b and MNGPEPU7c).

(4) The CLTP PRA uses 3 SRVs as the success criterion for RPV emergency depressurization during an ATWS. This success criterion remains applicable to the EPU condition (refer to MNGP EPU MAAP run MNGPEPU7a).

(5) By plant design and the EOPs, LPCI and Core Spray are successful for low pressure makeup during an ATWS. This is true for the EPU condition, as well (refer to MNGP EPU MAAP run MNGPEPU7a).

(6) Alternate low pressure injection systems are not credited because it is assumed that insufficient time is available to perform the alignments during an ATWS.

(7) The main condenser, RHR system and emergency containment vent options remain successful for the EPU condition for containment heat removal. The PRA credits the RHR suppression pool cooling mode for an ATWS. The EOPs do not direct use of SDC during an ATWS. Only the WW and DW paths are credited for containment venting during an ATWS, as it is uncertain whether the hard-pipe vent option is of sufficient capacity.

(8) The success criteria applied for the power uprate configuration are based on MAAP calculations or engineering judgement using conservative margins.

(9) Engineering judgment.

(10) Based on EPU ATWS analysis, 7 of 8 SRVs are required for the EPU condition for RPV initial overpressure protection during an ATWS scenario.

Table 4.1-8

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***INTERNAL FLOODS***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	Turbine bypass or 2 of 8 SRVs[9]	Same[9], [10]
Primary System Pressure Control (SRVs reclose)	All SVs/SRVs must reclose	Same (by definition)
High Pressure Injection	1 FW pump & 1 Cond. pump[1] or HPCI or RCIC or CRDH (2 pumps at nominal flow or 1 pump at "enhanced" flow) [3]	Same[11] (except nominal CRDH flow w/2 pumps now requires the RPV to be at reduced pressure to be successful for the EPU)
RPV Emergency Depressurization	1 of 8 SRVs (2/8 SRVs required for FPS and CSW injection sources)	Same[12]
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump or 1 Condensate pump[2]	Same[13]
Alternate Injection	1 CRDH pump at nominal flow for late injection[3] or RHRSW A crosstie to LPCI[4] or Condensate Service Water (CSW) Injection[4] or FPS crosstie to LPCI[4]	Same[14]

Table 4.1-8

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***INTERNAL FLOODS***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[8] (113% CLTP)
Containment Heat Removal	Main Condenser or 1 RHR Hx Loop[6] or Containment Venting[7]	Same[14]

Notes To Table 4.1-8:

(1) One FW pump injecting, with one condensate pump providing suction, is a success for high pressure injection for a transient type scenario (which is in general what an internal flood scenario is, other than the flood impacts on mitigation equipment). FW operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(2) One condensate pump injecting is a success for low pressure injection for a transient. Operation in the short-term does not require hotwell make-up; but the model requires hotwell makeup for the long-term.

(3) CRDH injection flow rate at MNGP is sufficiently large that it can be used as a the sole early injection source for non-LOCA and non-ATWS scenarios if a second CRDH pump is started in a timely manner, or the flow of a single pump is enhanced (via CRDH flow enhancement procedures) in a timely manner.

Later in accident sequences, many hours into the event after other injection sources have operated for some time (and have failed for some reason), CRDH is also a success but only requires one pump at nominal flow.

(4) The fire protection system alternate alignment is via LPCI and can provide 1000 gpm to the core when the RPV is at approximately 100 psi. Two (2) SRVs are required in the PRA for this alignment. Requires manual alignment. Any one of the following FPS pumping sources is a success: diesel fire pump, electric fire pump, screen wash fire pump, or pumper truck (longer term option).

Like FPS, Condensate Service Water RPV injection alignment also requires 2 SRVs for success in the PRA. CSW alignment also requires manual actions for alignment.

RHRSW A crosstie to LPCI provides significant flow and only requires a single SRV. Like FPS and CSW alignments, RHRSW crosstie also requires manual actions for alignment.

(5) <Not used.>

(6) 1 RHR pump, 1 RHR heat exchanger and 1 RHRSW pump are required for success.

(7) By design and EOPs, emergency containment venting is a success in the PRA for the containment heat removal function. The PRA credits the hard-pipe, wetwell, and drywell vent paths for containment heat removal.

(8) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgment using conservative margins.

(9) The previous 112% re-rate study (refer to MNGP document II.SMN.96.001) determined that 2 SRVs are required to lift for isolation transients for successful RPV overprotection (to prevent the RPV from exceeding 1500 psi, Service Level C). The MNGP 2005 PRA currently models that 8/8 SRVs must fail to open (basic event XVR8SRVCCN88); the PRA documentation acknowledges this, appropriately stating that 2 SRVs are required but that adjustment to this basic event to make it 7 out of 8 fail to open would not change the already very low probability (which is overwhelmingly dominated by common cause failure, such that the probability of CCF of 7 SRVs to open is the same value as CCF of 8 SRVs to open).

MNGP EPU MAAP runs MNGPEPU1a and MNGPEPU1a_a also show that two SRVs are required for initial RPV overpressure protection during an isolation transient for the EPU configuration.

MNGP EPU MAAP run MNGPEPU1ax shows that 1 SRV for the CLTP case is marginal (RPV pressure just below 1500 psi); so, the CLTP assumption requiring two is reasonable.

(10) By plant design the MNGP turbine bypass is sufficient for RPV overpressure protection during a transient with the condenser heat removal path available. (Refer to MNGP EPU transient analysis.)

(11) FW/Condensate, HPCI, and RCIC, by design, have more than enough capacity to provide coolant makeup at the EPU condition for a transient initiator.

Refer to MNGP EPU MAAP runs MNGPEPU5e – MNGPEPU5h that show that "enhanced CRDH" is sufficient for high pressure makeup for transients for the EPU condition. Nominal CRDH flow with 2 pumps is also successful as the only injection source for a transient for the EPU as long as the second pump is started in a timely manner (refer to MNGP EPU MAAP runs MNGPEPU5b and MNGPEPU5d); except for the case in which the RPV remains at pressure (refer to MNGP EPU MAAP runs MNGPEPU5a and MNGPEPU5c).

(12) MAAP run MNGPEPU1a shows that 1 SRV is sufficient for RPV Emergency Depressurization for the EPU configuration for a transient initiator. The EPU risk assessment reasonably assumes the 2 SRV success criterion for use of the alternate low flow LP injection sources in the CLTP PRA remains appropriate for the EPU.

(13) LPCI, Core Spray, and Condensate, by design, have more than enough capacity to provide coolant makeup at the EPU condition. (Also refer to MAAP run MNGPEPU1a) for a transient initiator.

(14) Engineering judgment.

By plant design, the main condenser, RHR system and emergency containment vent options remain successful for the EPU condition. Also refer to MNGPEPU3 MAAP run that shows that 1 loop of SPC is effective for 24 hrs. The PRA credits RHR suppression pool cooling, shutdown cooling, and drywell spray modes.

In addition, the MNGP EPU MAAP runs (e.g., MNGPEPU5e through MNGPEPU5h) that show the lower flow CRDH system injection option is a success as an early injection source for the EPU, supports the reasonable assumption that the alternative alignments remain a success for the EPU.

Table 4.1-9

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS (LEVEL 1) INITIATING EVENT: ***ISLOCA, BOC***

Safety Function	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[5] (113% CLTP)
Reactivity Control	All control rods inserted (RPS electrical and mechanical success)	Same (by definition)
Primary System Pressure Control (Overpressure)	Not required	Same
Vapor Suppression	Not required	Same
High Pressure Injection	N/A[1]	Same
RPV Emergency Depressurization	Not required	Same
Low Pressure Injection	1 LPCI pump or 1 Core Spray pump	Same[2]
External Injection Sources	RHRSW A crosstie to LPCI[3] or Condensate Service Water (CSW) Injection[3] or FPS crosstie to LPCI[3]	Same[6]
Containment Heat Removal	N/A[4]	Same

Notes To Table 4.1-9:

(1) Break outside containment initiators result in rapid depressurization of the RPV, precluding the use of the FW, HPCI, and RCIC high pressure injection systems. In addition, the CRDH system is of inadequate flow rate to keep up with the inventory loss.

(2) LPCI and Core Spray have more than enough capacity to provide coolant makeup at the EPU condition for Large LOCAs (ISLOCA and Break Outside Containment scenarios are modeled as large LOCA size breaks in the PRA). (Refer to MNGP EPU ECCS-LOCA analysis.)

(3) If a break outside containment is not isolated, reactor water inventory will continue to be discharged outside the drywell which will eventually deplete the suppression pool and disable low pressure injection via loss of suction and flooding. Consequently, external injection from a virtually unlimited supply and external pump is needed for long term core cooling. The MNGP credits FPS, RHRSW, and CWS alternate injection sources. These systems draw from the river and have a virtually infinite source of water.

(4) Decay heat removal active systems are not required for unisolated breaks outside containment, since the decay heat is carried out of containment via the break.

(5) The success criteria applied for the power uprate configuration are based on MAAP calculations, GE calculations, or engineering judgement using conservative margins.

(6) Engineering judgment.

Table 4.1-10

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS: **LEVEL 2 (LERF) PRA**

Safety Functions	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[3] (113% CLTP)
Containment Isolation	Containment penetrations >2" dia. isolated	Same (by definition)
RPV Depressurization post-core damage	1 of 8 SRVs (assumed same as Level 1 PRA)	Same
Arrest Core Melt Progression In-Vessel	1 LPCI pump or 1 Core Spray pump or 1 Condensate pump or FPS crosstie or RHRSW crosstie	Same[4]
Combustible Gas Venting	Inerted containment with no oxygen intrusion during the accident or Combustible gas purge / vent	Same (by definition)
Containment Remains Intact at RPV Breach	Containment Isolation and No early containment failure modes (e.g., steam explosions) compromise containment integrity	Same (by definition)
Ex-vessel Debris Coolability	1 LPCI pump or 1 Core Spray pump or 1 Condensate pump or DW Sprays or FPS crosstie or RHRSW crosstie	Same[4]

Table 4.1-10

KEY SAFETY FUNCTIONS AND MINIMUM SYSTEM REQUIREMENTS FOR SUCCESS: **LEVEL 2 (LERF) PRA**

Safety Functions	Minimum Systems Required	
	Current PRA Power (CLTP)	EPU Power[3] (113% CLTP)
Containment Heat Removal	1 RHR Hx Loop[1] or Containment Venting[2]	Same[4]
Fission Product Scrubbing	No failure in DW or For WW airspace failure: no SP bypass (i.e., no WW-DW vacuum breakers stuck open and no SRV tail pipe failures)	Same (by definition)

Notes To Table 4.1-10:

(1) 1 RHR pump, 1 RHR heat exchanger and 1 RHRSW pump are required for suppression pool cooling or DW Sprays for Level 2 containment heat removal for post-core damage accidents proceeding with an initially intact containment.

(2) Containment venting is also a success for Level 2 containment heat removal for post-core damage accidents proceeding with an initially intact containment. The wetwell and drywell vents, and the hard-piped vent are credited.

(3) The Level 2 success criteria assessments for the power uprate configuration are made based on MAAP calculations, engineering judgment using conservative margins and industry studies.

(4) Engineering judgment.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
020-ISOL-M-Y	Fail to isolate a medium or large leak within 20 minutes	20 min.	20 min.	3.00E-01	3.00E-01	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
030-ISOL-M-Y	Fail to isolate a medium or large leak within 30 minutes	30 min.	30 min.	3.00E-02	3.00E-02	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
030-ISOL-S-Y	Fail to isolate a small leak within 30 minutes	30 min.	30 min.	3.00E-01	3.00E-01	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
060-ISOL-M-Y	Fail to isolate a medium or large leak within 60 minutes	60 min.	60 min.	3.00E-03	3.00E-03	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
060-ISOL-S-Y	Fail to isolate a small leak within 60 minutes	60 min.	60 min.	3.00E-02	3.00E-02	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
120-ISOL-S-Y	Fail to isolate a small leak within 120 minutes	120 min.	120 min.	3.00E-03	3.00E-03	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
ALT-INJ-LY	Fail to align FPS, RHRSW, CSW, or SW - hour available TSC support	n/a	n/a	8.00E-04	8.00E-04	Execution Error: No impact on HEP, this event is solely execution error (diagnosis error addressed by separate event).
ALT-POWER-Y	Fail to align alternate power supplies directly to MCC-44	>4hrs	>4hrs	5.00E-03	5.00E-03	Timing based on battery life and not directly on reactor power (action timing for this HEP does not explicitly credit the additional time until core damage after DC batteries deplete).
ASDS-DEP-Y	Fail to implement depressurization from ASDS panel	1 hr	50 min.	1.00E-02	1.00E-02	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax show time window reduced to approximately 50 min. for EPU case. Screening HEP not impacted by EPU.
ATWS-SHT-Y	Operator fails to initiate ATWS (short time available)	<1 min.	<1 min.	1.00E+00	1.00E+00	ASEP Upper Bound TRC curve.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
CHR-DET–Y	Fail to identify need for containment heat removal	8 hrs	6.8 hrs	1.00E-06	1.00E-06	Diagnosis Error: Timing based on time to SP/T = 200F for transients with no SPC. MNGP EPU MAAP run MNGPEPU9 shows the time is t=6.8 hrs for EPU condition. ASEP Lower Bound TRC curve, and 1E-6 HEP minimum threshold in MNGP PRA. HEP unchanged.
CRD-LSBYPY	Fail to restore CRDH after LOSP and ECCS load shed	25 min.	21 min	8.00E-02	1.23E-01	MNGP EPU MAAP runs MNGPEPU5d and MNGPEPU5dx show that the time available is reduced approximately 15% for the EPU (using times to maximize core temp). EPU diagnosis time is 11 min. and execution time is 10 min. ASEP Median TRC curve.
CRD-PUMP-Y	Fail to start second CRDH pump from control room	25 min.	21 min	9.00E-03	1.40E-02	MNGP EPU MAAP runs MNGPEPU5d and MNGPEPU5dx show that the time available is reduced approximately 15% for the EPU (using times to maximize core temp). EPU diagnosis time is 20 min. and execution time is 1 min. ASEP Median TRC curve.
CRD-VALV-Y	Fail to maximize CRDH flow - valves in RB	25 min.	21 min	4.00E-02	5.27E-02	MNGP EPU MAAP runs MNGPEPU5i and MNGPEPU5ix show that the time available is reduced approximately 15% for the EPU (using times to maximize core temp). EPU diagnosis time is 14 min. and execution time is 7 min. ASEP Median TRC curve.
CRIT-DET-Y	Fail to detect criticality issue - long time available	30 min.	30 min.	1.18E-04	1.18E-04	Diagnosis Error: This action error applies to ATWS scenarios in which the turbine is online. An indefinite, long time is available to the operator; the PRA conservatively assumes 30 mins. available. This timing assumption is not changed by the EPU. ASEP Lower Bound TRC curve. Base PRA mistakenly used 40 min. for the HEP calculation; base HEP revised in this calculation to use the correct base value of 30 min.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
CST-FILL-Y	Fail to refill the CSTs	>15 hrs	<15 hrs	1.00E-03	1.00E-03	Timing based on CST inventory depletion due to use for RPV coolant makeup long term. CLTP PRA assumes time available >15 hrs, and 1 hr required for alignment. EPU time available would be reduced, but would have to be reduced unrealistically (by 10 hrs or more) to change the CLTP HEP which is dominated by execution error. ASEP Median TRC curve.
DEP-02MN-Y	Fail RPV depressurization within 2 minutes	5 min.	4.4 min.	2.50E-01	5.10E-01	This action used in isolation ATWS scenarios with failure of all HP injection. The CLTP PRA estimates 5 minutes available (diagnosis time of 2 min. and execution time of 3 min.). MNGP EPU MAAP runs MNGPEPU7a and MNGPEPU7ax show that this timing is not reduced significantly (<10%) for the EPU, a 13% reduction is assumed in the EPU risk assessment. EPU time available is estimated at 4.4 min. (diagnosis time of 1.4 min. and execution time of 3 min.). ASEP Lower Bound TRC curve. CLTP base PRA mistakenly used 3 min. diagnosis for the HEP calculation; base HEP revised in this calculation to use the correct base diagnosis time of 2 min.
DEP-12MN-Y	Fail RPV depressurization within 12 minutes	15 min.	13.1 min.	5.20E-03	9.84E-03	This action is applicable to MLOCA scenarios with no HP injection available. MNGP EPU MAAP runs MNGPEPU8b and MNGPEPU8bx indicate that the time is reduced 10-13% for the EPU, a 13% reduction is assumed for the EPU. EPU time available estimated at 13.1 min (diagnosis time of 10.1 min. and execution time of 3 min.). ASEP Lower Bound TRC curve.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
DEP-50MN-Y	Fail RPV depressurization within 50 minutes	50 min.	42 min.	1.80E-04	1.90E-04	This action is applicable to non-LOCA and non-ATWS scenarios with no HP injection available. MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min. (diagnosis time is 39 min. and execution time of 3 min). ASEP Lower Bound TRC curve.
DEP-HOUR-Y	Fail RPV depressurization >an hour available	103 min.	103 min.	1.60E-04	1.60E-04	This action is applicable to non-LOCA and non-ATWS scenarios with HP injection initially available, but RPV ED required later for other reasons (e.g., HCTL, HP injection failure). CLTP assumes a diagnosis time of 100 minutes, and an execution time of 3 mins. MNGP EPU MAAP runs MNGPEPU8c and MNGPEPU8d show that for scenarios requiring late RPV ED due to issues such as HCTL or HP injection failure significantly more than 100 mins. remain before core damage occurs. Thus, the CLTP time available for this action is unchanged for the EPU. ASEP Lower Bound TRC curve.
DEP-PD-Y	Fail to depressurize reactor after core damage, but before vessel penetration	2 hrs	~2 hrs	1.00E-01	1.00E-01	Timing based on post-core damage accident progression assumptions and time to RPV melt-through. Screening HEP not impacted by EPU.
DW-VENT-PRG	Fail to prevent H2 burn failing containment by vent/purge	< 30 min.	< 30 min.	1.00E+00	1.00E+00	
FLOODRB16Y	Fail to flood RB within 1-6 hours after torus leak	1-6 hrs	1-6 hrs.	3.00E-01	3.00E-01	Timing based on internal flooding issues and not directly on reactor power.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
FW-CNTRL-Y	Fail to control FW as high pressure injection source following transient	15 min.	12 min.	4.60E-03	5.46E-03	The available action time is based on the time to reach TAF for an isolation transient with loss of all HP injection. MNGP EPU MAAP run MNGPEPU8a show that this time is approximately t=12 min. for the EPU power level. EPU time available estimated at 12 mins (diagnosis time of 11 min. and execution time of 1 min.). ASEP Lower Bound TRC curve.
FW-REFLG-Y	Fail to identify reference leg leak	7 min.	5.5 min.	4.00E-02	6.94E-02	The time available is based on the time to reach TAF for a ref. leg break event with no high pressure injection. Time available for CLTP estimated at t=7 mins. MNGP EPU MAAP runs MNGPEPU6c, MNGPEPU6cx, MNGPEPU1b and MNGPEPU1bx indicate that this time frame is reduced approximately 20-22% due to the EPU. EPU time available estimated at 5.5 mins. (diagnosis time of 4.5 min. and execution time of 1 min.). ASEP Lower Bound TRC curve.
HPI-CSTS-Y	Fail to defeat high torus level suction transfer	1 hr	1 hr	3.00E-03	3.00E-03	This action applies to scenarios with pool temperature reaching 200F and need to switch HPCI/RCIC suction to CST to prevent failure of pump due to overheating. Timing of 1 hr. used in CLTP not based directly on reactor power, this time is not adjusted for the EPU. ASEP Lower Bound TRC curve.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
LEVEL-05-Y	Fail to detect need for injection within 5 minutes of compelling signal	30 min.	26 min.	5.00E-02	1.00E+00	Diagnosis Error: Time available in CLTP PRA based on time to core damage for SLOCA type scenarios with no HP injection, estimated at t=30 minutes and 25 minutes to execute the action (thus, 5 min. diagnosis time). MNGP EPU MAAP runs MNGPEPU6c and MNGPEPU6cx show that this time frame is reduced to approximately t=26 mins (thus, 1 min. diagnosis time). ASEP Lower Bound TRC curve.
LEVEL-25-Y	Fail to detect need for injection within 25 minutes of compelling signal	50 min.	42 min.	6.00E-04	1.72E-03	Diagnosis Error: This action is applicable to non-LOCA and non-ATWS scenarios with no HP injection available. The CLTP PRA estimates the available window at 50 minutes and 25 minutes to execute the action (thus, 25 min. diagnosis time). MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min. (diagnosis time is 17 min. and execution time of 25 min). ASEP Lower Bound TRC curve.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
LEVEL-45-Y	Fail to detect need for injection within 45 minutes of compelling signal	~1 hr	~1 hr.	1.00E-05	1.00E-05	Diagnosis Error: This action is applicable to non-LOCA and non-ATWS scenarios with HP injection initially available, but RPV ED required later for other reasons (e.g., HCTL, HP injection failure). CLTP assumes diagnosis time available is 45 minutes, then an additional 25 minutes for execution (thus, total time available greater than 1 hr.) MNGP EPU MAAP runs MNGPEPU8c and MNGPEPU8d show that for scenarios requiring late RPV ED due to issues such as HCTL or HP injection failure that significantly more than 1 hour remains before core damage occurs. Thus, the CLTP diagnosis time for this action of 45 mins. is unchanged for the EPU. ASEP Lower Bound TRC curve.
L-LONG—Y	Operator fails to inject boron using SBLC - long time available	>1 hr	>1 hr	4.00E-04	4.00E-04	This action error applies to ATWS scenarios in which the turbine is online. An indefinite, long time is available to the operator; the PRA conservatively assumes > 1 hr. available. This timing assumption would not be changed by the EPU. ASEP Lower Bound TRC curve. In addition, the HEP is dominated by execution error.
OIL-LOSS-HY	Fail to identify need to address loss of fuel flow to EDG day tanks - high	>1 hr	>1 hr	1.00E-01	1.00E-01	Timing based on EDG fuel consumption and not directly on reactor power. Screening HEP not impacted by EPU.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
PUMPER-L-Y	Fail to provide FPS supply from fire pumper truck - hours available	6 hrs	6 hrs	1.00E-03	1.00E-03	The available time is estimated in the CLTP PRA based on the time to core damage for an SBO, with HPCI or RCIC initial operation but subsequent failure due to battery depletion. The CLTP PRA estimates that >6hrs are available before core damage in such scenarios (t=6 hrs is used in the CLTP PRA for this HEP). MNGP EPU MAAP run MNGPEPU8c shows core damage occurs at t=6.6 hrs for such scenarios for the EPU. As such, the 6 hr available time for this action is not adjusted for the EPU. ASEP Median TRC curve. Dominated by execution error.
RCIC-MAN-Y	Fail to manually operate RCIC	n/a	n/a	5.00E-02	5.00E-02	Execution Error: No impact on HEP, this event is solely execution error (diagnosis error addressed by separate event).
REC-EDG-30	Fail to recover EDG within 30 minutes	30 min.	30 min.	8.5E-01	8.5E-01	Timing based on industry data and associated LOOP event tree modeling assumptions. Timing and probability not impacted by EPU.
REC-EDG-11/6	Fail to recover EDG within 11 hours, given failure to recover w/i 6 hours	11 hrs / 6 hrs	11 hrs / 6 hrs	7.3E-01	7.3E-01	Nominal times of 11 hrs and 6 hrs still appropriate for EPU (see EPU MAAP run MNGPEPU8c). Existing recovery failure probability already high. Time frame is long and AC recovery curves flatten out at such lengthy time frames, such that any postulated change to this recovery probability would not have a significant impact on risk.
REC-EDG-12/11	Fail to recover EDG within 12 hours, given failure to recover w/i 11 hours	12 hrs / 11 hrs	11 hrs / 11 hrs	9.3E-01	1.0E+00	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-EDG-16/12	Fail to recover EDG within 16 hours, given failure to recover w/i 12 hours	16 hrs / 12 hrs	16 hrs / 11 hrs	9.0E-01	8.5E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
REC-EDG-22/12	Fail to recover EDG within 22 hours, given failure to recover w/i 12 hours	22 hrs / 12 hrs	22 hrs / 11 hrs	7.3E-01	6.5E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-EDG-3/50	Fail to recover EDG within 3 hours, given failure to recover w/i 50 minutes	3 hrs /50 mins.	3 hrs /42 mins.	6.9E-01	6.6E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-EDG-50/30	Fail to recover EDG within 50 minutes, given failure to recover w/i 30 minutes	50 min. / 30 min.	42 min. / 30 min.	9.1E-01	9.4E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-EDG-6/3	Fail to recover EDG within 6 hours, given failure to recover w/i 3 hours	6 hrs / 3 hrs	6 hrs / 3 hrs	5.1E-01	5.1E-01	Nominal times of 6 hrs and 3 hrs still judged reasonable for EPU.
REC-OSP-30	Fail to recover offsite power within 30 minutes	30 min.	30 min.	6.8E-01	6.8E-01	Timing based on industry data and associated LOOP event tree modeling assumptions. Timing and probability not impacted by EPU.
REC-OSP-10/6	Fail to recover OSP within 10 hours, given failure to recover w/i 6 hours	10 hrs / 6 hrs	10 hrs / 6 hrs	8.0E-01	8.0E-01	Nominal times of 10 hrs and 6 hrs still judged reasonable for EPU.
REC-OSP-11/6	Fail to recover OSP within 11 hours, given failure to recover w/i 6 hours	11 hrs / 6 hrs	11 hrs / 6 hrs	7.5E-01	7.5E-01	Nominal times of 11 hrs and 6 hrs still appropriate for EPU (see EPU MAAP run MNGPEPU8c). Existing recovery failure probability already high. Time frame is long and AC recovery curves flatten out at such lengthy time frames, such that any postulated change to this recovery probability would not have a significant impact on risk.
REC-OSP-12/11	Fail to recover OSP within 12 hours, given failure to recover w/i 11 hours	12 hrs / 11 hrs	11 hrs / 11 hrs	9.2E-01	1.0E+00	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-OSP-16/12	Fail to recover OSP within 16 hours, given failure to recover w/i 12 hours	16 hrs / 12 hrs	16 hrs / 11 hrs	8.0E-01	7.3E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
REC-OSP-22/12	Fail to recover OSP within 22 hours, given failure to recover w/i 12 hours	22 hrs / 12 hrs	22 hrs / 11 hrs	5.0E-01	4.5E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-OSP-29/30	Fail to recover OSP within 2.9 hours, given failure to recover w/i 30 minutes	2.9 hrs / 30 min.	2.9 hrs / 30 min.	4.2E-01	4.2E-01	No change assumed for 2.9 hr post-core damage progression time frame, time reasonable.
REC-OSP-3/50	Fail to recover OSP within 3 hours, given failure to recover w/i 50 minutes	3 hrs / 50 mins.	3 hrs / 42 mins.	4.3E-01	4.1E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-OSP-34/22	Fail to recover OSP within 34 hours, given failure to recover w/i 22 hours	34 hrs / 22 hrs	34 hrs / 22 hrs	5.0E-01	5.0E-01	Existing recovery failure probability already high. Time frame is long and AC recovery curves flatten out at such lengthy time frames, such that any postulated change to this recovery probability would not have a significant impact on risk.
REC-OSP-50/30	Fail to recover OSP within 50 minutes, given failure to recover w/i 30 minutes	50 min. / 30 min.	42 min. / 30 min.	8.5E-01	9.0E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-OSP-6/3	Fail to recover OSP within 6 hours, given failure to recover w/i 3 hours	6 hrs / 3 hrs	6 hrs / 3 hrs	6.0E-01	6.0E-01	Nominal times of 6 hrs and 3 hrs still judged reasonable for EPU.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
RHRCS-MANY	Fail to manually operate equipment outside of control room before core damage	100 min.	100 min.	4.10E-03	4.10E-03	This action is applicable to non-LOCA and non-ATWS scenarios with HP injection initially available, but RPV ED required later for other reasons (e.g., HCTL, HP injection failure). CLTP assumes time available is 100 minutes (diagnosis time of 90 min. and execution time of 10 min.). MNGP EPU MAAP runs MNGPEPU8c and MNGPEPU8d show that for scenarios requiring late RPV ED due to issues such as HCTL or HP injection failure that more than 100 mins. remain before core damage occurs. Thus, the CLTP time in this action of 100 mins. is unchanged for the EPU. ASEP Median TRC curve. Dominated by execution error.
RHR-DHR-AY	Fail to align RHR for CHR - ATWS	25 min.	21.8 min.	1.40E-02	2.19E-02	This action is applicable to ATWS scenarios with HP injection and successful SLC. Time available to align SPC depends upon time of SLC injection and whether the initiator is an isolation event. CLTP PRA assumes that 25 minutes are available (diagnosis time of 20 mins. and execution time of 5 mins.). This time is judged conservative. MNGP EPU runs MNGPEPU7b, MNGPEPU7bx, MNGPEUP7c and MNGPEPU7cx show that with delayed SLC injection and no SPC initiation, critical impacts do not occur until about t=45 mins when the pool reaches 200F and HPCI operability become an issue. Although the 25 min. time available estimate from the CLTP is judged still appropriate for the EPU, the EPU risk assessment reduces this time available by 13% to t=21.8 mins (diagnosis time of 16.8 min. and execution time of 5 min.). ASEP Median TRC curve.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
RHR-DHR--Y	Fail to align RHR for CHR, when attempted (non-ATWS)	8 hrs.	6.8 hrs	1.60E-05	1.60E-05	Execution Error: Time window same as for CHR-DET--Y; however, this is an execution error contribution, the low error rate is due to multiple applicable error recovery factors (long time frame, other operators, etc.). The reduction in time available due to the EPU does not change the execution error rate. Diagnosis contribution treated by separate basic event CHR-DET--Y.
SD-NOTRIPY	Fail to prevent turbine trip while shutting down	5 min.	4.4 min.	2.00E-01	2.27E-01	This action is for bypassing the MSIV level interlocks and is applicable to ATWS scenarios with the MSIVs open. The time available depends upon a number of factors, such as which HP systems are available and how long operators take to reduce level. The CLTP PRA assumes the available diagnosis time is t=5 min. The CLTP diagnosis time is reduced 13% for the EPU. ASEP Median TRC curve. Base PRA mistakenly selected 0.3 off the ASEP curve instead of the correct base value of 0.20; base HEP revised in this calculation to use the correct base HEP of 0.20.
SHED-DET-Y	Fail to identify load shedding as cause of system failure	30 min.	30 min.	1.00E-03	1.00E-03	Timing based on battery life and load shedding impact. Timing and probability not impacted by EPU.
SLC-INI-LY	Fail to initiate SLC - long time available	>1 hr	>1 hr.	4.00E-04	4.00E-04	This action error applies to ATWS scenarios in which the turbine is online. An indefinite, long time is available to the operator; the PRA assumes > 1 hr. available. This timing assumption is not changed by the EPU. ASEP Lower Bound TRC curve. In addition, the HEP is dominated by execution error.

Table 4.1-11

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
SLC-INI-SY	Fail to initiate SLC - short time available	13.5 min.	11.8 min.	4.40E-03	6.17E-03	Total time available reduced 13%. MNGP EPU MAAP runs MNGPEPU7a, MNGPEPU7b, and MNGPEPU7c show that that such a time frame for SLC injection is successful for the EPU condition. ASEP Lower Bound TRC curve.
SLC-LVL1-Y	Fail to control reactor level (fail SLC), given nominal conditions	10 min.	8.7 min.	1.00E-02	1.53E-02	Total time available reduced 13%. EPU diagnosis time of 8.2 min. and execution time of 0.5 min. ASEP Lower Bound TRC curve.
SLC-LVL2-Y	Fail to control reactor level (fail SLC), given challenging conditions	13.5 min.	11.8 min.	1.30E-02	1.97E-02	Total time available reduced 13%. EPU diagnosis time of 11.3 min. and execution time of 0.5 min. ASEP Lower Bound TRC curve.
VENT-CHR-Y	Fail to align containment venting as means of CHR	8 hrs.	6.8 hrs	3.10E-05	3.68E-05	Timing based on time to SP/T = 200F for transients with no SPC. MNGP EPU MAAP run MNGPEPU9 shows the time is t=6.8 hrs for EPU condition. ASEP Median TRC curve.
X-DEP-15-Y	Operator fails to depressurize reactor within 15 minutes	15 min.	15 min.	5.20E-03	5.20E-03	This action is used in high pressure ATWS core damage scenarios. The CLTP PRA assumes 15 min. available (diagnosis time of 12 min. and execution time of 3 mins.). The time available is based on post-accident progression modeling assumptions and not directly on core power. This time frame is not changed for the EPU. ASEP Lower Bound TRC curve.

4.2 LEVEL 1 PRA

Section 4.1 summarized possible effects of the EPU by examining each of the PRA elements. This section examines possible EPU effects from the perspective of accident sequence progression. The dominant accident scenario types (classes) that can lead to core damage are examined with respect to the changes in the individual PRA elements discussed in Section 4.1.

Loss of Inventory Makeup Transients

Loss of inventory accidents (non-LOCA) are determined by the number of systems, their success criteria, and operator actions for responding to their demands. The following bullets summarize key issues:

- FW, Condensate, HPCI, RCIC and LP ECCS systems - all of these systems have substantial margin in their success criteria relative to the EPU power increase to match the coolant makeup flow required for postulated accidents.

- CRDH - CRDH remains a viable RPV makeup source at high and low pressures in the EPU. CRDH is a success in the CLTP PRA as the sole early injection source for transient and SORV scenarios, and is also successful late in accident scenarios. The CRDH success criteria for the EPU condition are relatively unchanged; the one exception is that early CRHD using two pumps at nominal flow requires the RPV to be depressurized for CRDH to be a success for the EPU. This model change is included in this EPU risk assessment.

- Alternative LP RPV Injection Systems – the CLTP PRA credits RHRSW crosstie, FPS crosstie, and Condensate Service Water (CSW) injection. The RHRSW and FPS alignments have the greater flow rate potential, but all require manual alignments. Their use is sequence specific. No changes are identified to the modeling of these systems for the EPU.

- The success criterion used in the CLTP PRA for the number of SRVs required to function to assure RPV emergency depressurization is a single (1) SRV. Based on the MAAP evaluations (e.g., MNGPEPU1a),

the one (1) SRV success criterion remains adequate for the EPU condition.

Operator actions include emergency depressurization and system control and initiation. The injection initiation/recovery and emergency depressurization timings are slightly impacted by the EPU. As such, changes to the existing risk profile associated with loss of inventory makeup accidents result.

ATWS

Following a failure to scram coupled with additional failures, a higher power level and increase in suppression pool temperature would result for the EPU configuration compared with the current Monticello configuration (assuming similar failures).

The necessary relief capacity to prevent exceeding the Service Level C RPV pressure limit of 1500 psig is modeled in the current MNGP CLTP PRA as requiring 6 of 8 SRVs to open. As discussed earlier in Section 4.1.2.5, this PRA success criterion is assessed to be 7 of 8 SRVs required to open for the EPU condition.

The increased power level reduces the time available to perform operator actions. Refer to Table 4.1-11 for changes in ATWS related HEPs, as well as HEPs for other accident types. Given these ATWS HEP changes, changes to the existing risk profile associated with ATWS accidents result.

LOCAs

The blowdown loads may be slightly higher because of the higher initial power. The Mark I Containment Loads Program and the Monticello specific containment loads program have shown that these loads are acceptable for the CLTP. The GE task analyses confirm that the SSCs remain acceptable after EPU.

The success criteria for the systems to respond to a LOCA are discretized by system trains. Sufficient margin is available in these success criteria to allow adequate core cooling for EPU.

The allowable timings associated with operator actions for RPV emergency depressurization for SLOCAs and MLOCAs (LLOCAs never require emergency depressurization) are slightly impacted for the EPU. As such, changes to the existing risk profile associated with LOCA accidents result.

SBO

Station Blackout represents a unique subset of the loss of inventory accidents identified above. The station blackout scenario response is almost totally dominated by AC and DC power issues. In all other respects, SBO sequences are like the transients discussed above. Extended power uprate will not increase the loads on diesel-generators or batteries. As discussed earlier, the success criteria for mitigating systems is unchanged for the EPU.

The dominant operator action during SBO accidents is offsite AC recovery. The AC recovery failure probability is based on statistical analyses of recovery of offsite power following industry LOOP events and not on HEP calculations. Offsite AC recovery failure probabilities in the MNGP PRA are not impacted by the EPU.

However, a few operator actions are impacted by the reduced available timings of the EPU, and are propagated through the SBO accident sequences (refer to Table 4.1-11).

In addition, an accident sequence assumption in the CLTP related to the length of time that HPCI or RCIC can operate in long term scenarios before the pool heats up to the 200F challenge point for HPCI and RCIC is adjusted for the EPU. The CLTP assumes that pool heatup to 200F during long-term SBO scenarios with HPCI or RCIC operating

(with batteries being charged) occurs at t=12 hrs. This time frame is reduced to t=11 hrs for the EPU condition (refer to Appendix E MAAP run MNGPEPU8d). This issue is addressed in the EPU risk assessment by requiring AC recovery for such sequences at t=11 hrs versus t=12 hrs (the risk impact is non-significant).

As such, minor changes to the existing risk profile associated with SBO accidents result.

Loss of Containment Heat Removal

Sequences which involve the loss of containment heat removal (Class II accident sequences) are affected slightly in terms of the time to reach containment Primary Containment Pressure Limit (PCPL) or ultimate pressure, however the success criteria for the key systems (RHR, Main Condenser, and containment vent) in the loss of containment heat removal accident sequences are not affected.

Other systems (e.g., DW coolers, RWCU) are considered marginal or inadequate for containment heat removal even for the CLTP PRA. Such systems remain inadequate for the EPU PRA.

The time available to initiate containment heat removal measures is measured in many hours in the PRA for non-ATWS scenarios. The reduction in this very long time frame due to the EPU has no significant impact on the HEPs for containment heat removal initiation for non-ATWS scenarios. The time available for ATWS scenarios is assumed in the CLTP PRA to be less than an hour; timing reductions due to the EPU result in a measurable change in the HEP for containment heat removal alignment for ATWS scenarios (refer to Table 4.1-11).

The increased power level decreases the time to reach the EOP HCTL curve and requiring RPV emergency depressurization. These HEP changes will have a minor impact on the Class II accident sequences.

Minor changes to the risk profile associated with Class II (loss of decay heat removal) accidents result.

4.3 INTERNAL FIRES INDUCED RISK

The Monticello plant risk due to internal fires was evaluated in 1995 as part of the MNGP Individual Plant Examination of External Events (IPEEE) Submittal. [10] EPRI FIVE Methodology and Fire PRA Implementation Guide screening approaches and data were used to perform the MNGP IPEEE fire PRA study. [5,6,7]

Consistent with the FIVE Methodology and the requests of the NRC IPEEE Program, the MNGP IPEEE fire PRA is an analysis that identifies the most risk significant fire areas in the plant using a screening process and by calculating conservative core damage frequencies for fire scenarios. As such, the accident sequence frequencies calculated for the MNGP fire PRA are not a best estimate calculation of plant fire risk and are not acceptable for integration with the best estimate MNGP internal events PRA results for comparison with Regulatory Guide 1.174 acceptance guidelines. The screening attributes of the fire PRA are summarized below.

4.3.1 Attributes of Fire PRA

Fire PRAs are useful tools to identify design or procedural items that could be clear areas of focus for improving the safety of the plant. Fire PRAs use a structure and quantification technique similar to that used in the internal events PRA.

Historically, since less attention has been paid to fire PRAs, conservative modeling is common in a number of areas of the fire analysis to provide a "bounding" methodology for fires. This concept is contrary to the base internal events PRA which has had more

analytical development and is judged to be closer to a realistic assessment (i.e., not conservative) of the plant.

There are a number of fire PRA topics involving technical inputs, data, and modeling that prevent the effective comparison of the calculated core damage frequency figure of merit between the internal events PRA and the fire PRA. These areas are identified as follows:

Initiating Events: The frequency of fires and their severity are generally conservatively overestimated. A revised NRC fire events database indicates the trend toward both lower frequency and less severe fires. This trend reflects the improved housekeeping, reduction in transient fire hazards, and other improved fire protection steps at nuclear utilities. The database used in the Monticello fire assessment used significantly older data that is not judged applicable. In addition, it reflects conservative judgments regarding fire severity.

System Response: Fire protection measures such a sprinklers, CO_2, fire brigades may be given minimal (conservative) credit in their ability to limit the spread of a fire. Therefore, the severity of the fire and its impact on requirements is exacerbated.

In addition, cable routings are typically characterized conservatively because of the lack of data regarding the routing of cables or the lack of the analytic modeling to represent the different routings. This leads to limited credit for balance of plant systems that are extremely important in CDF mitigation.

Sequences: Sequences may subsume a number of fire scenarios to reduce the analytic burden. The subsuming of initiators and sequences is done to envelope those sequences included. This causes additional conservatism.

Fire Modeling: Fire damage and fire propagation are conservatively characterized. Fire modeling presents bounding approaches regarding the fire immediate effects (e.g.,

all cables in a tray are always failed for a cable tray fire) and fire propagation.

HRA: There is little industry experience with crew actions under conditions of the types of fires modeled in fire PRAs. This has led to conservative characterization of crew actions in fire PRAs. Because the CDF is strongly correlated with crew actions, this conservatism has a profound influence on the calculated fire PRA results.

Level of Detail: The fire PRAs may have a reduced level of detail in the mitigation of the initiating event and consequential system damage.

Quality of Model: The peer review process for fire PRAs is less well developed than for internal events PRAs. For example, no industry standard, such as NEI 00-02, exists for the structured peer review of a fire PRA. This may lead to less assurance of the realism of the model.

The fire PRA is subject to more modeling uncertainty than the internal events PRA evaluations. While the fire PRA is generally self-consistent within its calculational framework, the fire PRA calculated quantitative risk metric does not compare well with internal events PRAs because of the number of conservatisms that have been included in the fire PRA process. Therefore, the use of the fire PRA figure of merit as a reflection of CDF may be inappropriate. Any use of fire PRA results and insights should properly reflect consideration of the fact that the "state of the technology" in fire PRAs is less evolved than the internal events PRA.

Relative modeling uncertainty is expected to narrow substantially in the future as more experience is gained in the development and implementation of methods and techniques for modeling fire accident progression and the underlying data.

Fire PRA risk is dominated by fire-induced equipment failures. As such, fire PRA results are less impacted by changes in operator actions timings than the internal events PRA

results. This can be seen in the fire risk results performed for the previous MNGP re-rate, as documented in Reference [8]. That study showed the percentage CDF increase for fire risk was estimated at approximately one-third the percentage increase for the internal events CDF increase. The re-rate analysis of Reference [8] was performed using the conservative fire screening quantifications from the MGNP IPEEE.

Like most sites in the U.S., MNGP does not currently maintain a fire PRA. Rather than re-perform the analyses from Reference [8] for this uprate, the general conclusions are used here to qualitatively estimate a percentage increase in the fire risk profile for MNGP.

It is estimated here that the MNGP fire PRA CDF would increase by approximately 2 to 3 percent due to the EPU (i.e., ~⅓ of the internal events 7.8% increase) based on the general conclusions of Reference [8].

This fire impact assessment did not involve re-performing the MNGP IPEEE internal fires analyses. Similarly, plant walkdowns for internal fire risk issues were not re-performed in support of this assessment.

The impact of the EPU on the different aspects of fire risk modeling are assessed here with the approach above, and based on knowledge of fire PRA and the modifications for the EPU (e.g., no significant changes to fire protection systems, combustible loadings, etc.). Based on this assessment, it is concluded that no unique or significant impacts on fire risk result from the EPU.

4.4 SEISMIC RISK

The Monticello seismic risk analysis was performed as part of the Individual Plant Examination of External Events (IPEEE). [10] Monticello performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The

SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the seismic risk evaluation.

Based on a review of the Monticello IPEEE and the key general conclusions identified earlier in this assessment, the conclusions of the SMA are judged to be unaffected by the EPU. The EPU has little or no impact on the seismic qualifications of the systems, structures and components (SSCs). Specifically, the power uprate results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event, are judged not to alter the results of the SMA.

The decrease in time available for operator actions, and the associated increases in calculated HEPs, is judged to have a non-significant impact on seismic-induced risk. Industry BWR seismic PRAs have typically shown (e.g., Peach Bottom NUREG-1150 study [18]; Limerick Generating Station Severe Accident Risk Assessment [19]; NUREG/CR-4448 [20]) that seismic risk is overwhelmingly dominated by seismic induced equipment and structural failures.

Based on the above discussion it is judged that the percentage increase in the MNGP seismic risk due to the EPU is much less than that calculated for internal events.

This seismic impact assessment did not involve re-performing the MNGP IPEEE SMA. Similarly, SMA plant walkdowns were not re-performed in support of this assessment.

EPU equipment replacements are judged to be installed using anchorages that are similar to the existing equipment anchorages. Based on this assessment, it is concluded that no unique or significant impacts on seismic risk result from the EPU.

4.5 OTHER EXTERNAL EVENTS RISK

In addition to internal fires and seismic events, the MNGP IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Other External Hazards

The MNGP IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that MNGP meets the applicable NRC Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards.

Note that internal flooding scenarios are analyzed as internal events and already are included in the MGNP internal events at-power PRA used in this EPU risk assessment.

4.6 SHUTDOWN RISK

The impact of the Extended Power Uprate (EPU) on shutdown risk is similar to the impact on the at-power Level 1 PRA. Based on the insights of the at-power PRA impact assessment, the areas of review appropriate to shutdown risk are the following:

- Initiating Events
- Success Criteria
- Human Reliability Analysis

The following qualitative discussion applies to the shutdown conditions of Hot Shutdown (Mode 3), Cold Shutdown (Mode 4), and Refueling (Mode 5). The EPU risk impact during the transitional periods such as at-power (Mode 1) to Hot Shutdown and Startup (Mode 2) to at-power are judged to be subsumed by the at-power Level 1 PRA. This is consistent with the U.S. PRA industry, and with NRC Regulatory Guide 1.174 which states that not all aspects of risk need to be addressed for every application. While higher conditional risk states may be postulated during these transition periods, the short time frames involved produce an insignificant impact on the long-term annualized plant risk profile.

4.6.1 Shutdown Initiating Events

Shutdown initiating events include the following major categories:

- Loss of RCS Inventory
  - Inadvertent Draindown
  - LOCAs
- Loss of Decay Heat Removal (includes LOOP)

No new initiating events or increased potential for initiating events during shutdown (e.g., loss of DHR train) can be postulated due to the 113% EPU.

4.6.2 Shutdown Success Criteria

The impact of the EPU on the success criteria during shutdown is similar to the Level 1 PRA. The increased power level decreases the time to boildown. However, because the reactor is already shutdown, the boildown times are much longer compared to the at-power PRA. Further discussion regarding boil down times is provided in Section 4.6.3 in the discussion of the impacts on shutdown operator action response times.

The increased decay heat loads associated with the EPU impacts the time when low capacity decay heat removal (DHR) systems can be considered successful alternate DHR systems. The EPU condition delays the time after shutdown when low capacity DHR systems may be used as an alternative to Shutdown Cooling (SDC). However, shutdown risk is dominated during the early time frame soon after shutdown when the decay heat level is high and, in this time frame, low capacity DHR alternatives are already not viable DHR systems.

Other success criteria are marginally impacted by the EPU. The EPU has a minor impact on shutdown RPV inventory makeup during loss of decay heat removal scenarios in shutdown because of the low decay heat level. The heat load to the suppression pool during loss of decay heat removal scenarios in shutdown (i.e., during shutdown phases with the RPV intact) is also lower because of the low decay heat level such that the margins for suppression pool cooling capacity are adequate for the EPU condition.

The EPU impact on the success criteria for blowdown loads, RPV overpressure margin, and SRV actuation is estimated to be negligible because of the low RPV pressure and low decay heat level during shutdown.

4.6.3 Shutdown HRA Impact

Similar to the at-power Level 1 PRA, the decreased boildown time due to the EPU decreases the time available for operator actions. The significant, time critical operator actions impacted in the at-power Level 1 PRA are related to RPV depressurization, SLC injection, and SLC level control. These operator actions do not directly apply to shutdown conditions because the RPV is at low pressure and the reactor is subcritical. The risk significant operator actions during shutdown conditions include recovering a failed DHR system or initiating alternate DHR systems. However, the longer boildown

times during shutdown results in the EPU having a minor impact on the shutdown HEPs associated with recovering or initiating DHR systems.

The calculations in Appendix B of this assessment show that the times available to perform loss of decay heat removal response actions during shutdown is many hours. The reductions in these times due to the EPU are shown in Appendix B to be in the range of 10 to 15% (depending on time after shutdown and water level configuration). Such small changes in already lengthy operator action response times result in negligible changes in human error probabilities.

4.6.4 Shutdown Risk Summary

Based on a review of the potential impacts on initiating events, success criteria, and HRA, the 113% EPU is assessed to have a non-significant impact (delta CDF of roughly 2% per calculations in Appendix B) on shutdown risk.

This assessment is consistent with GE's generic conclusions on this issue [15]:

> *"The shutdown risks for BWR plants are generally low and the impact of CPPU on the CDF and LERF during shutdown is expected to be negligible."*

4.7 RADIONUCLIDE RELEASE (LEVEL 2 PRA)

The Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. In the process of modeling severe accidents (i.e., the MAAP code), the complex plant structure has been reduced to a simplified mathematical model which uses basic thermal hydraulic principles and experimentally derived correlations to calculate the radionuclide release timing and magnitude. [9] Changes in plant response due to EPU

represent relatively small changes to the overall challenge to containment under severe accident conditions.

The following aspects of the Level 2 analysis are briefly discussed:

- Level 1 input
- Accident Progression
- Human Reliability Analysis
- Success Criteria
- Containment Capability
- Radionuclide Release Magnitude and Timing

Level 1 Input

The front-end evaluation (Level 1) involves the assessment of those scenarios that could lead to core damage. The subsequent treatment of mitigative actions and the inter-relationship with the containment after core damage is then treated in the Containment Event Tree (Level 2).

In the Monticello Level 1 PRA, accident sequences are postulated that lead to core damage and potentially challenge containment. The Monticello Level 1 PRA has identified discrete accident sequences that contribute to the core damage frequency and represent the spectrum of possible challenges to containment.

The Level 1 core damage sequences are also directly propagated through the Level 2 PRA containment event trees. Changes to the Level 1 PRA modeling directly impact the Level 2 PRA results. However, the percentage increase in total CDF due to the EPU is not a direct translation to the percentage increase in total LERF. For example, a change to loss of decay heat removal or long-term SBO core damage accidents would not impact the LERF results, as such accidents do not result in Level 2 LERF sequences.

Therefore, the Level 2 at-power internal events PRA model is also requantified as part of this EPU risk assessment.

Accident Progression

As discussed earlier in Section 4.1.3, the EPU does not change the plant configuration and operation in a manner that produces new accident sequences or changes accident sequence progression phenomenon. This is particularly true in the case of the Level 2 post-core damage accident progression phenomena. The minor changes in decay heat levels and system configurations of the EPU will not impact significantly quantification and modeling of post-core damage accident progression.

Therefore, no changes are made as part of this assessment to the Level 2 models (either in structure or basic event phenomenon probabilities) with respect to accident progression modeling.

Human Reliability Analysis

Risk significant Level 2 operator actions are, in general, conditional repair and recovery actions given that the operator failed in the Level 1 time frame (e.g., failure to depressurize the RPV in Level 2 PRA given failure to depressurize in Level 1 PRA). Any changes in the conditional HEPs due to the power uprate (based on reduced time available) are judged to be small and would have a minor impact on the Level 2 quantification results.

Success Criteria

No changes in success criteria have been identified with regard to the Level 2 containment evaluation. The slight changes in accident progression timing and decay heat load has a minor or negligible impact on Level 2 PRA safety functions, such as containment isolation, ex-vessel debris coolability and challenges to the ultimate

containment strength. (Refer to Section 4.1.2.8 of this report). Therefore, no changes to Level 2 modeling with respect to success criteria are made as part of this analysis.

Containment Capability

As discussed in Section 4.1.7 earlier in this report, no issues have been identified with respect to the EPU that have any impact on the capacity of the MNGP containment as analyzed in the PRA.

The MNGP containment capacity with respect to severe accidents is analyzed in the PRA using plant specific structural analyses as well as information from industry studies and experiments. The MNGP containment capacity is assessed in the Level 2 with respect to following challenge categories [9]:

1) Pressure Induced Containment Challenge: Containment pressures may increase from normal operating pressure along a saturation curve to very high pressures (i.e., beyond 100 psi), during accidents involving:

   - Insufficient long term decay heat removal; and
   - Inadequate reactivity control and consequential inadequate containment heat removal.

2) Temperature Induced Containment Challenge: Containment temperatures can rise without substantial pressure increases if containment pressure control measures (e.g., venting) are available. In such cases, containment temperature may increase to above 1000°F with the containment at less than design pressure during accidents involving core melt progression.

3) Combined Pressure and Temperature Induced Containment Challenge: Containment pressures and temperatures can both rise during a severe accident due to molten debris effects following RPV failure and subsequent core concrete interaction. For instance:

   - Containment temperatures can rise from approximately 300°F at core melt initiation to above 1000°F in time frames on the order of 10 hours.

- Additionally, containment pressure can rise due to non-condensible gas generation and RPV blowdown in the range of 40 psig to 100 psig over this same time frame.

4) Containment Dynamic Loading: Postulated accident sequences cover a broad spectrum of events, including failure of the containment under degraded conditions for which the following may be present:

   - High suppression pool temperature with substantial continuous blowdown occurring (i.e., equivalent to greater than 6% power),

     or

   - High suppression pool water levels coupled with equivalent LOCA loads and the consequential hydrodynamic loads, or
   - Other energetic events, such as steam explosion.

5) Containment Isolation: Containment isolation failure during a core damage event is modeled as leading to large early releases in the MNGP Level 2.

The minor changes to the plant from the EPU have no impact on the definition of these containment loading profiles or the likelihood of containment isolation failure. The slightly higher decay heat levels associated with the EPU will result in minor reductions in times to reach loading challenges; however, the time frames are long (many hours) and the accident timing reductions of 10-15% due to the EPU have an insignificant impact on the Level 2 results.

For example, MAAP cases MNGPEPU9 and 9x (refer to Appendix E of this report) performed in support of this analysis shows that the time to reach the primary containment ultimate failure pressure (as assessed in the MNGP PRA) for a loss of all decay heat removal sequence is over 40 hours both for the CLTP condition and the EPU condition. Changes in such long time frames due to the EPU have no quantifiable impact on the Level 2 results.

Release Magnitude and Timing

The following issues can substantially increase or decrease the ability to retain fission products or mitigate their release:

- Radionuclide removal processes
- Containment failure modes
- Phenomenology
- Accident sequence timings

Each of these issues is considered and analyzed in the MNGP Level 2 PRA. [9]

The "Early" timing threshold is defined in the MNGP PRA as a release from secondary containment beginning at 0 to 6 hours after declaration of a General Emergency. The 0-6 hour time frame is based upon experience data concerning non-nuclear offsite accident response and is conservatively (i.e., 0-4 hours is a justifiable "Early" range) assumed to include cases in which minimal offsite protection measures have been performed.

The "Large" magnitude threshold is defined in the MNGP Level 2 PRA as greater than 10% release of CsI inventory in the core. This is based on past industry studies that show once the average release fraction of CsI falls below approximately 0.1, the mean number of prompt fatalities is very small, or zero, except for a few outliers that correspond to pessimistic assumptions.

This release categorization and bases is consistent with U.S. BWR PRA industry techniques. [4, 21]

No modeling or success criteria changes are required in the post core damage Level 2 sequences due to the EPU. The Level 2 functions are either conservatively based or are driven by accident phenomena. Refer to Table 4.1-10.

The MNGP plant changes for the EPU have no impact on the usage and appropriateness of this release categorization scheme. As discussed earlier, fission product inventory in the reactor core is higher as a result of the increase in power due to the EPU. The increase in fission product inventory results in an increase in the total radioactivity available for release given a severe accident. However, this does not impact the definition or quantification of the LERF risk measure used in Regulatory Guide 1.174, and as the basis for this risk assessment. The MNGP PRA release categories are defined based on the percentage (as a function of EOC inventories) of CsI released to the environment, which is consistent with most, if not all, industry PRAs.

The following release categorizations were considered for possible changes to LERF due to the EPU:

- Medium-Late
- Medium-Early
- Large-Late

It can be postulated that the EPU could result in impacts on both the magnitude and timing of Medium-Late Level 2 PRA release sequences such that they become LERF sequences. Review of theses sequences in the MNGP Level 2 PRA shows that all Medium-Late release sequences are long term release sequences with no potential to drop into the Early release category due to the EPU. The Medium-Late sequences in the MNGP Level 2 PRA begin to release in time frames greater than t=15 hrs, and in many cases greater than t=30 hrs. The 113% EPU does not reduce such sequence timings in a manner that would make them Early releases. As such, the MNGP EPU

does not cause Medium-Late sequences to become LERF. No MAAP runs are necessary for this assessment.

It can be postulated that the EPU could increase the magnitude of the release for Medium-Early Level 2 PRA release sequences such that they become LERF sequences. MAAP runs MNGPEPU10c, MNGPEPU10cx, MNGPEPU10d and MNGPEPU10dx (refer to Appendix E) were performed to investigate if such a change would occur due to the EPU. These runs are for two typical Medium-Early scenarios. The results show that the CsI release percentage increases one to two percentage points for the EPU, but the magnitudes are still in the Medium category. As such, the MNGP EPU does not cause Medium-Early sequences to become LERF.

Similarly, it can be postulated that the EPU could decrease the timing of the release for Large-Late Level 2 PRA release sequences such that they become LERF sequences. MAAP runs MNGPEPU10a, MNGPEPU10ax, MNGPEPU10b and MNGPEPU10bx (refer to Appendix E) were performed to investigate if such a change would occur due to the EPU. Most Large-Late sequences release tens of hours after accident initiation, with no potential to become LERF sequences due to the EPU. However, there are a few Large-Late sequences in the MNGP Level 2 that begin releasing close to the Early time frame threshold. These MAAP runs are for the two fastest progressing Large-Late scenarios in the MNGP Level 2 PRA. The results show that the CsI release timings are reduced but not sufficiently to warrant their classification as LERF sequences. As such, the MNGP EPU does not cause Large-Late sequences to become LERF.

Level 2 Impact Summary

Based on the above discussion, the impact of the EPU on the MNGP Level 2 PRA results, independent of the Level 1 analysis, is judged to be minor. The only change in the Level 2 is due to changes in the Level 1 cutset frequencies (due to the HEP changes discussed in Section 4.1.6) used as input to the Level 2 quantification.

Section 5
CONCLUSIONS

The Extended Power Uprate (EPU) for Monticello has been reviewed to determine the net impact on the risk profile associated with Monticello operation at an increase in power level to 2004 MWt. This examination involved the identification and review of plant and procedural changes, plus changes to the risk spectrum due to changes in the plant response.

The change in plant response, procedures, hardware, and setpoints associated with the increase in power have been investigated using the 2005 Monticello PRA ***average maintenance*** model (fault tree ***Risk-T&M.caf***); the 1995 MNGP IPEEE study for seismic, internal fires and other external events; and a qualitative evaluation of shutdown events.

This section summarizes the risk impacts of the EPU implementation on the following areas:

- Level 1 Internal Events PRA
- Fire Induced Risk
- Seismic Induced Risk
- Internal Flooding Risk
- Shutdown Risk
- Level 2 PRA

The review has indicated that small perturbations on individual inputs could be identified.

Guidelines from the NRC (Regulatory Guide 1.174) are followed to assess the change in risk as characterized by core damage frequency (CDF) and Large Early Release Frequency (LERF)

5.1 LEVEL 1 PRA

Qualitative engineering insights regarding the adequacy of procedures and systems to prevent postulated core damage scenarios are among the principal results of the Level 1 portion of the PRA. These insights deal with the adequacy of, or improvements to, Monticello procedures or systems (frontline or support) to accomplish their safety mission of preventing core damage. The severe accident scenarios that have been identified in the Level 1 PRA have been reviewed and the relatively small perturbations due to power uprate do not affect the scenario development or the qualitative insights.

Table 5.1-1 provides a summary of the PRA model changes incorporated as a result of the power uprate evaluation. Table 5.1-1 provides the following information:

- Basic event identification and description
- Basic event probability in the current model
- Revised probability for EPU

Two modeling structure changes to the MNGP PRA were necessary to reflect the EPU. The first is the change to the SRV fault tree logic for RPV overpressure protection during an ATWS. The second modeling structure change was made to require the RPV to be at a depressurized state during transient and SORV scenarios to allow success of nominal flow CRD as the sole early injection source.

The results of the Level 1 PRA quantification for the MNGP EPU condition are summarized in Table 5.1-2 along side the CLTP MNGP PRA results as a function of initiating event type. The EPU is estimated to increase the Monticello internal events PRA CDF from the base value of 7.32E-6/yr to 7.89E-6/yr, an increase of 5.67E-7

(7.8%). As can be seen from Table 5.1-2, the distribution of the EPU results remains virtually unchanged with respect to the base MNGP PRA.

5.2 FIRE INDUCED RISK

Based on the results of the internal events PRA evaluation for a 113% power uprate and a review of the MNGP IPEEE, it is concluded that the effects on any increase in risk contribution associated with fire induced sequences is minor, estimated at a 2-3% increase in fire CDF (refer to Section 4.3 of this report).

5.3 SEISMIC RISK

Based on a review of the Monticello IPEEE, the conclusions of the MNGP seismic margins assessment (SMA) are judged to be unaffected by the EPU. The power uprate has little or no impact on the seismic qualifications of the systems, structures and components (SSCs). Specifically, the power uprate results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event, are judged not to alter the results of the SMA. Refer to Section 4.4 of this report for further discussion.

5.4 OTHER EXTERNAL HAZARDS

Based on review of the Monticello IPEEE, the power uprate has no significant impact on the plant risk profile associated with tornadoes, external floods, transportation accidents, and other external hazards. Refer to Section 4.5 of this report for further discussion.

Table 5.1-1

MNGP PRA MODEL CHANGES TO RELECT EPU

Change	Parameter ID	Model Element Description	MNGP CLTP PRA Value[1]	EPU Value
Human Error Probability (HEP) Changes to address reduced timings	CRD-LSBYPY	Fail to restore CRDH after LOSP and ECCS load shed	8.00E-02	1.23E-01[2]
	CRD-PUMP-Y	Fail to start second CRDH pump from control room	9.00E-03	1.40E-02[2]
	CRD-VALV-Y	Fail to maximize CRDH flow - valves in RB	4.00E-02	5.27E-02[2]
	DEP-02MN-Y	Fail RPV depressurization within 2 minutes	2.50E-01	5.10E-01[2]
	DEP-12MN-Y	Fail RPV depressurization within 12 minutes	5.20E-03	9.84E-03[2]
	DEP-50MN-Y	Fail RPV depressurization within 50 minutes	1.80E-04	1.90E-04[2]
	FW-CNTRL-Y	Fail to control FW as high pressure injection source following transient	4.60E-03	5.46E-03[2]
	FW-REFLG-Y	Fail to identify reference leg leak	4.00E-02	6.94E-02[2]
	LEVEL-05-Y	Fail to detect need for injection within 5 minutes of compelling signal	5.00E-02	1.00E+00[2]
	LEVEL-25-Y	Fail to detect need for injection within 25 minutes of compelling signal	6.00E-04	1.72E-03[2]
	RHR-DHR-AY	Fail to align RHR for CHR - ATWS	1.40E-02	2.19E-02[2]
	SD-NOTRIPY	Fail to prevent turbine trip while shutting down	2.00E-01	2.27E-01[2]
	SLC-INI-SY	Fail to initiate SLC - short time available	4.40E-03	6.17E-03[2]
	SLC-LVL1-Y	Fail to control reactor level (fail SLC), given nominal conditions	1.00E-02	1.53E-02[2]
	SLC-LVL2-Y	Fail to control reactor level (fail SLC), given challenging conditions	1.30E-02	1.97E-02[2]
	VENT-CHR-Y	Fail to align containment venting as means of CHR	3.10E-05	3.68E-05[2]
SORV Probability	XVRONESRVC	SRV fails to reclose as pressure drops (Transient)	2.00E-03	2.26E-03[3]
	XVR-ATWS-C	SRV fails to reclose as pressure drops (ATWS)	2.00E-02	2.26E-02[3]

Table 5.1-1

MNGP PRA MODEL CHANGES TO RELECT EPU

Change	Parameter ID	Model Element Description	MNGP CLTP PRA Value[1]	EPU Value
AC Recovery Failure Probabilities	REC-EDG-12/11	Fail to recover EDG within 12 hours, given failure to recover w/i 11 hours	9.3E-01	1.0E+00[2]
	REC-EDG-16/12	Fail to recover EDG within 16 hours, given failure to recover w/i 12 hours	9.0E-01	8.5E-01[2]
	REC-EDG-22/12	Fail to recover EDG within 22 hours, given failure to recover w/i 12 hours	7.3E-01	6.5E-01[2]
	REC-EDG-3/50	Fail to recover EDG within 3 hours, given failure to recover w/i 50 minutes	6.9E-01	6.6E-01[2]
	REC-EDG-50/30	Fail to recover EDG within 50 minutes, given failure to recover w/i 30 minutes	9.1E-01	9.4E-01[2]
	REC-OSP-12/11	Fail to recover OSP within 12 hours, given failure to recover w/i 11 hours	9.2E-01	1.0E+00[2]
	REC-OSP-16/12	Fail to recover OSP within 16 hours, given failure to recover w/i 12 hours	8.0E-01	7.3E-01[2]
	REC-OSP-22/12	Fail to recover OSP within 22 hours, given failure to recover w/i 12 hours	5.0E-01	4.5E-01[2]
	REC-OSP-3/50	Fail to recover OSP within 3 hours, given failure to recover w/i 50 minutes	4.3E-01	4.1E-01[2]
	REC-OSP-50/30	Fail to recover OSP within 50 minutes, given failure to recover w/i 30 minutes	8.5E-01	9.0E-01[2]
CRDH Nominal Flow for Early Injection Requires RPV Low Pressure	Fault Tree Gate J018	<Fault tree gate DEP-50 added as an input to CRDH Early "OR" gate J018>	n/a	n/a
RPV Overpressure Protection for ATWS	Fault Tree Gate X028	<Fault tree gate X028 revised to model 2 or more random failures of SRVs to open during an ATWS>	n/a	n/a
	XVR8SRVCCN38	3 SRVs Fail to Open (Common Cause Failure)	2.03E-06	1.16E-05[4]

Notes to Table 5.1-1:

(1) The following minor basic event changes were made to the MNGP 2005 PRA model of record to prepare its use as the CLTP reference model for this analysis:

- CRIT-DET-Y value revised from 3.00E-05 to 1.18E-04 (base PRA mistakenly used 40 min. diagnosis time as basis for human error probability instead of correct 30 min.). The 1.18E-04 base CLTP value is not changed by the EPU.
- DEP-02MN-Y value revised from 1.00E-01 to 2.50E-01 (base PRA mistakenly used 3 min. diagnosis time as basis for human error probability instead of correct 2 min.).
- SD-NOTRIPY value revised from 3.00E-01 to 2.00E-01 (base PRA mistakenly selected 0.3 off time reliability curve instead of correct 0.2 value).
- REC-EDG-22/12 value revised from 0.63 to 0.73 (base PRA mistakenly calculated the conditional recovery probability of 0.63 instead of the correct value of 0.73).
- MPRE-EXIST-LKG, "Pre-Existing Primary Containment Leakage (20La)" added to the containment isolation fault tree at gate BREACH. This event represents the likelihood of a pre-existing containment leak at t=0. This event was added to support the containment overpressure sensitivity.

These minor changes do not result in a significant change in the quantified risk result of the MNGP base PRA.

(2) Refer to Table 4.1-11.

(3) Refer to Section 4.1.2.6.

(4) Basic event XVR8SRVCCN38, "3 SRVs fail to Open (Common Cause Failure)", revised to a probability of 1.16E-05 to reflect that EPU requires this event to be 2 SRVs must fail to open to fail RPV initial overpressure protection during an ATWS (refer to Section 4.1.2.5). This probability is calculated using the random failure rate used in the MNGP PRA for an SRV failing to open (1.16E-4/demand) and the BETA common cause failure model with a 0.1 β factor.

Table 5.1-2

MNGP CLTP CDF VS EPU CDF AS A FUNCTION OF INITIATING EVENT TYPE

Initiating Event Type	Percentage of CDF	
	MNGP CLTP	EPU
Internal Floods	89.8%	87.1%
Turbine Trip	3.2%	4.0%
Manual Shutdown	2.3%	2.8%
LOCAs Inside Containment	2.1%	2.7%
Loss of Instrument Air	1.2%	1.3%
Other Transients	1.1%	1.9%
LOOP	0.3%	0.3%
Loss of AC or DC Bus	0.1%	0.1%
TOTAL CDF:	7.32E-06	7.89E-06

5.5 SHUTDOWN RISK

The impact of the Extended Power Uprate (EPU) on shutdown risk is similar to the impact on the at-power Level 1 PRA. Shutdown risk is affected by the increase in decay heat power. However, the lower power operating conditions during shutdown (e.g., lower decay heat level, lower RPV pressure) allow for additional margin for mitigation systems and operator actions. Based on a review of the potential impacts on initiating events, success criteria, and HRA, the EPU implementation is judged to have a minor impact (delta CDF ~2%) on shutdown risk. Refer to Section 4.6 and Appendix B of this report for further discussion.

5.6 LEVEL 2 PRA

The Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. The EPU change in power represents a relatively small change to the overall challenge to containment under severe accident conditions.

The EPU is estimated to increase the Monticello at-power internal events LERF from the base value of 3.64E-7/yr to 3.94E-7/yr, an increase of 3.00E-8/yr (8.2%).

5.7 QUANTITATIVE BOUNDS ON RISK CHANGE

5.7.1 Sensitivity Studies

As discussed in the previous sections, the best estimate change in the MNGP risk profile due to the EPU is a 7.8% increase in CDF and an 8.2% increase in LERF. One of the methods to provide valuable input into the decision-making process is to perform sensitivity calculations for situations with different assumed conditions to bound the results.

These sensitivity studies investigate the impact on the at-power internal events CDF and LERF. As the change in CDF and LERF is minor, only conservative sensitivity cases (i.e., those that will increase the calculated risk increases) are analyzed here. Nine (9) quantitative sensitivity cases are performed and discussed below.

Sensitivity #1

This sensitivity increases the Turbine Trip transient initiator frequency to bound the various changes to the BOP side of the plant (e.g., main turbine modifications). The revision to the Turbine Trip frequency using an approach that assumes an additional turbine trip is experienced in the first year following start-up in the EPU condition and an additional 0.5 event in the second year. The change in the long-term average of the Turbine Trip (IE_TURB-TRIP) frequency is calculated as follows for this sensitivity case:

- Base long-term Turbine Trip frequency is 9.90E-1/yr
- 10 years is used as the "long-term" data period
- End of 10 years does not reach the end-of-life portion of the bathtub curve
- Revised Turbine Trip frequency for this sensitivity case is calculated as:

$$\frac{(10 \times 0.99) + 1.0 + 0.5}{10} = 1.14/\text{yr}$$

All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #2

This sensitivity case conservatively assumes that the potential impact on transient initiator frequencies is manifested in the MSIV Closure initiator frequency and not the Turbine Trip frequency. The MNGP base MSIV Closure initiator frequency (IE_MSIV) of 3.80E-2 is revised in this sensitivity case in the same manner as that discussed in Sensitivity Case #1:

$$\frac{(10 \times 3.80E\text{-}2) + 1 + 0.5}{10} = 1.88E\text{-}1/yr$$

All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #3

The EPU base quantification does not modify the DBA LOCA frequency. Acknowledging that the increased flow rates of the EPU can result in increased piping erosion/corrosion rates, this sensitivity case conservatively doubles the Large LOCA initiator (IE_LLOCA) frequency. All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #4

This sensitivity case combines the changes of Sensitivity Case #1 with the changes of Sensitivity Case #3. All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #5

This sensitivity case combines the changes of Sensitivity Case #2 with the changes of Sensitivity Case #3. All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #6

This sensitivity case conservatively assumes aligning a second CRDH pump with no enhanced valve positioning is not successful as an early injection source. This sensitivity is made by removing the fault tree gate J018, "1 OF 2 CRDH PUMPS NOT AVAILABLE (NOMINAL FLOW OPTION)", as an input to gate J017, "CRDH FLOW NOT SUFFICIENT FOR EARLY INJECTION (two pumps or enhanced flow path)".

All other parameters are the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #7

This sensitivity case combines the changes of Sensitivity Case #4 with the changes of Sensitivity Case #6. All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

Sensitivity #8

This sensitivity case investigates the impact of EPU containment overpressure credit on low pressure ECCS NPSH determination during DBA accidents. This sensitivity study is discussed in detail in Appendix F of this risk assessment.

The results of the analysis in Appendix F are summarized in Table 5.7-1.

Sensitivity #9

This sensitivity case combines the changes of Sensitivity Case #7 with the changes of Sensitivity Case #8. All other parameters are maintained the same as the EPU base case. The model changes made for this sensitivity case are summarized in Table 5.7-1.

5.7.1.2 Sensitivity Results

The results of the nine (9) sensitivity cases performed in support of this risk assessment are summarized in Table 5.7-1.

5.7.2 Results Summary

The key result of the PRA evaluation is the following:

> Minor risk increases were calculated for both CDF and LERF. The risk increase is primarily associated with reduced times available for certain operator actions.

The best estimate of the risk increase for at-power internal events due to the EPU is a delta CDF of 5.67E-7 (an increase of 7.8% over the base CLTP CDF of 7.32E-6/yr). The best estimate at-power internal events LERF increase due to the EPU is a delta LERF of 3.00E-8 (an increase of 8.2% over the base CLTP LERF of 3.64E-7/yr).

Using the NRC guidelines established in Regulatory Guide 1.174 and the calculated results from the Level 1 and 2 PRA, the best estimate for the CDF risk increase (5.67E-7/yr) and the best estimate for the LERF increase (3.00E-8/yr) are both within Region III (i.e., changes that represent very small risk changes).

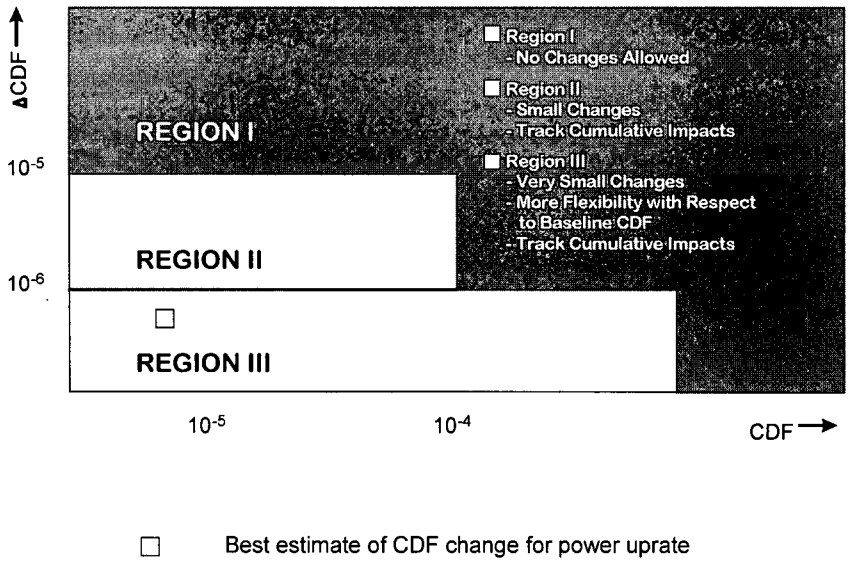
The quantitative sensitivity cases performed in this analysis show that both the delta CDF and the delta LERF remain within Region III (refer to Figures 5.7-1 and 5.7-2)

Based on these results, the proposed MNGP 113% Extended Power Uprate is acceptable on a risk basis.

Table 5.7-1

RESULTS OF MNGP EPU PRA SENSITIVITY CASES

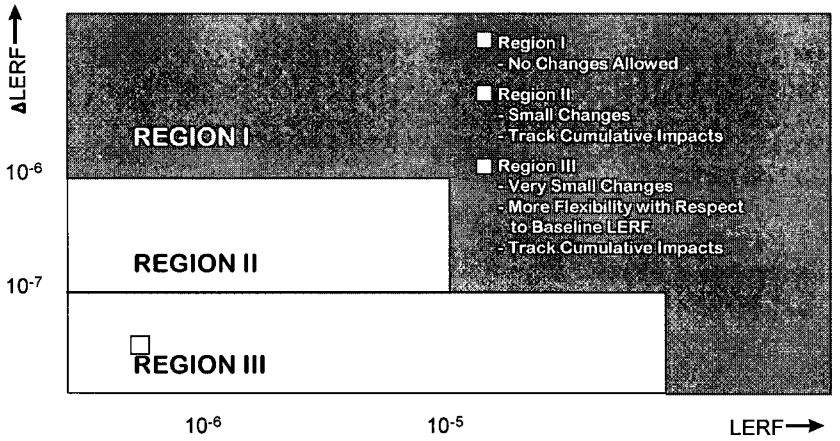
Parameter ID	MNGP CLTP PRA	EPU Base PRA	Sensitivity Case #1	Sensitivity Case #2	Sensitivity Case #3	Sensitivity Case #4	Sensitivity Case #5	Sensitivity Case #6	Sensitivity Case #7	Sensitivity Case #8	Sensitivity Case #9
Post-Initiator HEPs	Base CLTP Values	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)
Turbine Trip IE	Base CLTP (9.90E-1)	Base CLTP (9.90E-1)	1.14	Base CLTP Value	Base CLTP Value	1.14	Base CLTP Value	Base CLTP Value	1.14	Base CLTP Value	1.14
MSIV Closure IE	Base CLTP (3.80E-2)	Base CLTP (3.80E-2)	Base CLTP Value	1.88E-01	Base CLTP Value	Base CLTP Value	1.88E-01	Base CLTP Value	Base CLTP Value	Base CLTP Value	Base CLTP Value
LLOCA IE	Base CLTP (1.64E-4)	Base CLTP (1.64E-4)	Base CLTP Value	Base CLTP Value	3.28E-4	3.28E-4	3.28E-4	Base CLTP Value	3.28E-4	Base CLTP Value	3.28E-4
Nominal CRDH for Early Injection	Base CLTP (Yes)	Base EPU (Yes, but LP RPV)	Base EPU	Base EPU	Base EPU	Base EPU	Base EPU	No CRDH Nominal Early	No CRDH Nominal Early	Base EPU	No CRDH Nominal Early
LP ECCS NPSH Impact from DBA COP	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Yes (App. F)	Yes (App. F)
CDF:	7.32E-06	7.89E-06	7.94E-06	7.93E-06	7.94E-06	7.99E-06	7.98E-06	7.93E-06	8.04E-06	7.90E-06	8.05E-06
delta CDF:	-	**5.67E-07**	6.15E-07	6.05E-07	6.22E-07	6.69E-07	6.60E-07	6.13E-07	7.17E-07	5.76E-07	7.26E-07
LERF:	3.64E-07	3.94E-07	4.07E-07	4.06E-07	3.97E-07	4.10E-07	4.09E-07	3.94E-07	4.10E-07	4.03E-07	4.19E-07
delta LERF:	-	**3.00E-08**	4.31E-08	4.24E-08	3.30E-08	4.62E-08	4.53E-08	3.03E-08	4.65E-08	3.90E-08	5.55E-08



□ Best estimate of CDF change for power uprate

Figure 5.7-1 MNGP EPU Risk Assessment CDF Result Versus RG 1.174 Acceptance Guidelines* for Core Damage Frequency (CDF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision-making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.



□ Best estimate of LERF change for power uprate

Figure 5.7-2 MNGP EPU Risk Assessment LERF Result Versus RG 1.174 Acceptance Guidelines* for (LERF)

* The analysis will be subject to increased technical review and management attention as indicated by the darkness of the shading of the figure. In the context of the integrated decision-making, the boundaries between regions should not be interpreted as being definitive; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

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Appendix A

MONTICELLO EPU PRA QUANTIFICATION RESULTS

Appendix A
PRA QUANTIFICATION RESULTS

The quantification runs performed for the MNGP EPU risk assessment are summarized in Table A-1. These quantifications were performed using the EPRI R&R Workstation software (i.e., the PRA software used to develop, maintain, and quantify the MNGP PRA)

Table A-1

RESULTS OF MNGP EPU PRA SENSITIVITY CASES

Parameter ID	MNGP CLTP PRA	EPU Base PRA	Sensitivity Case #1	Sensitivity Case #2	Sensitivity Case #3	Sensitivity Case #4	Sensitivity Case #5	Sensitivity Case #6	Sensitivity Case #7	Sensitivity Case #8	Sensitivity Case #9
Post-Initiator HEPs	Base CLTP Values	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)	EPU Values (Tbl 4.1-11)
Turbine Trip IE	Base CLTP (9.90E-1)	Base CLTP (9.90E-1)	1.14	Base CLTP Value	Base CLTP Value	1.14	Base CLTP Value	Base CLTP Value	1.14	Base CLTP Value	1.14
MSIV Closure IE	Base CLTP (3.80E-2)	Base CLTP (3.80E-2)	Base CLTP Value	1.88E-01	Base CLTP Value	Base CLTP Value	1.88E-01	Base CLTP Value	Base CLTP Value	Base CLTP Value	Base CLTP Value
LLOCA IE	Base CLTP (1.64E-4)	Base CLTP (1.64E-4)	Base CLTP Value	Base CLTP Value	3.28E-4	3.28E-4	3.28E-4	Base CLTP Value	3.28E-4	Base CLTP Value	3.28E-4
Nominal CRDH for Early Injection	Base CLTP (Yes)	Base EPU (Yes, but LP RPV)	Base EPU	Base EPU	Base EPU	Base EPU	Base EPU	No CRDH Nominal Early	No CRDH Nominal Early	Base EPU	No CRDH Nominal Early
LP ECCS NPSH Impact from DBA COP	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Base CLTP (No)	Yes (App. F)	Yes (App. F)
CDF:	7.32E-06	7.89E-06	7.94E-06	7.93E-06	7.94E-06	7.99E-06	7.98E-06	7.93E-06	8.04E-06	7.90E-06	8.05E-06
delta CDF:	-	**5.67E-07**	6.15E-07	6.05E-07	6.22E-07	6.69E-07	6.60E-07	6.13E-07	7.17E-07	5.76E-07	7.26E-07
LERF:	3.64E-07	3.94E-07	4.07E-07	4.06E-07	3.97E-07	4.10E-07	4.09E-07	3.94E-07	4.10E-07	4.03E-07	4.19E-07
delta LERF:	-	**3.00E-08**	4.31E-08	4.24E-08	3.30E-08	4.62E-08	4.53E-08	3.03E-08	4.65E-08	3.90E-08	5.55E-08

Appendix B

IMPACT OF EPU ON SHUTDOWN OPERATOR ACTION RESPONSE TIMES

Appendix B

IMPACT OF EPU ON SHUTDOWN OPERATOR ACTION RESPONSE TIMES

This appendix describes the thermal hydraulic analyses performed to support the assessment that the MNGP EPU has a negligible impact on human response times during plant shutdown accident scenarios.

B.1 INTRODUCTION

The risk due to accidents during shutdown is strongly dependent upon the time available from the start of the event to the onset of core damage. As time elapses after shutdown, accidents leading to boiling of coolant within the RPV and consequential inventory losses take more time to evolve. The burden on plant systems decreases as well, introducing the chance of accident mitigation with non-safety, low capacity systems.

The effect of decreasing decay heat on the times to boil and core damage is accounted for in two ways. The first is the calculation of decay heat present at a particular point in the outage. The second takes into consideration the heat capacity of the water and structures in the system available to absorb decay heat before boiling and core damage occur. Both of these aspects are addressed in this appendix to support the assessment of the relationship of decay heat levels and times available in which to perform human actions to prevent core damage during shutdown accident scenarios.

B.2 ASSUMPTIONS

The following assumptions were used in the calculation of the times to boil off the fuel coolant and reach core damage. These assumptions allow for some simplifications in the calculation, and also allow for an appropriate degree of conservatism in the results.

- The time to boil and time to core damage calculations are appropriate for

conditions of RPV vented and maintained at atmospheric pressure.

- The time to core damage is conservatively estimated by calculating the time to reach 2/3 core height, and then extrapolating the time to gap release based on decay heat level ratios by assuming that gap release occurs 0.5 hours after 2/3 core height is reached one day after shutdown. Gap release is the release of fission products in the fuel pin gap, which occurs immediately after failure of the fuel cladding and is the first radiological indication of core damage. This approach is based on calculations performed by Sandia and summarized in SECY-93-190. [B-4]

- There is no heat loss from the system to the surroundings via the water surface or through the vessel walls.

- The calculation of decay heat levels and times to boiling and core damage in this assessment conservatively do not include removal of spent fuel out of the core.

- The decay heat as a function of time after shutdown is derived from a curve fit to the ASB 9-2 Branch Technical Position methodology assuming 100% initial power and 16,000 hours of power operation.

B.3 DECAY HEAT LEVEL CALCULATION

There are several methods available to calculate decay heat as a function of time after shutdown. The NRC has provided an acceptable method of calculating the decay heat rate in Branch Technical Position ASB 9-2 [B-1]. This method uses the following equation:

$$P_s = P_o \left[(1+K)(1/200) \sum_{n=1}^{11} A_n \exp(-a_n t_s) - (1/200) \sum_{n=1}^{11} A_n \exp\left[-a_n (t_s + t_o)\right] \right] \qquad \text{(B-1)}$$

Where:

P_s	=	decay heat level (MBtu/hr)
P_o	=	normal operating power (MBtu/hr)
t_s	=	time after shutdown (seconds)
t_o	=	operating history
K	=	uncertainty factor

= 0.2 for $t_s < 10^3$, 0.1 for $10^3 < t_s < 10^7$

A_n, a_n = fit coefficients as specified in Reference B-1.

Other less complex formulas have been developed and provide reasonable estimates of decay heat rates. Reference B-2 provides the simplest of these, assuming an infinite power history:

$$P_S(t) = P_O\,(0.0950)\,t_S^{-0.26} \qquad \text{(B-2)}$$

where $P_s(t)$, P_O and t_S are as defined above. A comparison of Equation B-2 to Equation B-1, assuming 16,000 hours of power operation, shows that Equation B-2 underestimates the decay heat in the first day or two by 10-20%, and it overestimates the decay heat thereafter (by 10-75%). At 70 days after shutdown, the decay heat calculated by Equation B-2 is about 75% higher than that calculated using the ASB 9-2 method [B-1].

Another abbreviated formula is found in Reference B-3. This formula, called the Wigner-Way formula, also includes a factor for the power history:

$$P_S(t) = P_O\,(0.0622)\,[t_S^{-0.2} - (t_O + t_S)^{-0.2}] \qquad \text{(B-3)}$$

As with Equation B-1, t_O is the operating history in seconds, also assumed to be 16,000 hours for comparison purposes. Equation B-3 shows a better correlation late in the outage, but the first twenty to thirty days after shutdown are under predicted (by 10-20% compared to the ASB 9-2 formula). A separate curve fit to the ASB 9-2 equation can be developed of the form:

$$P_S(t) = P_O\,(0.02561)\,t_{S(hrs)}^{-0.42371} \qquad \text{(B-4)}$$

where $t_{S(hrs)}$ is the time since shutdown in hours. This simple equation is considered to have an advantage over Equations B-2 and B-3 because it agrees with the ASB 9-2 data

to within about 10% over the full time period of interest. Although the agreement is not quite as good as the Wigner-Way formula after about 40 days, the agreement at the critical earlier times is much better. Equation B-4 is often used in industry BWR PSSAs to support boil-off timing calculations.

Using Equation B-4, the decay heat level as a function of time after shutdown is given as:

MNGP CLTP: $P_S(t) = (1775\ \text{MWt})\ (3.4118\text{E}6\ \text{Btu/hr} / 1\ \text{MWt})\ (0.02561)\ t_{S(hrs)}^{-0.42371}$

$$P_S(t) = (1.55\text{E}8)\ t_{S(hrs)}^{-0.42371}\ \text{Btu/hr} \qquad \text{(B-5a)}$$

MNGP 113% CLTP: $P_S(t) = (2004\ \text{MWt})\ (3.4118\text{E}6\ \text{Btu/hr} / 1\ \text{MWt})\ (0.02561)\ t_{S(hrs)}^{-0.42371}$

$$P_S(t) = (1.75\text{E}8)\ t_{S(hrs)}^{-0.42371}\ \text{Btu/hr} \qquad \text{(B-5b)}$$

B.4 RPV HEATUP AND BOILOFF CALCULATIONS

Once the core decay heat rate has been calculated using Equation B-5, the times to fuel coolant boiling and core damage can be calculated using simple heat transfer formulas based on the volume of water available. The principal shutdown states are represented by the following water level configurations:

- normal level
- at the flange level
- reactor cavity flooded

Nominal water volumes and associated heat capacities for use in this calculation are summarized in Table B-1.

Time to Boil

The time required for the vessel water to reach the boiling temperature (given loss of coolant decay heat removal) is represented by the following equation:

$$t_b = E_{boil} / P_s(t) \quad \text{hrs.} \qquad \text{(B-6)}$$

where:

t_b = time to boil (hours)

E_{boil} = $E_{water} + E_{struct}$

E_{water} = energy absorbed by heated water volume to reach saturation (MBtu)

E_{struct} = energy absorbed by fuel and clad (MBtu)

$P_s(t)$ = decay heat level (MBtu/hr),

and

E_{water} = $V/v * (h_{Tsat} - h_{Tinit})$

V = volume of water that heats up to the saturation temperature (ft^3)

v = specific volume of water at T_{init} (assumed constant at 0.0167 ft^3/lb_m over the temperature range of interest)

h_{Tsat} = enthalpy of water at T_{sat}, 212°F (Btu/lb_m),

h_{Tinit} = enthalpy of water at the initial RPV temperature, T_{init} (Btu/lb_m),

and

E_{struct} = $MCp_{struct} (T_{sat} - T_{init})$

MCp_{struct} = configuration specific structure heat capacity (Btu/°F - See Table B-1)

Since the specific heat of water is 1.0 Btu/lb_m°F, the difference in the enthalpies in the E_{water} expression above (h_{Tsat} - h_{Tinit}) is equivalent to the temperature difference in the E_{struct} expression (T_{sat} - T_{init}). This allows the complete expression for E_{boil} to simplify to:

$$E_{boil} = [(V/v) + MCp_{STRUCT}] * [T_{SAT} - T_{init}] \quad \text{(B-7)}$$

Substituting in the appropriate constant values, Equation B-7 can be rewritten as:

$$E_{boil} = C * [212 - T_{init}] \quad \text{(B-8)}$$

where the constant C is calculated for each of the water volumes and structure capacities given in Table B-1. Thus, with the initial temperature, T_{init} in °F and the decay heat load, $P_s(t)$ in Btu/hr, the time to reach saturation for the different configurations are given by Equations B-9 through B-13.

$$t_{b,\,2/3\ \text{core height}} = 2.02E5 * (212 - T_{init}) / P_s(t) \quad \text{hours} \quad \text{(B-9)}$$

$$t_{b,TAF} = 2.26E5 * (212 - T_{init}) / P_s(t) \quad \text{hours} \quad \text{(B-10)}$$

$$t_{b,\text{Normal Level}} = 4.85E5 * (212 - T_{init}) / P_s(t) \quad \text{hours} \quad \text{(B-11)}$$

$$t_{b,\text{Flange Level}} = 6.35E5 * (212 - T_{init}) / P_s(t) \quad \text{hours} \quad \text{(B-12)}$$

$$t_{b,\text{Cavity Flooded}} = 1.85E6 * (212 - T_{init}) / P_s(t) \quad \text{hours} \quad \text{(B-13)}$$

where $P_s(t)$ is the decay heat level (refer to Equation B-5) and T_{init} is the initial water temperature (e.g., 140F early in the outage before cavity flooded and 100F later in the outage after the cavity flooded).

Table B-1

NOMINAL WATER VOLUMES AND HEAT CAPACITIES FOR THE TIME TO BOIL AND TIME TO CORE DAMAGE CALCULATIONS

Water Level	Water Volume (ft^3)	Heat Capacity (Btu/°F) (1)	
		Water	Structure
2/3 Core Height	3374 (3)	2.06E5	(2)
Top of Active Fuel	3769 (4)	2.26E5	(2)
Normal Level	8103 (5)	4.85E5	(2)
Flange Level	10608 (6)	6.35E5	(2)
Cavity Flooded	30965 (7)	1.85E6	(2)

NOTES TO TABLE B-1:

(1) The term heat capacity is used in Eq. B-8. The water heat capacity is defined as Volume/v (where v is the specific volume of water and is assumed constant at 0.0167 ft3/lbm). Refer to text on preceding pages for further details.

(2) Structural heat capacities are conservatively not credited in this calculation.

(3) Calculated using RPV zone volumes from Reference [5]:

$= TAF_{watervolume} - 1/3\ (TAF_{watervolume} - BAF_{watervolume})$

$= 3769 - 1/3\ [\ (4733.4 - 964.0) - (2864.1 - 278.8)\ ]$

$= 3374\ ft^3$

(4) Calculated using RPV zone volumes from Reference [5]:

$= TAF_{totalvolume} - TAF_{solidvolume}$

$= 4773.4 - 964.0$

$= 3769\ ft^3$

(5) Calculated using RPV zone volumes from Reference [5]:

$= RPV_{watervolume}$ - [Water volume for Zones Q, P, N, M]

$= (\ 13303.6 - 1390.3\ ) - [\ (1305.22 + 1337.05 + 1070.18 + 206.96\ ) - (\ 0 + 79.38 + 18.72 + 11.15\ )\ ]$

$= 8103\ ft^3$

(6) Calculated using RPV zone volumes from Reference [5]:

$= RPV_{watervolume}$ - Zone Q water volume

$= (\ 13303.6 - 1390.3\ ) - [\ 1305.22 - 0\ ]$

$= 10608\ ft^3$

(7) Calculated using References [7, 8 and 11] and assuming water level is one (1) ft. below refuel floor:

$= Flange_{watervolume}$ + Reactor Cavity water volume

$= 10608 + [\ \pi\ (18\ ft)^2\ (20\ ft)\ ]$

$= 30965\ ft^3$

Time to Uncover Fuel (Boil Off) and Core Damage

The time to uncover the core due to boil off (due to loss of coolant decay heat removal) is the sum of the time required to bring the full heated water volume to saturation and the time to boil off an equivalent volume of water that lies above the core. This can be represented by an equation similar in format to the time to boil equation (Equation B-6):

$$t_{cu} = E_{total} / P_S (t) \qquad \text{(B-14)}$$

where:

t_{cu} = time to uncover the core (hours)

E_{total} = $E_{boil} + E_{boiloff}$

E_{boil} = energy absorbed to reach saturation as defined for Equation B-6 (MBtu)

$E_{boiloff}$ = energy absorbed by the water that vaporizes during boiloff (MBtu),

and

$E_{boiloff}$ = $V_b / v_{sat} * (h_{fg})$

V_b = equivalent volume of water that must vaporize for the collapsed level to reach TAF (ft^3)

v_{sat} = specific volume of water at saturation (T_{sat} = 212°F), or 0.0167 ft^3/lb_m

h_{fg} = heat of vaporization at 212°F and 14.7 psia, or 970.32 Btu/lb_m.

With constant values again assumed where appropriate, Equations B-15 through B-17 below provide the time to uncover the core for the different shutdown water level configurations:

$$t_{cu,Normal\ Level} = [4.85E5 * (212 - T_{init}) + 2.52E8] / P_S(t) \quad \text{hours} \qquad \text{(B-15)}$$

$$t_{cu,Flange\ Level} = [6.35E5 * (212 - T_{init}) + 3.97E8] / P_S(t) \quad \text{hours} \qquad \text{(B-16)}$$

$$t_{cu,Cavity\ Flooded} = [1.85E6 * (212 - T_{init}) + 1.58E9] / P_S(t) \quad \text{hours} \qquad \text{(B-17)}$$

where $P_s(t)$ is the decay heat level (refer to Equation B-5)

This analysis assumes the initial bulk water temperatures is 140F for days 0 through 5; 120F for days 6 through 10; and 100F for days 11 and beyond.

The time to uncover the core with the existing power level (CLTP) is 10.7 hours (9.5 hrs for the 113% CLTP case) at one day into the outage from the flange level configuration. The available time greatly exceeds 24 hours after a couple days into the outage when the water level is flood up into the refueling cavity.

For the impact on shutdown human error probabilities, it is necessary to know the approximate time of core damage so that this time can be used as the maximum allowable time window rather than conservatively estimating the time to reach an uncovered core. As stated in Section B.2, the time to core damage is estimated by incorporating the additional time available from boiloff from TAF down to 2/3 core height, and then extrapolating the time to gap release by assuming that gap release occurs 0.5 hours after 2/3 core height is reached one day after shutdown. The resulting equation for core damage, t_{cd}, is:

$$t_{cd} = t_{cu} + [2.3E7 + 0.5 * P_S(1d)] / P_S(t) \quad \text{hours} \qquad \text{(B-18)}$$

where:

2.3E7 represents the amount of decay heat required to boildown from TAF to 2/3 core height

$P_S(1d)$ is the decay heat 1 day after shutdown (refer to Eq. B-5)

$P_S(t)$ is the decay heat as a function of time after shutdown (refer to Eq. B-5)

This equation for estimating the time to core damage during refueling incidents is the approach typically used in U.S. industry BWR PSSAs. This equation was developed in the BWR PSSA industry to reflect BWR fuel heatup timing estimates provided in NSAC-169 and SECY-93-190. [B-4,10] SECY-93-190 reports that fuel heatup calculations performed for Grand Gulf by Sandia show that at 4 days after shutdown approximately 5 hours are available between reaching TAF and before fuel pin gap release occurs; and almost 9 hours is available at 15 days after shutdown.

Given the nature of shutdown risk, the time to core damage due to boil-off is not static but increases with increasing times after shutdown. An equation is used for ease of modeling shutdown incidents. Although one may use MAAP runs to estimate the time to core damage (as is done in the at-power PRA), it is not practical given that numerous different runs would be required for different times after shutdown.

Comparisons of the time to core damage due to boil off (given loss of coolant decay heat removal) for the normal and RPV flange water level configurations for the CLTP and the 113% CLTP cases are provided in Tables B-2 and B-3. For example, at one day into the outage from the flange level configuration, the time to core damage for the existing power level (CLTP) is 11.8 hours versus 10.5 hrs for the 113% CLTP case.

Information is not summarized for the flood-up configuration as the times to core damage are 50 hours and greater (much longer than the time frames typically considered in PRAs, and time frames at which changes in human error probabilities are negligible) after 2-3 days into the shutdown (i.e., the approximate time flood-up would have been completed).

B.5 EPU IMPACT ON SHUTDOWN RISK

Impact Due to Changes in HEPs

The primary impact of the EPU on risk during shutdown operations is the decrease in allowable operator action times in responding to off-normal events.[1] However, as can be seen from Tables B-2 and B-3, the reduction in times to core damage (i.e., 113% CLTP case compared to CLTP case) are on the order of 10-15%. Such small changes in already lengthy allowable operator response times result in negligible changes (<<1%) in calculated human error probabilities.

The allowable operator action timings to respond to loss of heat removal scenarios during shutdown operations are many hours long. Very early in an outage the times available for operator response to prevent core damage for loss of shutdown cooling events are 8-9 hours; later in an outage the times are dozens of hours. A reduction from 9 hours to 8 hours in allowable action timings would not result in any significant increase in human error probabilities for most operator actions using current human reliability analysis methods.

Decay Heat Curve Method Sensitivity Case

As a sensitivity, the timing estimates were also calculated using the decay heat curve generated by GE for the MNGP EPU. The timings from the use of this decay heat curve are very similar to the calculations based on use of the decay heat curve of Eq. B-5 and the results are the same (i.e., the changes in allowable action times are 10-15%). For example, at one day into the outage from the flange level configuration, the time to core damage for the existing power level (CLTP) is 12.6 hours versus 10.9 hrs for the 113% CLTP case (compared to 11.8 and 10.5, respectively when using the decay heat curve from Equation B-5). Like the calculations performed using the decay heat curve from Equation B-5, the available time before core damage using the GE-calculated decay heat curves for MNGP greatly exceeds 24 hours after a couple days into the outage when the water level is flood up into the refueling cavity.

[1] Another postulated impact is any changes to system success criteria during shutdown operations (specifically with respect to decay heat removal systems) that may result from the EPU. A postulated impact would be that the time into the outage at which

Impact Due to Changes in Offsite AC Recovery Failure Probabilities

In addition to traditional human error probabilities, the offsite AC recovery failure probabilities can be influenced by changes in allowable timings. An approximate calculation is performed here to estimate the impact on shutdown risk due to changes in the offsite AC recovery failure probability. The calculation is described as follows:

- A 30-day refueling outage is assumed and is divided into the following five (5) phases:
  - Day 1 of the outage
  - Day 2 of the outage
  - Day 3 of the outage
  - Days 4-28 of the outage
  - Days 29-30 of the outage
- These phases are defined to address the higher decay heat in the beginning days (1-3) of the outage, the "flooded-up" days (4-28) in the middle of the outage when decay heat issues are not the main risk contributor, and the end of the outage (29-30) when the coolant level is lowered back down into the vessel.
- The following initial water level configurations are assumed for the phases:
  - Day 1 of the outage (NORMAL RPV LEVEL)
  - Day 2 of the outage (RPV FLANGE LEVEL)
  - Day 3 of the outage (FLOODED UP)
  - Days 4-28 of the outage (FLOODED UP)
  - Days 29-30 of the outage (NORMAL)
- A review of industry BWR PSSAs (Cooper, Dresden, Fermi, Quad Cities, LaSalle, WNP-2) was performed to assist in defining the contribution of LOOP/SBO accident scenarios to the CDF of each of the above general phases. Based on the review, the CDF contribution from LOOP/SBO scenarios is high (40%-90%) in the first few days of the outage when the decay heat is higher, it drops significantly (e.g., 20%-40%) in the middle of the outage when decay heat is lower and the

backup low capacity heat removal options would be sufficient to prevent coolant boiling would be extended a number of hours. Such a postulated impact is judged to result in an insignificant change in shutdown risk (e.g., 1% or less change in shutdown CDF).

cavity is flooded (draindown events dominate these periods), and then it increases at the end of the outage when the coolant level is lowered back down into the vessel.

- The review of industry PSSAs also supported the estimation of the contributions to overall shutdown CDF during the different phases of the outage.
- Table 4-1 of NUREG/CR-6890 is used here to estimate changes in offsite AC recovery failure probabilities due to reductions in allowable timings. [B-6]
- The assessment is performed on a normalized CDF basis.

This calculation is summarized In Table B-4. As can be seen from Table B-4, the increase in shutdown CDF due to increases in AC power recovery failure probabilities due to the EPU is estimated at approximately 2%.

Summary

Based on the above discussions and calculations, the qualitative conclusion of this assessment is that the MNGP EPU has an insignificant impact on shutdown risk. The impact is approximated as roughly a 2% increase in shutdown CDF.

Table B-2

TIME TO CORE DAMAGE DUE TO BOIL OFF
(Initial Water Level: Normal Level)

Days After Shutdown	Initial Water Temperature	Time to Core Damage (hrs.)	
		CLTP	113% CLTP
1	140°F	8.2	7.3
5 (1)	140°F	16.2	14.4
10 (1)	120°F	22.3	19.9
15 (1)	100°F	27.3	24.3
20 (1)	100°F	30.8	27.5
25 (1)	100°F	33.9	30.2
30	100°F	36.6	32.6

NOTE:

(1) This list of days after shutdown is summarized to show the increasing trend of time available. Thirty days is shown here to correspond with the current industry trend toward refueling outages on the order of a month in duration. Note that the days marked with the footnote are not directly applicable to a real outage schedule for this water level configuration (i.e., the first day or two the water level will be low, but then for the majority of the outage the water level will be at the spent fuel pool level, and then will be lowered again at the end of the outage).

Table B-3

TIME TO CORE DAMAGE DUE TO BOIL OFF
(Initial Water Level: RPV Flange Level)

Days After Shutdown	Initial Water Temperature	Time to Core Damage (hrs.)	
		CLTP	113% CLTP
1	140°F	11.8	10.5
5 (1)	140°F	23.3	20.8
10 (1)	120°F	31.9	28.4
15 (1)	100°F	38.6	34.4
20 (1)	100°F	43.6	38.9
25 (1)	100°F	48.0	42.7
30	100°F	51.8	46.1

NOTE:

(1) This list of days after shutdown is summarized to show the increasing trend of time available. Thirty days is shown here to correspond with the current industry trend toward refueling outages on the order of a month in duration. Note that the days marked with the footnote are not directly applicable to a real outage schedule for this water level configuration (i.e., the first day or two the water level will be low, but then for the majority of the outage the water level will be at the spent fuel pool level, and then will be lowered again at the end of the outage).

Table B-4

ESTIMATED IMPACT ON SHUTDOWN RISK DUE TO
OFFSITE AC RECOVERY FAILURE PROBABILITY INCREASES DUE TO EPU

Outage Phase	Initial Water Level	Phase Contribution to Overall SD CDF (CLTP)[1]	LOOP/SBO Contribution to Phase CDF[1]	Time to Core Damage (hrs)		Factor Increase in Offsite AC Recovery Failure Probability[3]	Phase Contribution to Overall S/D CDF (113% CLTP)[4]
				CLTP[2]	113% CLTP[2]		
Day 1	Normal	0.10	0.75	8.2	7.3	1.12	0.109
Day 2	RPV Flange	0.10	0.50	15.8	14.1	1.11	0.106
Day 3	Flooded	0.10	0.25	65.5	58.1	negligible	0.100
Days 4-28	Flooded	0.60	0.25	131.0	116.2	negligible	0.600
Days 29-30	Normal	0.10	0.50	36.6	32.6	negligible	0.100
Normalized CDF (CLTP):		1.00	Normalized CDF (113% CLTP):				1.02

Notes to Table B-4:

(1) Approximated based on review of industry BWR PSSAs (Cooper, Dresden, Fermi, Quad Cities, LaSalle, WNP-2).

(2) Calculated using Eq. B-18.

(3) Based on use of generic offsite AC recovery failure probability information from NUREG/CR-6890. The integrated (i.e., integration of plant-centered, grid, and severe weather contributions) AC recovery failure data for shutdown conditions from Table 4-1 of NUREG/CR-6890 is used. For example, at t=8.2 hours the NUREG/CR-6890 AC recovery failure probability is 6.35E-2 and at t=7.3 hours the failure probability is 7.14E-2 (a factor of 1.12 higher).

(4) Calculated as:

$$[\,(1.0 - 4^{th}\text{ Column}) \times 3^{rd}\text{ Column}\,] + [\,4^{th}\text{ Column} \times 3^{rd}\text{ Column} \times 7^{th}\text{ Column}\,]$$

The first contribution is the non-LOOP portion of the phase CDF (i.e., the portion unaffected by changes in offsite AC recovery failure probabilities). The second contribution is the LOOP portion of the phase CDF (i.e., the portion impacted by changes in offsite AC recovery failure probabilities).

REFERENCES

[B-1] USNRC, Branch Technical Position 9-2, "Residual Decay Heat Energy for Light-Water Reactors for Long-Term Cooling."

[B-2] M.M. El-Wakil, Nuclear Heat Transport, International Textbook Company, 1971.

[B-3] K. Way, E. Wigner, "The Rate of Decay of Fission Products," (Phys. Rev., 73, 1948, pp. 1318-1330)

[B-4] USNRC, "Regulatory Approach to Shutdown and Low Power Operations," SECY-93-190, July 12, 1993, Enclosure: Draft Regulatory Analysis in Accordance with 10CFR50.109 dated February 1993.

[B-5] Monticello drawing NX-8290-168-1, Rev. A, "Reactor Primary Sys. Wts. & Volumes."

[B-6] NUREG/CR-6890, Re-Evaluation of Station Blackout at Nuclear Power Plants: 1986-2004, Volume 1, December 2005.

[B-7] Monticello drawing NF-36510, Rev. H, "Area-3 Piping Drawings Section B-B".

[B-8] Monticello drawing NF-36063, Rev. A, "Equipment Location – Reactor Building Section B-B."

[B-9] Electric Power Research Institute, Safety Assessment of BWR Risk During Shutdown Operations, NSAC-175L, Final Report, August 1992.

[B-10] Electric Power Research Institute, Analysis of BWR Fuel Heatup During a Loss of Coolant While Refueling, NSAC-169, September 1991.

[B-11] Monticello drawing NF-36507, Rev. B, "Area-3 Piping Drawings Plan Below Elev. 1027' - 8"."

Appendix C

MONTICELLO PRA QUALITY

Appendix C
MONTICELLO PRA QUALITY

The quality of the Monticello PRA models used in performing the risk assessment for the Monticello EPU is manifested by the following:

- Level of detail in PRA
- Maintenance of the PRA
- Comprehensive Critical Reviews

C.1 LEVEL OF DETAIL

The Monticello PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events.

C.1.1 Initiating Events

The Monticello at-power PRA explicitly models a large number of internal initiating events:

- General transients
- LOCAs
- Support system failures
- Internal Flooding events

The initiating events explicitly modeled in the Monticello at-power PRA are summarized in Table C-1. The number of internal initiating events modeled in the Monticello at-power PRA is similar to the majority of U.S. BWR PRAs currently in use.

Table C-1

INITIATING EVENTS FOR MONTICELLO PRA

Initiator ID	Description
IE_125VDC	Loss of both divisions of 125V DC
IE_125VDC1	Loss of division I 125V DC power
IE_125VDC2	Loss of division II 125V DC power
IE_AIR	Loss of instrument air
IE_BUS13	Loss of electrical bus 13
IE_BUS14	Loss of electrical bus 14
IE_BUS15	Loss of electrical bus 15
IE_BUS16	Loss of electrical bus 16
IE_CRDH	Loss of CRDH
IE_DW-COOL	Loss of drywell cooling
IE_FW	Loss of feedwater
IE_LLOCA	Large LOCA initiating event
IE_LOOP	Loss of offsite power initiating event
IE_MLOCA	Medium LOCA initiating event
IE_MSIV	MSIV closure
IE_RBCCW	Loss of RBCCW
IE_REFLAB	Break in both reference legs
IE_REFLEGA	Break in 2-3-2A reference leg
IE_REFLEGB	Break in 2-3-2B reference leg
IE_SHUTDOWN	Manual shutdown of reactor
IE_SLOCA	Small LOCA initiating event
IE_SORV	Relief valve spuriously fails open
IE_SW	Loss of service water
IE_TURB-TRIP	Turbine trip

Table C-1

INITIATING EVENTS FOR MONTICELLO PRA

Initiator ID	Description
IE_VACUUM	Loss of condenser vacuum
IE_XLOCA	RPV rupture
ISLOCA	Interfacing Systems LOCA (numerous unique IEs)
Breaks Outside Containment	LOCA Outside Containment (Numerous unique IEs)
Floods	Internal Flooding initiators (numerous unique IEs)

C.1.2 System Models

The Monticello at-power PRA explicitly models a large number of frontline and support systems that are credited in the accident sequence analyses. The Monticello systems are modeled in the Monticello at-power PRA using fault tree structures for the majority of the systems. The number and level of detail of plant systems modeled in the Monticello at-power PRA is consistent with industry practices.

C.1.3 Operator Actions

The Monticello at-power PRA explicitly models a large number of operator actions:

- Pre-Initiator actions
- Post-Initiator actions
- Recovery Actions

Over one hundred operator actions are explicitly modeled. Given the large number of actions modeled in the Monticello at-power internal events PRA, a summary table of the individual actions modeled is not provided here.

The human error probabilities for the actions are modeled with accepted industry HRA techniques and include input based on discussion with plant operators, trainers, and other cognizant personnel.

The number of operator actions modeled in the Monticello at-power PRA, and the approach to their quantification is consistent with industry practices.

C.1.4 Common Cause Events

The Monticello at-power PRA explicitly models a large number of common cause

component failures. Approximately two hundred common cause terms are included in the MNGP PRA. Given the large number of CCF terms modeled in the Monticello at-power internal events PRA, a summary table of them is not provided here. The number and level of detail of common cause component failures modeled in the Monticello at-power PRA is consistent with industry practices.

C.1.5 Level 2 PRA

The Monticello Level 2 links the Level 1 PRA accident sequences and systems logic with Level 2 containment event tree sequence logic and systems logic.

The following aspects of the Level 2 model reflect the more than adequate level of detail and scope:

- Dependencies from Level 1 accidents are carried forward directly into the Level 2 by transfer of sequences to ensure that their effects on Level 2 response is accurately treated.
- Virtually all phenomena identified by the NRC and industry for inclusion in BWR Mark I Level 2 analyses are treated explicitly within the model.
- The model truncation is sufficiently low to be consistent with the NEI PRA Peer Review Guidelines for Risk-Informed Applications.

C.2 MAINTENANCE OF PRA

The Monticello PRA model and documentation has been maintained living and is routinely updated to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data.

The Monticello PRA has been updated multiple times since the original IPE.

The PRA models are routinely implemented and studied by plant PRA personnel in the performance of their duties.

Formal comprehensive model reviews are discussed in Section C.3.

C.3 COMPREHENSIVE CRITICAL REVIEWS

The Monticello PRA model has benefited from the following comprehensive technical reviews:

- NEI PRA Peer Review Process
- Recent assessments against the ASME PRA Standard

NEI PRA Peer Review

The Monticello internal events PRA received a formal industry PRA Peer Review in October 1997. [C-1] The purpose of the PRA Peer Review process is to provide a method for establishing the technical quality of a PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. The PRA Peer Review process uses a team composed of PRA and system analysts, each with significant expertise in both PRA development and PRA applications. This team provides both an objective review of the PRA technical elements and a subjective assessment, based on their PRA experience, regarding the acceptability of the PRA elements. The team uses a set of checklists as a framework within which to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA products available.

The Monticello review team used the "BWROG PSA Peer Review Certification Implementation Guidelines", Revision 3, January 1997.

The general scope of the implementation of the PRA Peer Review includes review of

eleven main technical elements, using checklist tables (to cover the elements and sub-elements), for an at-power PRA including internal events, internal flooding, and containment performance, with focus on large early release frequency (LERF). The eleven technical elements are shown in Tables C-2 through C-4.

The comments from the PRA Peer Review were prioritized into four categories A-D based upon importance to the completeness of the model. All comments in Categories A and B (recommended actions and items for consideration) were identified to Monticello as priority items to be resolved in the next model update. The comments in Categories C and D (good practices and editorial) are potential enhancements and remain for consideration in future updates of the Level 1 and 2 PRA models.

All of the 'A' and 'B' priority PRA Peer Review comments have been addressed by MNGP and incorporated into the MNGP PRA model as appropriate.

Assessments Against ASME PRA Standard

Consistent with current industry practices, the MNGP has been compared against the ASME PRA Standard a number of times in recent years to identify areas of improvement.

The first assessment against the ASME PRA Standard was performed by Applied Reliability Engineering (ARE), Inc. in early 2004. That assessment compared the 2003 Monticello PRA model to the ASME Standard and NRC draft Regulatory Guide DG-1122.

Since that assessment, the MNGP PRA has evolved to include a much more extensive and detailed internal flooding analysis. Several other less significant model enhancements have occurred since the ARE, Inc. assessment, some of which were made to address insights from the assessment.

A self-assessment of the 2005 MNGP PRA against the ASME Standard was performed by NMC PRA personnel in 2006. This assessment compared the model containing the updated detailed internal flooding analysis and plant improvements to the Standard.

In anticipation of an upcoming industry peer review of the MNGP PRA, another assessment of the MNGP PRA against the ASME Standard was performed in early 2007 with the intent of determining the resources necessary to apply to the current model in order to address gaps with respect to Capability Category II of the ASME PRA Standard and RG 1.200.

C.4 PRA QUALITY SUMMARY

The quality of modeling and documentation of the Monticello PRA models has been demonstrated by the foregoing discussions on the following aspects:

- Level of detail in PRA
- Maintenance of the PRA
- Comprehensive Critical Reviews

The Monticello Level 1 and Level 2 PRAs provide the necessary and sufficient scope and level of detail to allow the calculation of CDF and LERF changes due to the Extended Power Uprate for the full power internal events challenges.

Table C-2

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Initiating Events	• Guidance Documents for Initiating Event Analysis • Groupings - Transient - LOCA - Support System/Special - ISLOCA - Break Outside Containment - Internal Floods • Subsumed Events • Data • Documentation
Accident Sequence Evaluation (Event Trees)	• Guidance on Development of Event Trees • Event Trees (Accident Scenario Evaluation) - Transients - SBO - LOCA - ATWS - Special - ISLOCA/BOC - Internal Floods • Success Criteria and Bases • Interface with EOPs/AOPs • Accident Sequence Plant Damage States • Documentation
Thermal Hydraulic Analysis	• Guidance Document • Best Estimate Calculations (e.g., MAAP) • Generic Assessments • FSAR • Room Heat Up Calculations • Documentation

Table C-2 (Continued)

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
System Analysis (Fault Trees)	• System Analysis Guidance Document(s) • System Models - Structure of models - Level of Detail - Success Criteria - Nomenclature - Data (see Data Input) - Dependencies (see Dependency Element) - Assumptions • Documentation of System Notebooks
Data Analysis	• Guidance • Component Failure Probabilities • System/Train Maintenance Unavailabilities • Common Cause Failure Probabilities • Unique Unavailabilities or Modeling Items - AC Recovery - Scram System - EDG Mission Time - Repair and Recovery Model - SORV - LOOP Given Transient - BOP Unavailability - Pipe Rupture Failure Probability • Documentation
Human Reliability Analysis	• Guidance • Pre-Initiator Human Actions - Identification - Analysis - Quantification • Post-Initiator Human Actions and Recovery - Identification - Analysis - Quantification • Dependence among Actions • Documentation

Table C-2 (Continued)

PRA PEER REVIEW TECHNICAL ELEMENTS FOR LEVEL 1

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Dependencies	• Guidance Document on Dependency Treatment • Intersystem Dependencies • Treatment of Human Interactions (see also HRA) • Treatment of Common Cause • Treatment of Spatial Dependencies • Walkdown Results • Documentation
Structural Capability	• Guidance • RPV Capability (pressure and temperature) - ATWS - Transient • Containment (pressure and temperature) • Reactor Building • Pipe Overpressurization for ISLOCA • Documentation
Quantification/Results Interpretation	• Guidance • Computer Code • Simplified Model (e.g., cutset model usage) • Dominant Sequences/Cutsets • Non-Dominant Sequences/Cutsets • Recovery Analysis • Truncation • Uncertainty • Results Summary

Table C-3

PRA CERTIFICATION TECHNICAL ELEMENTS FOR LEVEL 2

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Containment Performance Analysis	• Guidance Document • Success Criteria • L1/L2 Interface • Phenomena Considered • Important HEPs • Containment Capability Assessment • End state Definition • LERF Definition • CETs • Documentation

Table C-4

PRA CERTIFICATION TECHNICAL ELEMENTS
FOR MAINTENANCE AND UPDATE PROCESS

PRA ELEMENT	CERTIFICATION SUB-ELEMENTS
Maintenance and Update Process	• Guidance Document • Input - Monitoring and Collecting New Information • Model Control • PRA Maintenance and Update Process • Evaluation of Results • Re-evaluation of Past PRA Applications • Documentation

REFERENCES

[C-1] Monticello PRA Peer Review Certification Report, GE Document BWROG/PSA-9704, October 1997.

Appendix D

HEP ASSESSMENTS

Appendix D
HUMAN ERROR PROBABILITY (HEP) ASSESSMENTS

The Monticello risk profile, like other plants, is dependent on the operating crew actions for successful accident mitigation. The success of these actions is in turn dependent on a number of performance shaping factors. The performance shaping factor that is principally influenced by the power uprate is the time available within which to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some actions. To quantify the potential impact of this performance shaping factor, deterministic thermal hydraulic calculations using the MAAP computer code are used.

Not all operator actions in the MNGP PRA have a significant impact on the results. To minimize the resources required to requantify all operator actions in the PRA due to the EPU, a screening process was first performed to identify those operator actions that have an impact on the PRA results. This is consistent with past EPU risk assessments and is reasonable. Potential HEP changes for operator actions screened out from explicit assessment in this EPU risk assessment will not have a significant impact on the quantitative results. Given that the EPU impacts on the significant HEPs modified for this study results in increasing the plant risk profile by about 7%, the non-significant HEPs if adjusted would be expected to impact the risk profile by a fraction of a percent.

The screening process was performed against the following criteria:

6. F-V (with respect to CDF) importance measure >= 5E-3
7. RAW (with respect to CDF) importance measure >= 2.0
8. F-V (with respect to LERF) importance measure >= 5E-3
9. RAW (with respect to LERF) importance measure >= 2.0
10. Time critical (<=30 min. available) action

These criteria have been used in past EPU risk assessments. If any of the above criteria are met for an operator action, the action is maintained for explicit consideration in the EPU risk assessment. The HEP screening process is summarized in Table D-1.

As can be seen from Table D-1, thirty-eight (38) operator actions of risk importance in the PRA were identified; and an additional seven (7) time critical HEPs (i.e., less than or equal to 30 minutes available for operator action, and not necessarily risk significant) were identified.

These operator actions were then investigated for changes in allowable operator action timings using the MAAP runs performed for this analysis (refer to Appendix E). The HEPs were then recalculated using the same human reliability analysis techniques (HRA) as used in the MNGP PRA.

The changes in allowable operator action timings are not always directly linear with respect to the EPU power increase (i.e., a 13% power uprate does not always correspond to a 13% reduction in operator action timings):

- Allowable time windows for some actions are not impacted by the power uprate (e.g., timings based on battery life, timings based on internal flood rates, etc.)
- Allowable time windows for LOCAs may be driven more by the inventory loss than the decay heat.
- Allowable time windows for actions related directly to RCS boil-off time during non-LOCA events are also not necessarily linear with respect to the power uprate percentage. It is not uncommon that some actions have reductions many percentage points more than the uprate percentage. This is due to various factors, such as higher initial fuel temperature for the EPU providing more initial sensible heat to the RCS water in the early time frame after a plant trip than the CLTP condition, or more integrated fluid release out SRVs in the early time frame compared to the CLTP condition.

The HEPs for the MNGP 2005 base (CLTP) PRA and for the EPU condition are summarized in Table D-2.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
020-ISOL-M-Y	Fail to isolate a medium or large leak within 20 minutes	**8.42E-02**	1.2	**7.40E-02**	1.17	**20 min.**	PRA-CALC-04-007	
030-ISOL-M-Y	Fail to isolate a medium or large leak within 30 minutes	**3.73E-02**	**2.21**	**1.63E-02**	1.53	**30 min.**	PRA-CALC-04-007	
030-ISOL-S-Y	Fail to isolate a small leak within 30 minutes	**1.04E-01**	1.24	**2.05E-02**	1.05	**30 min.**	PRA-CALC-04-007	
060-ISOL-M-Y	Fail to isolate a medium or large leak within 60 minutes	**4.92E-02**	**17.35**	**1.17E-01**	**39.86**	60 min.	PRA-CALC-04-007	
060-ISOL-S-Y	Fail to isolate a small leak within 60 minutes	**1.62E-02**	1.52	**1.46E-02**	1.47	60 min.	PRA-CALC-04-007	
120-ISOL-S-Y	Fail to isolate a small leak within 120 minutes	**6.90E-03**	**3.29**	**8.54E-03**	**3.84**	120 min.	PRA-CALC-04-007	
ALT-INJ-LY	Fail to align FPS, RHRSW, CSW, or SW - hour available TSC support	3.98E-03	**5.98**	2.10E-03	**3.63**	n/a	II.SMR.02.008	Execution error contribution, not time-based. Diagnosis contribution treated by a separate basic event.
ALT-POWER-Y	Fail to align alternate power supplies directly to MCC-44	**3.55E-02**	**8.07**	**7.28E-03**	**2.45**	>4hrs	PRA-CALC-04-040	Timing based on battery life.
ASDS-DEP-Y	Fail to implement depressurization from ASDS panel	**1.40E-01**	**14.88**	**2.86E-02**	**3.83**	1 hr	PRA-CALC-04-015	
ATWS-SHT-Y	Operator fails to initiate ATWS (short time available)	**1.13E-02**	1.00	**1.40E-01**	1.00	**<1 min.**	II.SMR.02.008	Specific timing not listed in II.SMR.02.008, states short time available and HEP=1 assumed. Diagnosis HEP of 1.0 occurs at 1 min., per ASEP Median and Lower Bound curves.
CHR-DET–Y	Fail to identify need for containment heat removal	**1.03E-02**	**10000**	**3.61E-02**	**36100**	10 hrs	II.SMR.02.008	
CRD-LSBYPY	Fail to restore CRDH after LOSP and ECCS load shed	2.21E-04	1.00	(<5E-3)	(<2)	**25 min.**	II.SMR.02.008	Diagnosis timing stated to be 15 minutes in II.SMR.02.008. 10 minutes assumed for execution.
CRD-PUMP-Y	Fail to start second CRDH pump from control room	**6.31E-03**	1.69	1.44E-03	1.16	**25 min.**	II.SMR.02.008	
CRD-VALV-Y	Fail to maximize CRDH flow - valves in RB	**2.75E-02**	1.66	**7.32E-03**	1.18	**25 min.**	II.SMR.02.008	
CRIT-DET-Y	Fail to detect criticality issue - long time available	(<5E-3)	(<2)	7.12E-05	**3.37**	**30 min.**	II.SMR.02.008	

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
CST-FILL-Y	Fail to refill the CSTs	3.13E-03	**4.13**	3.23E-03	**4.22**	15 hrs	PRA-CALC-04-041	
DEP-02MN-Y	Fail RPV depressurization within 2 minutes	1.57E-04	1.00	(<5E-3)	(<2)	**5 min.**	II.SMR.02.008	
DEP-12MN-Y	Fail RPV depressurization within 12 minutes	**9.20E-03**	**2.76**	**6.83E-03**	**2.31**	**15 min.**	II.SMR.02.008	
DEP-50MN-Y	Fail RPV depressurization within 50 minutes	**9.38E-03**	**53.1**	1.15E-03	**7.38**	50 min.	II.SMR.02.008	
DEP-HOUR-Y	Fail RPV depressurization >an hour available	**2.60E-02**	**163.49**	**2.00E-02**	**126.18**	103 min.	II.SMR.02.008	
DEP-PD-Y	Fail to depressurize reactor after core damage, but before vessel penetration	(<5E-3)	(<2)	**3.54E-02**	1.32	2 hrs	PRA-CALC-05-008	Assumed to be same time frame as ALT-INJ-PD-Y.
DW-VENT-PRG	Fail to prevent H2 burn failing containment by vent/purge	(<5E-3)	(<2)	**3.51E-02**	1	**<30 min.**	II.SMR.02.037	
FLOODRB16Y	Fail to flood RB within 1-6 hours after torus leak	**3.23E-02**	1.08	**2.54E-02**	1.06	1-6 hrs	PRA-CALC-04-021	
FW-CNTRL-Y	Fail to control FW as high pressure injection source following transient	**8.62E-02**	**19.65**	**5.90E-02**	**13.78**	**15 min.**	II.SMR.02.008	
FW-REFLG-Y	Fail to identify reference leg leak	5.33E-04	1.01	9.41E-05	1.00	**7 min.**	II.SMR.02.008	
HPI-CSTS-Y	Fail to defeat high torus level suction transfer	2.04E-03	1.68	**8.75E-03**	**3.91**	1 hr	II.SMR.02.008	
LEVEL-05-Y	Fail to detect need for injection within 5 minutes of compelling signal	(<5E-3)	(<2)	6.15E-05	1.00	**5 min.**	II.SMR.02.008	
LEVEL-25-Y	Fail to detect need for injection within 25 minutes of compelling signal	2.43E-03	**5.05**	4.87E-05	1.08	**25 min.**	II.SMR.02.008	
LEVEL-45-Y	Fail to detect need for injection within 45 minutes of compelling signal	**3.92E-02**	**3870**	2.31E-03	**231.61**	**45 min.**	II.SMR.02.008	
L-LONG—Y	Operator fails to inject boron using SBLC - long time available	(<5E-3)	(<2)	(<5E-3)	(<2)	>1 hr	II.SMR.02.008	Turbine is online and not isolated from reactor. Many hours available to align SLC.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
OIL-LOSS-HY	Fail to identify need to address loss of fuel flow to EDG day tanks - high	**9.23E-02**	1.83	**6.94E-02**	1.62	>1 hr	PRA-CALC-05-005	
PUMPER-L-Y	Fail to provide FPS supply from fire pumper truck - hours available	2.30E-04	1.23	1.76E-03	**2.75**	6 hrs	PRA-CALC-04-042	
RCIC-MAN-Y	Fail to manually operate RCIC	**6.83E-02**	**2.30**	**9.86E-02**	**2.87**	n/a	PRA-CALC-04-039	Execution error contribution, not time-based. Diagnosis contribution treated by a separate basic event.
REC-EDG-30	Fail to recover EDG within 30 minutes	**1.55E-01**	1.03	**1.10E-01**	1.02	**30 min.**	II.SMR.02.009	Timing basis is an industry data / modeling preference - timing of this event not impacted by EPU.
REC-EDG-11/6	Fail to recover EDG within 11 hours, given failure to recover w/i 6 hours	**1.45E-01**	1.05	**1.06E-01**	1.04	11 hrs	II.SMR.02.009	Nominal 6 hr reference point is time to core damage in SBO, HPCI or RCIC running until battery failure at t=4hrs, then CD occurs at ~t=6.3 hrs. Nominal 11 hr. point is time to core damage for SBO w/extended HPCI or RCIC operation by allowing large RPV level swings, but batteries ultimately deplete in t=6-8 hrs, and CD in ~t=11hrs.
REC-EDG-16/12	Fail to recover EDG within 16 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	**1.41E-02**	1.00	16 hrs	II.SMR.02.009	Nominal 12 hr reference point is time to 200F in pool with extended HPCI/RCIC operation during SBO. Nominal 16 hr point is based on Level 2 PRA phenomena issues post RPV melt-through and leading to containment failure.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
REC-EDG-3/50	Fail to recover EDG within 3 hours, given failure to recover w/i 50 minutes	**1.55E-01**	1.07	**1.10E-01**	1.00	3 hrs	II.SMR.02.009	The nominal 50 min. reference time is the time to CD during an SBO with no injection at t=0. The nominal 3 hrs point is based on a SBO w/SORV scenario with HPCI operating but eventually runs out of steam power and CD occurs at t=3.3 hrs.
REC-EDG-50/30	Fail to recover EDG within 50 minutes, given failure to recover w/i 30 minutes	**1.55E-01**	1.02	**1.10E-01**	1.01	50 min.	II.SMR.02.009	Nominal 30 min. reference point is an industry data / modeling preference (not changed by EPU). The nominal 50 min. point is the time to CD during an SBO with no injection at t=0.
REC-EDG-6/3	Fail to recover EDG within 6 hours, given failure to recover w/i 3 hours	**1.55E-01**	1.15	**1.10E-01**	1.11	6 hrs	II.SMR.02.009	The nominal 3 hrs reference point is based on a SBO w/SORV scenario with HPCI operating but eventually runs out of steam power and CD occurs at t=3.3 hrs. Nominal 6 hr point is time to core damage in SBO, HPCI or RCIC running until battery failure at t=4hrs, then CD occurs at ~t=6.3 hrs.
REC-OSP-30	Fail to recover offsite power within 30 minutes	2.17E-03	1	**1.21E-01**	1.06	**30 min.**	II.SMR.02.009	Timing basis is an industry data / modeling preference - timing of this event not impacted by EPU.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
REC-OSP-10/6	Fail to recover OSP within 10 hours, given failure to recover w/i 6 hours	(<5E-3)	(<2)	**2.61E-02**	1.01	10 hrs	II.SMR.02.009	Nominal 6 hr reference point is time to core damage in SBO, HPCI or RCIC running until battery failure at t=4hrs, then CD occurs at ~t=6.3 hrs. Nominal 10 hr time is based on Level 2 containment flooding scenario in which DW vent not initiated, and containment fails at 10 hrs during flood process.
REC-OSP-11/6	Fail to recover OSP within 11 hours, given failure to recover w/i 6 hours	1.76E-03	1.00	**5.27E-02**	1.02	11 hrs	II.SMR.02.009	Nominal 6 hr reference point is time to core damage in SBO, HPCI or RCIC running until battery failure at t=4hrs, then CD occurs at ~t=6.3 hrs. Nominal 11 hr. point is time to core damage for SBO w/extended HPCI or RCIC operation by allowing large RPV level swings, but batteries ultimately deplete in t=6-8 hrs, and CD in ~t=11hrs.
REC-OSP-12/11	Fail to recover OSP within 12 hours, given failure to recover w/i 11 hours	1.06E-04	1.00	**9.91E-03**	1.00	12 hrs	II.SMR.02.009	Nominal 11 hr. reference point is time to core damage for SBO w/extended HPCI or RCIC operation by allowing large RPV level swings, but batteries ultimately deplete in t=6-8 hrs, and CD in ~t=11hrs. Nominal 12 hr point is time to 200F in pool with extended HPCI/RCIC operation during SBO.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
REC-OSP-16/12	Fail to recover OSP within 16 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	**4.28E-02**	1.01	16 hrs	II.SMR.02.009	Nominal 12 hr reference point is time to 200F in pool with extended HPCI/RCIC operation during SBO. Nominal 16 hr point is based on Level 2 PRA phenomena issues post RPV melt-through and leading to containment failure.
REC-OSP-22/12	Fail to recover OSP within 22 hours, given failure to recover w/i 12 hours	1.06E-04	1.00	**9.91E-03**	1.01	22 hrs	II.SMR.02.009	Nominal 12 hr reference point is time to 200F in pool with extended HPCI/RCIC operation during SBO. Nominal 22 hr point is based on SBO with extended HPCI operation off the CST, but ultimately fails on low steam pressure and CD occurs at about t=22 hrs.
REC-OSP-29/30	Fail to recover OSP within 2.9 hours, given failure to recover w/i 30 minutes	(<5E-3)	(<2)	**4.15E-02**	1.06	2.9 hrs	II.SMR.02.009	Nominal 30 min. reference point is an industry data / modeling preference (not changed by EPU). 2.9 hr time based on post-core damage accident progression issues.
REC-OSP-3/50	Fail to recover OSP within 3 hours, given failure to recover w/i 50 minutes	1.79E-03	1.00	**7.92E-02**	1.11	3 hrs	II.SMR.02.009	The nominal 50 min. reference time is the time to CD during an SBO with no injection at t=0. The nominal 3 hrs is based on a SBO w/SORV scenario with HPCI operating but eventually runs out of steam power and CD occurs at t=3.3 hrs.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
REC-OSP-34/22	Fail to recover OSP within 34 hours, given failure to recover w/i 22 hours	(<5E-3)	(<2)	**9.64E-03**	1.01	34 hrs	II.SMR.02.009	Nominal 22 hr point is based on SBO with extended HPCI operation off the CST, but ultimately fails on low steam pressure and CD occurs at about t=22 hrs.
REC-OSP-50/30	Fail to recover OSP within 50 minutes, given failure to recover w/i 30 minutes	2.17E-03	1.00	**1.21E-01**	1.02	50 min.	II.SMR.02.009	Nominal 30 min. reference point is an industry data / modeling preference (not changed by EPU). The nominal 50 min. point is the time to CD during an SBO with no injection at t=0.
REC-OSP-6/3	Fail to recover OSP within 6 hours, given failure to recover w/i 3 hours	1.79E-03	1.00	**7.88E-02**	1.05	6 hrs	II.SMR.02.001	The nominal 3 hrs reference point is based on a SBO w/SORV scenario with HPCI operating but eventually runs out of steam power and CD occurs at t=3.3 hrs. Nominal 6 hr point is time to core damage in SBO, HPCI or RCIC running until battery failure at t=4hrs, then CD occurs at ~t=6.3 hrs.
RHRCS-MANY	Fail to manually operate equipment outside of control room before core damage	**2.62E-01**	**64.57**	**6.51E-02**	**16.82**	100 min.	II.SMR.02.008	
RHR-DHR-AY	Fail to align RHR for CHR - ATWS	**5.39E-03**	1.38	**1.84E-02**	**2.30**	**25 min.**	II.SMR.02.008	Diagnosis timing stated to be 20 minutes in II.SMR.02.008. 5 minutes assumed for execution.
RHR-DHR--Y	Fail to align RHR for CHR, when attempted (non-ATWS)	1.57E-03	**98.89**	**5.85E-03**	**366.33**	n/a	II.SMR.02.008	Execution error contribution, not time-based. Diagnosis contribution treated by a separate basic event.
SD-NOTRIPY	Fail to prevent turbine trip while shutting down	(<5E-3)	(<2)	2.73E-04	1.00	**5 min.**	II.SMR.02.008	

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
SHED-DET-Y	Fail to identify load shedding as cause of system failure	1.17E-03	**2.17**	3.51E-03	**4.50**	**30 min.**	II.SMR.02.008	Timing based on battery life and load shedding impact, and not directly on reactor power.
SLC-INI-LY	Fail to initiate SLC - long time available	1.30E-04	1.32	1.63E-03	**5.08**	>1 hr	II.SMR.02.008	Turbine is online and not isolated from reactor. Many hours available to align SLC.
SLC-INI-SY	Fail to initiate SLC - short time available	1.67E-03	1.38	**1.95E-02**	**5.41**	**13.5 min.**	II.SMR.02.008	
SLC-LVL1-Y	Fail to control reactor level (fail SLC), given nominal conditions	3.88E-03	1.38	**4.51E-02**	**5.46**	**10 min.**	II.SMR.02.008	Table 3.3-5 of MNGP IPE Submittal (referenced by II.SMR.02.008), and assuming a manipulation time of 0.5 mins.
SLC-LVL2-Y	Fail to control reactor level (fail SLC), given challenging conditions	7.00E-04	1.05	**8.61E-03**	1.65	**13.5 min.**	II.SMR.02.008	Timing not listed in II.SMR.02.008. Time assumed to be same as SLC-INI-SY.
VENT-CHR-Y	Fail to align containment venting as means of CHR	(<5E-3)	(<2)	1.69E-04	**6.45**	10 hrs.	II.SMR.02.008	Timing not listed in II.SMR.02.008. Time assumed to be same as COND-CHR-Y.
X-DEP-15-Y	Operator fails to depressurize reactor within 15 minutes	(<5E-3)	(<2)	6.31E-05	1.01	**15 min.**	II.SMF.02.037	Referenced from time of core damage.
< THRESHOLD FOR ACTIONS SCREENED FROM FURTHER ANALYSIS >								
ALT-INJ-EY	Fail to align FPS, RHRSW, CSW, or SW within 25 minutes of attempt	(<5E-3)	(<2)	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
ALT-INJ-MY	Fail to align FPS, RHRSW, CSW, or SW - hour available	2.18E-04	1.05	(<5E-3)	(<2)	50 min.	II.SMR.02.008	
ALT-INJ-PB-Y	Fail to align injection before containment breach, given RPV breach	(<5E-3)	(<2)	(<5E-3)	(<2)	8.5 hrs	II.SMF.02.037	

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
ALT-INJ-PD-Y	Fail to align injection before RPV breach, given core damage	(<5E-3)	(<2)	(<5E-3)	(<2)	2 hrs	II.SMF.02.037	
ALT-OIL-Y	Fail to align fuel oil supply from gas powered pump	(<5E-3)	(<2)	(<5E-3)	(<2)	>1 hr	PRA-CALC-05-005	
ATWS-LNG-Y	Fail to initiate ATWS when attempted	(<5E-3)	(<2)	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
C4H-BOOT-Y	Fail to restore loads (boot needed) per C.4-H, given load shed identified	1.03E-04	1.00	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
C4H-EASY-Y	Fail to restore loads (simple CR action) per C.4-H, given load shed identified	1.73E-05	1.00	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
COND-CHR-Y	Operator fails to maintain/establish condenser vacuum	(<5E-3)	(<2)	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
DC-CHARG-Y	Fail to identify battery charger failure, align swing charger	(<5E-3)	(<2)	(<5E-3)	(<2)	45 min.	II.SMR.02.008	
DG13-BFD-Y	Fail to identify DG13 as means of mitigation, implement backfeed	(<5E-3)	(<2)	(<5E-3)	(<2)	>4 hrs	II.SMR.02.008	
FLOODRB12Y	Fail to flood RB within 6-12 hours after torus leak	1.53E-03	1.05	1.18E-03	1.04	6-12 hrs	PRA-CALC-04-021	

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
FLOODRB24Y	Fail to flood RB to allow SRV operation (>12 hours available)	(<5E-3)	(<2)	(<5E-3)	(<2)	>12 hrs	PRA-CALC-04-021	
FW-DVRSN-Y	Fail to identify FW check valve failure, manually isolate	(<5E-3)	(<2)	(<5E-3)	(<2)	50 min.	II.SMR.02.008	
HPV-MAN-Y	Operator fails to manually open HPV	(<5E-3)	(<2)	3.19E-04	1.01	n/a	PRA-CALC-04-044	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
LSBLCALTXY	Operator fails to inject boron using CRDH	(<5E-3)	(<2)	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.
OIL-LOSS-MY	Fail to align power to non-emergency MCC-31 before EDG fuel is depleted	(<5E-3)	(<2)	(<5E-3)	(<2)	>1 hr	PRA-CALC-05-005	
OIL-LOSS-Y	Fail to identify need to address loss of fuel flow to EDG day tanks - nominal	5.63E-04	1.56	2.55E-04	1.25	>1 hr	PRA-CALC-05-005	
OIL-SUPPLY-Y	Fail to maintain oil inventory in EDG fuel oil storage tank	(<5E-3)	(<2)	(<5E-3)	(<2)	>1 hr	PRA-CALC-05-005	
PUMPER-S-Y	Fail to provide FPS supply from fire pumper truck - 50 minutes available	1.41E-05	1.00	(<5E-3)	(<2)	50 min.	PRA-CALC-04-042	
RCIC-BYP-Y	Fail to bypass RCIC high exhaust pressure interlock	(<5E-3)	(<2)	(<5E-3)	(<2)	> 4 hrs	PRA-CALC-04-021	Time between time of core damage for loss of DHR-induced failure of RCIC due to reaching the RCIC high exhaust pressure interlock setpoint until the time of core damage. Ref. PRA-CALC-04-021.

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
R-DHR-PB-Y	Fail to align RHR for DHR before containment failure, given core damage	(<5E-3)	(<2)	(<5E-3)	(<2)	12 hrs	II.SMF.02.037	Time between time of core damage for loss of DHR-induced SRV closure (t=~22 hrs) until loss of DHR-induced containment failure (t=~34hrs). Ref. II.SMF.02.037.
REC-EDG-10/6	Fail to recover EDG within 10 hours, given failure to recover w/i 6 hours	(<5E-3)	(<2)	3.11E-04	1.00	10 hrs	II.SMR.02.009	
REC-EDG-12/11	Fail to recover EDG within 12 hours, given failure to rec+B28over w/i 11 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	12 hrs	II.SMR.02.009	
REC-EDG-15/12	Fail to recover EDG within 15 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	15 hrs	II.SMR.02.009	
REC-EDG-17/12	Fail to recover EDG within 17 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	17 hrs	II.SMR.02.009	
REC-EDG-19/12	Fail to recover EDG within 19 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	19 hrs	II.SMR.02.009	
REC-EDG-2/30	Fail to recover EDG within 2.1 hours, given failure to recover w/i 30 minutes	(<5E-3)	(<2)	(<5E-3)	(<2)	2.1 hrs	II.SMR.02.009	
REC-EDG-22/12	Fail to recover EDG within 22 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	22 hrs	II.SMR.02.009	
REC-EDG-24/22	Fail to recover EDG within 24 hours, given failure to recover w/i 22 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	24 hrs	II.SMR.02.009	
REC-EDG-29/30	Fail to recover EDG within 2.9 hours, given failure to recover w/i 30 minutes	(<5E-3)	(<2)	2.90E-05	1.00	2.9 hrs	II.SMR.02.009	

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
REC-EDG-34/22	Fail to recover EDG within 34 hours, given failure to recover w/i 22 hours	(<5E-3)	(<2)	2.18E-03	(<2)	34 hrs	II.SMR.02.009	
REC-EDG-35/22	Fail to recover EDG within 35 hours, given failure to recover w/i 22 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	35 hrs	II.SMR.02.009	
REC-EDG-5/3	Fail to recover EDG within 5.3 hours, given failure to recover w/i 3 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	5.3 hrs	II.SMR.02.009	
REC-EDG-9/6	Fail to recover EDG within 9 hours, given failure to recover w/i 6 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	9 hrs	II.SMR.02.009	
REC-OSP-15/12	Fail to recover OSP within 15 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	15 hrs	II.SMR.02.009	
REC-OSP-17/12	Fail to recover OSP within 17 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	17 hrs	II.SMR.02.009	
REC-OSP-19/12	Fail to recover OSP within 19 hours, given failure to recover w/i 12 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	19 hrs	II.SMR.02.009	
REC-OSP-2/30	Fail to recover OSP within 2.1 hours, given failure to recover w/i 30 minutes	(<5E-3)	(<2)	9.93E-05	1.00	2.1 hrs	II.SMR.02.009	
REC-OSP-24/22	Fail to recover OSP within 24 hours, given failure to recover w/i 22 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	24 hrs	II.SMR.02.009	
REC-OSP-35/22	Fail to recover OSP within 35 hours, given failure to recover w/i 22 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	35 hrs	II.SMR.02.009	
REC-OSP-5/3	Fail to recover OSP within 5.3 hours, given failure to recover w/i 3 hours	(<5E-3)	(<2)	3.79E-04	1.00	5.3 hrs	II.SMR.02.009	
REC-OSP-9/6	Fail to recover OSP within 9 hours, given failure to recover w/i 6 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	9 hrs	II.SMR.02.009	

Table D-1

SUMMARY OF OPERATOR ACTION SCREENING PROCESS[1]

Operator Action ID	Action Description	FV (CDF)	RAW (CDF)	FV (LERF)	RAW (LERF)	Allowable Action Timing	Basis for Action Timing	
							MNGP Reference	Comment
RHR-SDC--Y	Fail to align RHR for DHR in SDC mode	(<5E-3)	(<2)	(<5E-3)	(<2)	10 hrs.	II.SMR.02.008	Timing not listed in II.SMR.02.008. Time assumed to be same as COND-CHR-Y.
RHRSW-CHRY	Fail to bypass load shed and start pump	(<5E-3)	(<2)	(<5E-3)	(<2)	10 hrs.	II.SMR.02.008	Timing not listed in II.SMR.02.008. Time assumed to be same as COND-CHR-Y.
SBO-ALIGNY	Fail to align HPCI/RCIC or load shed to prolong injection capabilities	1.40E-04	1.03	(<5E-3)	(<2)	70 min.	II.SMR.02.008	
SBODETECTY	Fail to determine need to address SBO within 5 hours	(<5E-3)	(<2)	(<5E-3)	(<2)	5 hrs	II.SMR.02.008	
SLC-CRD--Y	Fail to inject boron using CRDH	(<5E-3)	(<2)	(<5E-3)	(<2)	n/a	II.SMR.02.008	Execution error contribution, HEP calculation not directly influenced by available time window. Diagnosis contribution treated by a separate basic event.

(1) This operator action screening was performed using the 2005 Monticello PRA ***average maintenance*** model (fault tree ***Risk-T&M.caf***).

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
020-ISOL-M-Y	Fail to isolate a medium or large leak within 20 minutes	20 min.	20 min.	3.00E-01	3.00E-01	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
030-ISOL-M-Y	Fail to isolate a medium or large leak within 30 minutes	30 min.	30 min.	3.00E-02	3.00E-02	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
030-ISOL-S-Y	Fail to isolate a small leak within 30 minutes	30 min.	30 min.	3.00E-01	3.00E-01	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
060-ISOL-M-Y	Fail to isolate a medium or large leak within 60 minutes	60 min.	60 min.	3.00E-03	3.00E-03	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
060-ISOL-S-Y	Fail to isolate a small leak within 60 minutes	60 min.	60 min.	3.00E-02	3.00E-02	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
120-ISOL-S-Y	Fail to isolate a small leak within 120 minutes	120 min.	120 min.	3.00E-03	3.00E-03	Based on time to equipment submergence due to internal flooding and not dependent on reactor power.
ALT-INJ-LY	Fail to align FPS, RHRSW, CSW, or SW - hour available TSC support	n/a	n/a	8.00E-04	8.00E-04	Execution Error: No impact on HEP, this event is solely execution error (diagnosis error addressed by separate event).
ALT-POWER-Y	Fail to align alternate power supplies directly to MCC-44	>4hrs	>4hrs	5.00E-03	5.00E-03	Timing based on battery life and not directly on reactor power (action timing for this HEP does not explicitly credit the additional time until core damage after DC batteries deplete).
ASDS-DEP-Y	Fail to implement depressurization from ASDS panel	1 hr	50 min.	1.00E-02	1.00E-02	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax show time window reduced to approximately 50 min. for EPU case. Screening HEP not impacted by EPU.
ATWS-SHT-Y	Operator fails to initiate ATWS (short time available)	<1 min.	<1 min.	1.00E+00	1.00E+00	ASEP Upper Bound TRC curve.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
CHR-DET--Y	Fail to identify need for containment heat removal	8 hrs	6.8 hrs	1.00E-06	1.00E-06	Diagnosis Error: Timing based on time to SP/T = 200F for transients with no SPC. MNGP EPU MAAP run MNGPEPU9 shows the time is t=6.8 hrs for EPU condition. ASEP Lower Bound TRC curve, and 1E-6 HEP minimum threshold in MNGP PRA. HEP unchanged.
CRD-LSBYPY	Fail to restore CRDH after LOSP and ECCS load shed	25 min.	21 min	8.00E-02	1.23E-01	MNGP EPU MAAP runs MNGPEPU5d and MNGPEPU5dx show that the time available is reduced approximately 15% for the EPU (using times to maximize core temp). EPU diagnosis time is 11 min. and execution time is 10 min. ASEP Median TRC curve.
CRD-PUMP-Y	Fail to start second CRDH pump from control room	25 min.	21 min	9.00E-03	1.40E-02	MNGP EPU MAAP runs MNGPEPU5d and MNGPEPU5dx show that the time available is reduced approximately 15% for the EPU (using times to maximize core temp). EPU diagnosis time is 20 min. and execution time is 1 min. ASEP Median TRC curve.
CRD-VALV-Y	Fail to maximize CRDH flow - valves in RB	25 min.	21 min	4.00E-02	5.27E-02	MNGP EPU MAAP runs MNGPEPU5i and MNGPEPU5ix show that the time available is reduced approximately 15% for the EPU (using times to maximize core temp). EPU diagnosis time is 14 min. and execution time is 7 min. ASEP Median TRC curve.
CRIT-DET-Y	Fail to detect criticality issue - long time available	30 min.	30 min.	1.18E-04	1.18E-04	Diagnosis Error: This action error applies to ATWS scenarios in which the turbine is online. An indefinite, long time is available to the operator; the PRA conservatively assumes 30 mins. available. This timing assumption is not changed by the EPU. ASEP Lower Bound TRC curve. Base PRA mistakenly used 40 min. for the HEP calculation; base HEP revised in this calculation to use the correct base value of 30 min.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
CST-FILL-Y	Fail to refill the CSTs	>15 hrs	<15 hrs	1.00E-03	1.00E-03	Timing based on CST inventory depletion due to use for RPV coolant makeup long term. CLTP PRA assumes time available >15 hrs, and 1 hr required for alignment. EPU time available would be reduced, but would have to be reduced unrealistically (by 10 hrs or more) to change the CLTP HEP which is dominated by execution error. ASEP Median TRC curve.
DEP-02MN-Y	Fail RPV depressurization within 2 minutes	5 min.	4.4 min.	2.50E-01	5.10E-01	This action used in isolation ATWS scenarios with failure of all HP injection. The CLTP PRA estimates 5 minutes available (diagnosis time of 2 min. and execution time of 3 min.). MNGP EPU MAAP runs MNGPEPU7a and MNGPEPU7ax show that this timing is not reduced significantly (<10%) for the EPU, a 13% reduction is assumed in the EPU risk assessment. EPU time available is estimated at 4.4 min. (diagnosis time of 1.4 min. and execution time of 3 min.). ASEP Lower Bound TRC curve. CLTP base PRA mistakenly used 3 min. diagnosis for the HEP calculation; base HEP revised in this calculation to use the correct base diagnosis time of 2 min.
DEP-12MN-Y	Fail RPV depressurization within 12 minutes	15 min.	13.1 min.	5.20E-03	9.84E-03	This action is applicable to MLOCA scenarios with no HP injection available. MNGP EPU MAAP runs MNGPEPU8b and MNGPEPU8bx indicate that the time is reduced 10-13% for the EPU, a 13% reduction is assumed for the EPU. EPU time available estimated at 13.1 min (diagnosis time of 10.1 min. and execution time of 3 min.). ASEP Lower Bound TRC curve.

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RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
DEP-50MN-Y	Fail RPV depressurization within 50 minutes	50 min.	42 min.	1.80E-04	1.90E-04	This action is applicable to non-LOCA and non-ATWS scenarios with no HP injection available. MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min. (diagnosis time is 39 min. and execution time of 3 min). ASEP Lower Bound TRC curve.
DEP-HOUR-Y	Fail RPV depressurization >an hour available	103 min.	103 min.	1.60E-04	1.60E-04	This action is applicable to non-LOCA and non-ATWS scenarios with HP injection initially available, but RPV ED required later for other reasons (e.g., HCTL, HP injection failure). CLTP assumes a diagnosis time of 100 minutes, and an execution time of 3 mins. MNGP EPU MAAP runs MNGPEPU8c and MNGPEPU8d show that for scenarios requiring late RPV ED due to issues such as HCTL or HP injection failure significantly more than 100 mins. remain before core damage occurs. Thus, the CLTP time available for this action is unchanged for the EPU. ASEP Lower Bound TRC curve.
DEP-PD-Y	Fail to depressurize reactor after core damage, but before vessel penetration	2 hrs	~2 hrs	1.00E-01	1.00E-01	Timing based on post-core damage accident progression assumptions and time to RPV melt-through. Screening HEP not impacted by EPU.
DW-VENT-PRG	Fail to prevent H2 burn failing containment by vent/purge	< 30 min.	< 30 min.	1.00E+00	1.00E+00	
FLOODRB16Y	Fail to flood RB within 1-6 hours after torus leak	1-6 hrs	1-6 hrs.	3.00E-01	3.00E-01	Timing based on internal flooding issues and not directly on reactor power.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
FW-CNTRL-Y	Fail to control FW as high pressure injection source following transient	15 min.	12 min.	4.60E-03	5.46E-03	The available action time is based on the time to reach TAF for an isolation transient with loss of all HP injection. MNGP EPU MAAP run MNGPEPU8a show that this time is approximately t=12 min. for the EPU power level. EPU time available estimated at 12 mins (diagnosis time of 11 min. and execution time of 1 min.). ASEP Lower Bound TRC curve.
FW-REFLG-Y	Fail to identify reference leg leak	7 min.	5.5 min.	4.00E-02	6.94E-02	The time available is based on the time to reach TAF for a ref. leg break event with no high pressure injection. Time available for CLTP estimated at t=7 mins. MNGP EPU MAAP runs MNGPEPU6c, MNGPEPU6cx, MNGPEPU1b and MNGPEPU1bx indicate that this time frame is reduced approximately 20-22% due to the EPU. EPU time available estimated at 5.5 mins. (diagnosis time of 4.5 min. and execution time of 1 min.). ASEP Lower Bound TRC curve.
HPI-CSTS-Y	Fail to defeat high torus level suction transfer	1 hr	1 hr	3.00E-03	3.00E-03	This action applies to scenarios with pool temperature reaching 200F and need to switch HPCI/RCIC suction to CST to prevent failure of pump due to overheating. Timing of 1 hr. used in CLTP not based directly on reactor power, this time is not adjusted for the EPU. ASEP Lower Bound TRC curve.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
LEVEL-05-Y	Fail to detect need for injection within 5 minutes of compelling signal	30 min.	26 min.	5.00E-02	1.00E+00	Diagnosis Error: Time available in CLTP PRA based on time to core damage for SLOCA type scenarios with no HP injection, estimated at t=30 minutes and 25 minutes to execute the action (thus, 5 min. diagnosis time). MNGP EPU MAAP runs MNGPEPU6c and MNGPEPU6cx show that this time frame is reduced to approximately t=26 mins (thus, 1 min. diagnosis time). ASEP Lower Bound TRC curve.
LEVEL-25-Y	Fail to detect need for injection within 25 minutes of compelling signal	50 min.	42 min.	6.00E-04	1.72E-03	Diagnosis Error: This action is applicable to non-LOCA and non-ATWS scenarios with no HP injection available. The CLTP PRA estimates the available window at 50 minutes and 25 minutes to execute the action (thus, 25 min. diagnosis time). MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min. (diagnosis time is 17 min. and execution time of 25 min). ASEP Lower Bound TRC curve.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
LEVEL-45-Y	Fail to detect need for injection within 45 minutes of compelling signal	~1 hr	~1 hr.	1.00E-05	1.00E-05	Diagnosis Error: This action is applicable to non-LOCA and non-ATWS scenarios with HP injection initially available, but RPV ED required later for other reasons (e.g., HCTL, HP injection failure). CLTP assumes diagnosis time available is 45 minutes, then an additional 25 minutes for execution (thus, total time available greater than 1 hr.) MNGP EPU MAAP runs MNGPEPU8c and MNGPEPU8d show that for scenarios requiring late RPV ED due to issues such as HCTL or HP injection failure that significantly more than 1 hour remains before core damage occurs. Thus, the CLTP diagnosis time for this action of 45 mins. is unchanged for the EPU. ASEP Lower Bound TRC curve.
L-LONG---Y	Operator fails to inject boron using SBLC - long time available	>1 hr	>1 hr	4.00E-04	4.00E-04	This action error applies to ATWS scenarios in which the turbine is online. An indefinite, long time is available to the operator; the PRA conservatively assumes > 1 hr. available. This timing assumption would not be changed by the EPU. ASEP Lower Bound TRC curve. In addition, the HEP is dominated by execution error.
OIL-LOSS-HY	Fail to identify need to address loss of fuel flow to EDG day tanks - high	>1 hr	>1 hr	1.00E-01	1.00E-01	Timing based on EDG fuel consumption and not directly on reactor power. Screening HEP not impacted by EPU.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
PUMPER-L-Y	Fail to provide FPS supply from fire pumper truck - hours available	6 hrs	6 hrs	1.00E-03	1.00E-03	The available time is estimated in the CLTP PRA based on the time to core damage for an SBO, with HPCI or RCIC initial operation but subsequent failure due to battery depletion. The CLTP PRA estimates that >6hrs are available before core damage in such scenarios (t=6 hrs is used in the CLTP PRA for this HEP). MNGP EPU MAAP run MNGPEPU8c shows core damage occurs at t=6.6 hrs for such scenarios for the EPU. As such, the 6 hr available time for this action is not adjusted for the EPU. ASEP Median TRC curve. Dominated by execution error.
RCIC-MAN-Y	Fail to manually operate RCIC	n/a	n/a	5.00E-02	5.00E-02	Execution Error: No impact on HEP, this event is solely execution error (diagnosis error addressed by separate event).
REC-EDG-30	Fail to recover EDG within 30 minutes	30 min.	30 min.	8.5E-01	8.5E-01	Timing based on industry data and associated LOOP event tree modeling assumptions. Timing and probability not impacted by EPU.
REC-EDG-11/6	Fail to recover EDG within 11 hours, given failure to recover w/i 6 hours	11 hrs / 6 hrs	11 hrs / 6 hrs	7.3E-01	7.3E-01	Nominal times of 11 hrs and 6 hrs still appropriate for EPU (see EPU MAAP run MNGPEPU8c). Existing recovery failure probability already high. Time frame is long and AC recovery curves flatten out at such lengthy time frames, such that any postulated change to this recovery probability would not have a significant impact on risk.
REC-EDG-12/11	Fail to recover EDG within 12 hours, given failure to recover w/i 11 hours	12 hrs / 11 hrs	11 hrs / 11 hrs	9.3E-01	1.0E+00	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-EDG-16/12	Fail to recover EDG within 16 hours, given failure to recover w/i 12 hours	16 hrs / 12 hrs	16 hrs / 11 hrs	9.0E-01	8.5E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.

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RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
REC-EDG-22/12	Fail to recover EDG within 22 hours, given failure to recover w/i 12 hours	22 hrs / 12 hrs	22 hrs / 11 hrs	7.3E-01	6.5E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-EDG-3/50	Fail to recover EDG within 3 hours, given failure to recover w/i 50 minutes	3 hrs /50 mins.	3 hrs /42 mins.	6.9E-01	6.6E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-EDG-50/30	Fail to recover EDG within 50 minutes, given failure to recover w/i 30 minutes	50 min. / 30 min.	42 min. / 30 min.	9.1E-01	9.4E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-EDG-6/3	Fail to recover EDG within 6 hours, given failure to recover w/i 3 hours	6 hrs / 3 hrs	6 hrs / 3 hrs	5.1E-01	5.1E-01	Nominal times of 6 hrs and 3 hrs still judged reasonable for EPU.
REC-OSP-30	Fail to recover offsite power within 30 minutes	30 min.	30 min.	6.8E-01	6.8E-01	Timing based on industry data and associated LOOP event tree modeling assumptions. Timing and probability not impacted by EPU.
REC-OSP-10/6	Fail to recover OSP within 10 hours, given failure to recover w/i 6 hours	10 hrs / 6 hrs	10 hrs / 6 hrs	8.0E-01	8.0E-01	Nominal times of 10 hrs and 6 hrs still judged reasonable for EPU.
REC-OSP-11/6	Fail to recover OSP within 11 hours, given failure to recover w/i 6 hours	11 hrs / 6 hrs	11 hrs / 6 hrs	7.5E-01	7.5E-01	Nominal times of 11 hrs and 6 hrs still appropriate for EPU (see EPU MAAP run MNGPEPU8c). Existing recovery failure probability already high. Time frame is long and AC recovery curves flatten out at such lengthy time frames, such that any postulated change to this recovery probability would not have a significant impact on risk.
REC-OSP-12/11	Fail to recover OSP within 12 hours, given failure to recover w/i 11 hours	12 hrs / 11 hrs	11 hrs / 11 hrs	9.2E-01	1.0E+00	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-OSP-16/12	Fail to recover OSP within 16 hours, given failure to recover w/i 12 hours	16 hrs / 12 hrs	16 hrs / 11 hrs	8.0E-01	7.3E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
REC-OSP-22/12	Fail to recover OSP within 22 hours, given failure to recover w/i 12 hours	22 hrs / 12 hrs	22 hrs / 11 hrs	5.0E-01	4.5E-01	MAAP run MNGPEPU8d indicates that the t=12 hr time frame is reduced to approximately t=11 hrs for the EPU.
REC-OSP-29/30	Fail to recover OSP within 2.9 hours, given failure to recover w/i 30 minutes	2.9 hrs / 30 min.	2.9 hrs / 30 min.	4.2E-01	4.2E-01	No change assumed for 2.9 hr post-core damage progression time frame, time reasonable.
REC-OSP-3/50	Fail to recover OSP within 3 hours, given failure to recover w/i 50 minutes	3 hrs / 50 mins.	3 hrs / 42 mins.	4.3E-01	4.1E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-OSP-34/22	Fail to recover OSP within 34 hours, given failure to recover w/i 22 hours	34 hrs / 22 hrs	34 hrs / 22 hrs	5.0E-01	5.0E-01	Existing recovery failure probability already high. Time frame is long and AC recovery curves flatten out at such lengthy time frames, such that any postulated change to this recovery probability would not have a significant impact on risk.
REC-OSP-50/30	Fail to recover OSP within 50 minutes, given failure to recover w/i 30 minutes	50 min. / 30 min.	42 min. / 30 min.	8.5E-01	9.0E-01	MNGP EPU MAAP runs MNGPEPU8a and MNGPEPU8ax shows that this timing is reduced approximately 16% for the EPU. EPU time available estimated at 42 min.
REC-OSP-6/3	Fail to recover OSP within 6 hours, given failure to recover w/i 3 hours	6 hrs / 3 hrs	6 hrs / 3 hrs	6.0E-01	6.0E-01	Nominal times of 6 hrs and 3 hrs still judged reasonable for EPU.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
RHRCS-MANY	Fail to manually operate equipment outside of control room before core damage	100 min.	100 min.	4.10E-03	4.10E-03	This action is applicable to non-LOCA and non-ATWS scenarios with HP injection initially available, but RPV ED required later for other reasons (e.g., HCTL, HP injection failure). CLTP assumes time available is 100 minutes (diagnosis time of 90 min. and execution time of 10 min.). MNGP EPU MAAP runs MNGPEPU8c and MNGPEPU8d show that for scenarios requiring late RPV ED due to issues such as HCTL or HP injection failure that more than 100 mins. remain before core damage occurs. Thus, the CLTP time in this action of 100 mins. is unchanged for the EPU. ASEP Median TRC curve. Dominated by execution error.
RHR-DHR-AY	Fail to align RHR for CHR - ATWS	25 min.	21.8 min.	1.40E-02	2.19E-02	This action is applicable to ATWS scenarios with HP injection and successful SLC. Time available to align SPC depends upon time of SLC injection and whether the initiator is an isolation event. CLTP PRA assumes that 25 minutes are available (diagnosis time of 20 mins. and execution time of 5 mins.). This time is judged conservative. MNGP EPU runs MNGPEPU7b, MNGPEPU7bx, MNGPEUP7c and MNGPEPU7cx show that with delayed SLC injection and no SPC initiation, critical impacts do not occur until about t=45 mins when the pool reaches 200F and HPCI operability become an issue. Although the 25 min. time available estimate from the CLTP is judged still appropriate for the EPU, the EPU risk assessment reduces this time available by 13% to t=21.8 mins (diagnosis time of 16.8 min. and execution time of 5 min.). ASEP Median TRC curve.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
RHR-DHR--Y	Fail to align RHR for CHR, when attempted (non-ATWS)	8 hrs.	6.8 hrs	1.60E-05	1.60E-05	Execution Error: Time window same as for CHR-DET--Y; however, this is an execution error contribution, the low error rate is due to multiple applicable error recovery factors (long time frame, other operators, etc.). The reduction in time available due to the EPU does not change the execution error rate. Diagnosis contribution treated by separate basic event CHR-DET--Y.
SD-NOTRIPY	Fail to prevent turbine trip while shutting down	5 min.	4.4 min.	2.00E-01	2.27E-01	This action is for bypassing the MSIV level interlocks and is applicable to ATWS scenarios with the MSIVs open. The time available depends upon a number of factors, such as which HP systems are available and how long operators take to reduce level. The CLTP PRA assumes the available diagnosis time is t=5 min. The CLTP diagnosis time is reduced 13% for the EPU. ASEP Median TRC curve. Base PRA mistakenly selected 0.3 off the ASEP curve instead of the correct base value of 0.20; base HEP revised in this calculation to use the correct base HEP of 0.20.
SHED-DET-Y	Fail to identify load shedding as cause of system failure	30 min.	30 min.	1.00E-03	1.00E-03	Timing based on battery life and load shedding impact. Timing and probability not impacted by EPU.
SLC-INI-LY	Fail to initiate SLC - long time available	>1 hr	>1 hr.	4.00E-04	4.00E-04	This action error applies to ATWS scenarios in which the turbine is online. An indefinite, long time is available to the operator; the PRA assumes > 1 hr. available. This timing assumption is not changed by the EPU. ASEP Lower Bound TRC curve. In addition, the HEP is dominated by execution error.

Table D-2

RE-ASSESSMENT OF KEY OPERATOR ACTION HEPs FOR THE EPU

Action ID	Action Description	Allowable Action Time		Base HEP	EPU HEP	Comment
		Current PRA Power (CLTP)	EPU Power (113% CLTP)			
SLC-INI-SY	Fail to initiate SLC - short time available	13.5 min.	11.8 min.	4.40E-03	6.17E-03	Total time available reduced 13%. MNGP EPU MAAP runs MNGPEPU7a, MNGPEPU7b, and MNGPEPU7c show that that such a time frame for SLC injection is successful for the EPU condition. ASEP Lower Bound TRC curve.
SLC-LVL1-Y	Fail to control reactor level (fail SLC), given nominal conditions	10 min.	8.7 min.	1.00E-02	1.53E-02	Total time available reduced 13%. EPU diagnosis time of 8.2 min. and execution time of 0.5 min. ASEP Lower Bound TRC curve.
SLC-LVL2-Y	Fail to control reactor level (fail SLC), given challenging conditions	13.5 min.	11.8 min.	1.30E-02	1.97E-02	Total time available reduced 13%. EPU diagnosis time of 11.3 min. and execution time of 0.5 min. ASEP Lower Bound TRC curve.
VENT-CHR-Y	Fail to align containment venting as means of CHR	8 hrs.	6.8 hrs	3.10E-05	3.68E-05	Timing based on time to SP/T = 200F for transients with no SPC. MNGP EPU MAAP run MNGPEPU9 shows the time is t=6.8 hrs for EPU condition. ASEP Median TRC curve.
X-DEP-15-Y	Operator fails to depressurize reactor within 15 minutes	15 min.	15 min.	5.20E-03	5.20E-03	This action is used in high pressure ATWS core damage scenarios. The CLTP PRA assumes 15 min. available (diagnosis time of 12 min. and execution time of 3 mins.). The time available is based on post-accident progression modeling assumptions and not directly on core power. This time frame is not changed for the EPU. ASEP Lower Bound TRC curve.

Appendix E

MNGP EPU MAAP CALCULATIONS

Appendix E
MNGP EPU MAAP CALCULATIONS

The Modular Accident Analysis Package (MAAP) is used to calculate changes in the thermal hydraulic profile for specific issues (e.g., boildown timing) to support the MNGP EPU risk assessment.

MAAP is an industry recognized thermal hydraulics code used to evaluate design basis and beyond design basis accidents. MAAP (Version 4.0.6) and the latest MNGP MAAP parameter file (*MO406_061907.par*) have been used in this evaluation. The parameter file contains plant specific parameters representing the primary system and containment.

MAAP cases of various accident scenarios were defined and run to identify changes in timings and success criteria due to the EPU. A separate run was made for the CLTP power and for the EPU power level for each analyzed accident scenario. The pre-EPU version of each scenario is identified with an 'x' in the case identifier (e.g., Case MNGPEPU1a is an EPU power run and Case MNGPEPU1ax is the corresponding CLTP power run). A summary of the MAAP runs performed in support of this risk assessment is provided in Tables E-1 (Level 1 PRA runs) and Table E-2 (Level 2 PRA runs).

LOFW, SORV and RCIC

In addition to performing MAAP runs to identify accident timing and success criteria changes for consideration in the EPU risk assessment, multiple MAAP runs were performed to address NUREG-0737 Item II.K.3.44 (adequate core cooling for LOFW with an additional single failure) for the MNGP EPU. These scenarios are identified here as cases MNGPEPU2a and MNGPEPU2b. These scenarios are Loss of Feedwater (LOFW) initiated events with a SORV and RCIC as the initial high pressure injection source.

Case MNGPEPU2a is designed to prevent RPV emergency depressurization. In this scenario, LOFW is the initiating event (no credit is given for FW coast down flow into the RPV). One (1) SRV sticks open during the initial pressure transient and remains stuck open throughout the run. RCIC is the only high pressure injection source and it auto initiates as designed. RCIC is not sufficient to prevent RPV level dipping below TAF; however, adequate core cooling is maintained throughout the sequence. When RPV pressure reduces sufficiently to the LP ECCS interlock pressure, one (1) train of LPCI auto injects into the RPV (RCIC subsequently trips on low steam pressure). LPCI flow into the RPV begins at t=25 mins. (pool temperature at this time is 110F).

Case MNGPEPU2b is similar to the case above except that RPV emergency depressurization is initiated at TAF using 2 SRVs. Like the previous case, RCIC is not sufficient to prevent level dipping below TAF; however, adequate core cooling is maintained throughout the sequence. After RPV depressurization at TAF, one (1) train of LPCI auto injects into the RPV. LPCI flow into the RPV begins at t=7 mins in this case (pool temperature at this time is 100F).

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU1a	MSIV Closure, no HP injection, delayed ED, and 1 LPCI pump • EPU power level • MSIV Closure at t=0 (FW coast down flow credited) • Only 1 SRV available for initial pressure transient • No HP injection • Initiate Emergency RPV Depressurization (using only 1 SRV) at MSCWLL • Initiate 1 LPCI pump at LP interlock • SPC w/1 RHR train initiated at pool temp. 90F[3]	• Verify 1 SRV sufficient for pressure control to prevent exceeding RPV pressure operability limits for Transients • Verify 1 SRV sufficient for RPV ED for Transients	13 min.	25 min.	32 min. Max. temp. of 1400°F	18 min. MSCWLL	5 hr.	Max RPV pressure of 1530 psia when only 1 SRV available. Sensitivity case MNGPEPU1a_a shows success with 2 SRVs available (max. RPV press. of 1427 psia). Thus, EPU requires 2 SRVs for RPV Overpressure Protection during isolation transients. RPV ED initiated at t=18 (MSCWLL) min. with 1 SRV. Thus, one SRV sufficient for RPV ED for EPU for Transients and SLOCAs when LP ECCS available.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU1ax	Same as MNGPEPU1a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	16 min.	30 min.	36 min. Max. temp. of 1225°F	23 min. MSCWLL	5 hr.	Max RPV Press 1443 psia. Thus, one SRV sufficient for RPV Overpressure Protection for CLTP for transients with MSIV closure. RPV ED initiated at t=23 min. (MSCWLL) with 1 SRV. Thus, one SRV sufficient for RPV ED for CLTP for Transients and SLOCAs when LP ECCS available.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU1b	LOFW, no HP injection, delayed ED, and 1 LPCI pump • EPU power level • LOFW at t=0 (no FW coast down flow credited) • MSIVs remain open until isolate on low RPV level • Only 1 SRV available for initial pressure transient • No HP injection • Initiate Emergency RPV Depressurization (using only 1 SRV) at MSCWLL • Initiate 1 LPCI pump at LP interlock • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify 1 SRV sufficient for pressure control to prevent exceeding RPV pressure operability limits for Transients • Verify 1 SRV sufficient for RPV ED for Transients	7 min.	20 min.	26 min. Max. temp. of 1425°F	13 min. MSCWLL	5 hr.	Same as Case MNGPEPU1a except LOFW at t=0 and MSIVs initially open until then isolate on low RPV water Level. Max RPV Press 1068 psia. Therefore, cases MNGPEPU1a and MNGPEPU1a_a bound the RPV Overpressure SRV success criteria for Transients with MSIV closure for the EPU condition. RPV ED initiated at t=13 min. (MSCWLL) with 1 SRV. Thus, one SRV sufficient for RPV ED for EPU for Transients and SLOCAs when LP ECCS available.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU1bx	Same as MNGPEPU1b except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	9 min.	24 min.	30 min. Max. temp. of 1230°F	16 min. MSCWLL	5 hr.	Max RPV Press 1068 psia. Therefore, case MNGPEPU1ax bounds the RPV Overpressure SRV success criteria for Transients with MSIV closure for the CLTP condition. RPV ED initiated at t=16 min. (MSCWLL) with 1 SRV. Thus, one SRV sufficient for RPV ED for CLTP for Transients and SLOCAs when LP ECCS available.
MNGPEPU2	MSIV Closure, no initial HP injection, no RPV ED, and RCIC initiated late • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SRVs/SVs available for initial pressure control • No HP injection initially • No RPV ED • Iterate to determine time when initiation of RCIC prevents core damage • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify time allowable for manual initiation of RCIC during Transient	12 min.	35 min.	53 min. Max. temp. of 2060°F	18 min. MSCWLL	2.5 hr.	RCIC initiated at t=45 min. RCIC initiation time iterated to determine latest time allowable for initiation in order to prevent core damage. This case shows that RCIC initiation can be delayed for EPU until t=45 min. and prevent core damage for an MSIV Closure with loss of all other injection and no RPV ED.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU2x	Same as MNGPEPU2 except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	16 min.	43 min.	1.0 hr. Max. temp. of 1930°F	23 min. MSCWLL	2.5 hr.	RCIC initiated at t=55 min. RCIC initiation time iterated to determine latest time allowable for initiation in order to prevent core damage. This case shows that RCIC initiation can be delayed for CLTP until t=55 min. and prevent core damage for an MSIV Closure with loss of all other injection and no RPV ED.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU2a	LOFW, SORV, RCIC for initial injection, no RPV ED, and 1 LPCI pump • EPU power level • LOFW at t=0 (no FW coast down flow credited) • MSIVs remain open until isolate on low RPV level • All SRVs/SVs available for initial pressure control • One (1) SORV • Only HP injection is RCIC (auto initiates) • No RPV ED • 1 LPCI pump injects at ECCS LP interlock • SPC w/1 RHR train initiated at pool temp. 90F[3]	• Verify that RCIC and then 1 LPCI pump is sufficient to prevent core damage during LOFW w/SORV	4 min.	n/a	n/a Max. temp. at t=0	6 min. MSCWLL	2.5 hr.	RCIC auto initiates but fails to maintain level such that level dips below TAF. One (1) LPCI pump injects when RPV pressure drops to the ECCS LP interlock (RCIC subsequently trips on low steam pressure). Adequate core cooling maintained throughout. LPCI flow into vessel begins at t=25 mins. (SP/T at this time is 110F). This case addresses Item II.K.3.44 of NUREG-0737 (adequate core cooling for LOFW with an additional single failure) for EPU.
MNGPEPU2ax	Same as MNGPEPU2a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	4 min.	n/a	n/a Max. temp. at t=0	6 min. MSCWLL	2.5 hr.	Same comment as for Case MNGPEPU2a, except this case is for CLTP. LPCI flow into vessel for this case begins at t=16 mins. (SP/T at this time is 105F).

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU2b	LOFW, SORV, RCIC for initial injection, RPV ED, and 1 LPCI pump • EPU power level • LOFW at t=0 (no FW coast down flow credited) • MSIVs remain open until RPV level reaches L1 (low low) • All SRVs/SVs available for initial pressure control • One (1) SORV • Only HP injection is RCIC (auto initiates) • RPV ED at TAF with 2 additional SRVs • 1 LPCI pump injects at ECCS LP interlock • SPC w/1 RHR train initiated at pool temp. 90F[3]	<Same as case above>	4 min.	n/a	n/a Max. temp. at t=0	5 min. MSCWLL	2.5 hr.	Same comment as for Case MNGPEPU2a, except this case involves RPV ED at TAF. LPCI flow into vessel for this case begins at t=7 mins. (SP/T at this time is 100F).
MNGPEPU2bx	Same as MNGPEPU2b except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	4 min.	n/a	n/a Max. temp. at t=0	5 min. MSCWLL	2.5 hr.	Same comment as for Case MNGPEPU2b, except this case is for CLTP. LPCI flow into vessel for this case begins at t=7 mins. (SP/T at this time is 100F).

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU3	Small Water Break LOCA and HPCI auto initiated • EPU power level • SLOCA (2" ID break in recirc suction line) at t=0 • All SRVs/SVs available for initial pressure control • HPCI auto initiates • Control HPCI flow to constant 1500 gpm after the first auto restart, and then continues to auto cycle • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify that HPCI can function as the only injection source for 24 hours during SLOCA • Verify that 1 train of SPC is sufficient for a non-ATWS scenario	N/A	N/A	N/A	N/A	24 hr.	HPCI first auto initiates at 30 sec. This case shows that HPCI can function as the only RPV injection source for a SLOCA for 24 hours for the EPU condition. SP/T=137F and SP/P=16.6psi at t=24 hrs.
MNGPEPU3x	Same as MNGPEPU3 except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	N/A	N/A	N/A	N/A	24 hr.	HPCI first auto initiates at 30 sec. This case shows that HPCI can function as the only RPV injection source for a SLOCA for 24 hours for the CLTP condition.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU4	Med. Water Break LOCA, HPCI and 1 LPCI pump • EPU power level • MLOCA .0873 ft^2 (4" ID water break in recirc suction line) at t=0 • HPCI auto initiates and auto cycles • No RPV Emergency Depressurization • Initiate 1 LPCI pump at LP interlock • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify viability of LP injection for MLOCA with HPCI initial injection and no RPV ED (MLOCA ET success criterion)	2 min.	13 min.	24 min. Max. temp. of 2000°F	2.5 min. MSCWLL	10 hr.	HPCI first auto initiates at 30 sec. LPCI flow > 0 at t=19 min. This case shows that RPV ED is not needed to allow LP ECCS for MLOCA for EPU if HPCI initially operates.
MNGPEPU4x	Same as MNGPEPU4 except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	2 min.	14 min.	20 min. Max. temp. of 1475°F	3 min. MSCWLL	10 hr.	HPCI first auto initiates at 30 sec. min. This case shows that RPV ED is not needed to allow LP ECCS for MLOCA for CLTP if HPCI initially operates.
MNGPEPU4a	Large Water Break LOCA, HPCI and 1 LPCI pump • EPU power level • LLOCA 4.27 ft^2 (28" ID water break in recirc suction line) at t=0 • No RPV Emergency Depressurization • Initiate 1 LPCI pump at LP interlock SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify viability of LPCI injection for LLOCA (LLOCA ET success criterion)	2 sec	21 sec	N/A	2 sec. MSCWLL	10 hr	LPCI flow > 0 at t=21 sec LPCI is a success for LLOCA case
MNGPEPU4ax	Same as MNGPEPU4a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	2 sec	30 sec.	20 min. Max. temp. of 1475°F	2 sec. MSCWLL	10 hr	LPCI flow > 0 at t=21 sec

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU5a	MSIV Closure, SORV, and only CRDH (Nominal flow, with delayed start of 2nd pump) available for injection • EPU power level • MSIV Closure at t=0 (no credit for FW coast down flow) • One (1) SORV • No injection other than 1 CRDH pump (no enhanced flow) available at t=0 • No RPV ED • Iterate to determine time when initiation of 2nd CRDH pump (no enhanced flow) prevents core damage • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify CRDH (Nominal flow, 2 pumps) success criteria for early injection for a Transient with an SORV and no RPV ED	17 min.	N/A	1.3 hr. Core Damage	27 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=1 min. CRDH (w/2 CRDH pumps at nominal flow) not a success as the only early injection source for EPU condition for transients w/SORV and no RPV ED.
MNGPEPU5ax	Same as MNGPEPU5a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	19 min.	N/A	1.7 hr. Max. temp. of 2160°F	29 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=40 min. Two CRDH pumps at nominal flow is a success as the only early injection source for CLTP condition for transient w/SORV and no RPV ED, as long as 2nd pump is initiated by t=40 min. CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU5b	Same as MNGPEPU5a except RPV ED (using only 1 additional SRV) at MSCWLL.	• Verify CRDH (Nominal flow, 2 pumps) success criteria for early injection for a Transient with an SORV and RPV ED	15 min.	34 min.	55 min. Max. temp. of 2040°F	23 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=37 min. Two CRDH pumps at nominal flow is a success as the only early injection source for EPU condition for transient w/SORV and RPV ED, as long as 2nd pump is initiated by t=37 min. The time conservatively used in the CLTP base PRA for alignment of a second CRDH pump is more restrictive than this result.
MNGPEPU5bx	Same as MNGPEPU5b except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	19 min.	43 min.	1.2 hr. Max. temp. of 2150°F	30 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=57 min. Two CRDH pumps at nominal flow is a success as the only early injection source for CLTP condition for transient w/SORV and RPV ED, as long as 2nd pump is initiated by t=57 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU5c	MSIV Closure and only CRDH (Nominal flow, with delayed start of 2nd pump) available for injection • EPU power level • MSIV Closure at t=0 (no credit for FW coast down flow) • No injection other than 1 CRDH pump (no enhanced flow) available at t=0 • No RPV ED • Iterate to determine time when initiation of 2nd CRDH pump (no enhanced flow) prevents core damage • SPC w/1 RHR train initiated at pool temp. 90°F(3)	• Verify CRDH (Nominal flow, 2 pumps) success criteria for early injection for a Transient without an SORV and no RPV ED	17 min.	N/A	1.3 hr. Core Damage	26 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=1 min. CRDH (w/2 CRDH pumps at nominal flow) not a success as the only early injection source for EPU condition for transients w/o SORV and no RPV ED.
MNGPEPU5cx	Same as MNGPEPU5c except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	20 min.	N/A	1.6 hr. Max. temp. of 2165°F	30 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=43 min. Two CRDH pumps at nominal flow is a success as the only early injection source for CLTP condition for transient w/o SORV and no RPV ED, as long as 2nd pump is initiated by t=43 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU5d	Same as MNGPEPU5c except RPV ED (using only 2 SRVs) at MSCWLL.	• Verify CRDH (Nominal flow, 2 pumps) success criteria for early injection for a Transient without an SORV and RPV ED	14 min.	29 min.	52 min. Max. temp. of 2100°F	23 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=26 min. Two CRDH pumps at nominal flow is a success as the only early injection source for EPU condition for transient w/o SORV and RPV ED, as long as 2nd pump is initiated by t=26 min. The time conservatively used in the CLTP base PRA for alignment of a second CRDH pump is more restrictive than this result.
MNGPEPU5dx	Same as MNGPEPU5d except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	20 min.	36 min.	1.0 hr. Max. temp. of 2185°F	29 min. MSCWLL	2.5 hr.	Second CRDH pump initiated at t=42 min. Two CRDH pumps at nominal flow is a success as the only early injection source for CLTP condition for transient w/o SORV and RPV ED, as long as 2nd pump is initiated by t=42 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU5e	MSIV Closure, SORV and only CRDH (Enhanced flow, 1 pump) available for injection • EPU power level • MSIV Closure at t=0 (no credit for FW coast down flow) • One (1) SORV • No injection other than 1 CRDH pump (no enhanced flow) available at t=0 • No RPV ED • Iterate to determine time when initiation of CRDH enhanced flow (still only one pump) prevents core damage • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify CRDH (Enhanced flow, 1 pump) success criteria for early injection for a Transient with an SORV and no RPV ED	15 min.	N/A	1.2 hr. Max. temp. of 1960°F	23 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=43 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for EPU condition for transient w/SORV and no RPV ED, as long as flow enhancement is initiated by t=43 min. The time conservatively used in the CLTP base PRA for alignment of enhanced CRDH is more restrictive than this result.
MNGPEPU5ex	Same as MNGPEPU5e except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	19 min.	63 min.	1.4 hr. Max. temp. of 2125°F	30 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=64 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for CLTP condition for transient w/SORV and no RPV ED, as long as flow enhancement is initiated by t=64 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU5f	Same as MNGPEPU5e except RPV ED (using only 1 additional SRV) at MSCWLL.	• Verify CRDH (Enhanced flow, 1 pump) success criteria for early injection for a Transient with an SORV and RPV ED	15 min.	34 min.	49 min. Max. temp. of 1950°F	23 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=41 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for EPU condition for transient w/SORV and RPV ED, as long as flow enhancement is initiated by t=41 min. The time conservatively used in the CLTP base PRA for alignment of enhanced CRDH is more restrictive than this result.
MNGPEPU5fx	Same as MNGPEPU5f except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	19 min.	43 min.	1.1 hr. Max. temp. of 2115°F	30 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=59 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for CLTP condition for transient w/SORV and RPV ED, as long as flow enhancement is initiated by t=59 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU5g	MSIV Closure and only CRDH (Enhanced flow, 1 pump) available for injection • EPU power level • MSIV Closure at t=0 (no credit for FW coast down flow) • No injection other than 1 CRDH pump (no enhanced flow) available at t=0 • No RPV ED • Iterate to determine time when initiation of CRDH enhanced flow (still only one pump) prevents core damage • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify CRDH (Enhanced flow, 1 pump) success criteria for early injection for a Transient without an SORV and no RPV ED	14 min.	N/A	1.2 hr. Max. temp. of 2075°F	22 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=44 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for EPU condition for transient w/o SORV and no RPV ED, as long as flow enhancement is initiated by t=44 min. The time conservatively used in the CLTP base PRA for alignment of enhanced CRDH is more restrictive than this result.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU5gx	Same as MNGPEPU5g except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	20 min.	34 min.	1.4 hr. Max. temp. of 2085°F	30 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=64 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for CLTP condition for transient w/o SORV and no RPV ED, as long as flow enhancement is initiated by t=64 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU5h	Same as MNGPEPU5g except RPV ED (using only 2 SRVs) at MSCWLL.	• Verify CRDH (Enhanced flow, 1 pump) success criteria for early injection for a Transient without an SORV and RPV ED at TAF	14 min.	28 min.	45 min. Max. temp. of 2060°F	22 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=34 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for EPU condition for transient w/o SORV and RPV ED, as long as flow enhancement is initiated by t=34 min. The time conservatively used in the CLTP base PRA for alignment of enhanced CRDH is more restrictive than this result.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU5hx	Same as MNGPEPU5h except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	20 min.	36 min.	56 min. Max. temp. of 2090°F	30 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=47 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for CLTP condition for transient w/o SORV and RPV ED, as long as flow enhancement is initiated by t=47 min. The CLTP PRA conservatively uses t=25 mins. based on a surrogate time of time to core damage for a SLOCA.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU5i	Same as MNGPEPU5h except RPV ED (using only 3 SRVs) at MSCWLL.	• Verify CRDH (Enhanced flow, 1 pump) success criteria for early injection for a Transient without an SORV and RPV ED at TAF	14 min.	27 min.	47 min. Max. temp. of 1975°F	22 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=34 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for EPU condition for transient w/o SORV and RPV ED, as long as flow enhancement is initiated by t=30 min. The time conservatively used in the CLTP base PRA for alignment of enhanced CRDH is more restrictive than this result.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU5ix	Same as MNGPEPU5i except Pre-EPU (CLTP) power of 1775 MWTh.	<Same as case above>	20 min.	34 min.	55 min. Max. temp. of 2060°F	30 min. MSCWLL	2.5 hr.	Enhanced CRDH flow initiated at t=45 min. Enhanced CRDH flow (1 CRDH pump) is a success as the only early injection source for EPU condition for transient w/o SORV and RPV ED, as long as flow enhancement is initiated by t=45 min. The time conservatively used in the CLTP base PRA for alignment of enhanced CRDH is more restrictive than this result.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU6a	Med. Water Break LOCA, No HP injection, delayed RPV ED and 1 LPCI pump • EPU power level • MLOCA .0873 ft^2 (4" ID water break in recirc suction line) at t=0 • No HP injection • Iterate to determine time when initiation of Emergency RPV Depressurization (using only 1 SRV) is successful to prevent core damage • Initiate 1 LPCI pump at LP interlock • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify time allowable for manual initiation of ADS during MLOCA with no HP injection	1 min.	7 min.	10 min. Max. temp. of 1118°F	2 min. MSCWLL	5 hr.	RPV ED initiated at t=7 min. (1/3 core height) with 1 SRV. LPCI flow into RPV > 0 at t=10 min. This case shows that manual RPV ED can be delayed for EPU until t=7 min. and prevent core damage for a MLOCA with loss of HP injection.
MNGPEPU6ax	Same as MNGPEPU6a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	1 min.	8 min.	12 min. Max. temp. of 1200°F	2 min. MSCWLL	5 hr.	RPV ED initiated at t=8 min. (1/3 core height) with 1 SRV. LPCI flow > 0 at t=11 min. This case shows that manual RPV ED can be delayed for CLTP until t=8 min. and prevent core damage for a MLOCA with loss of HP injection.

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LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU6b	Med. Water Break LOCA, no injection and no RPV ED • EPU power level • MLOCA .0873 ft^2 (4" ID water break in recirc suction line) at t=0 • No HP or LP injection • No RPV ED • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify time to core damage for MLOCA w/o RPV injection and w/o RPV ED	85 sec.	7 min.	17 min. Core Damage	117 sec.	5 hr.	This case shows that the time to core damage for a Med water break LOCA w/o RPV injection and w/o RPV ED is t=17 min. for the EPU condition.
MNGPEPU6bx	Same as MNGPEPU6b except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	97 sec.	8 min.	19 min. Core Damage	126 sec.	5 hr.	This case shows that the time to core damage for a Med water break LOCA w/o RPV injection and w/o RPV ED is t=19 min. for the CLTP condition.
MNGPEPU6c	Small Water Break LOCA, no injection and no RPV ED • EPU power level • SLOCA (2" ID water break in recirc suction line) at t=0 • No HP or LP injection • No RPV ED • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify time to core damage for SLOCA w/o RPV injection and w/o RPV ED	4 min.	14 min.	26 min. Core Damage	6 min.	5 hr.	This case shows that the time to core damage for a Small water break LOCA w/o RPV injection and w/o RPV ED is t=26 min. for the EPU condition.

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LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU6cx	Same as MNGPEPU6c except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	5 min.	16 min.	31 min. Core Damage	7 min.	5 hr.	This case shows that the time to core damage for a Small water break LOCA w/o RPV injection and w/o RPV ED is t=31 min. for the CLTP condition.
MNGPEPU7a	Isolation ATWS, No HP Injection, Level Control, delayed RPV ED, 1 SLC pump delayed, and 1 LPCI pump • EPU power level • MSIV Closure ATWS at t=0 (no FW coast down flow credited) • RPT (both pumps) at t=0 • No HP injection • RPV ED at 1/3 core height (using only 3 SRVs) • Initiate 1 LPCI pump at LP interlock and control level at TAF until SLC injection completed • SLC initiated at t=12 min • Increase level to normal RPV level after SLC achieves hot shutdown • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify 3 SRVs sufficient for RPV ED during an isolation ATWS (success criteria) • Verify time available to initiate RPV ED during an isolation ATWS with no HP injection	73 sec.	7.4 min.	13 min. Max. temp. of 1315°F	91 sec. MSCWLL	2.5 hr.	SLC initiated at 12 min. RPV ED initiated at t=7.4 min. (1/3 core height) with 3 SRVs This case shows that 3 SRVs (the current CLTP PRA success criterion for such scenarios) is still sufficient for the EPU condition, and that RPV ED can be delayed until approximately t=7.4 mins. during an isolation ATWS with no high pressure injection.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU7ax	Same as MNGPEPU7a except: • Pre-EPU (CLTP) power of 1775 MWth, and • SLC initiated at t=13.5 min. (time used in CLTP base PRA)	<Same as case above>	78 sec.	7.5 min.	13 min. Max. temp. of 1205°F	96 sec. MSCWLL	2.5 hr.	SLC initiated at 13.5 min. RPV ED initiated at t=7.5 min. (1/3 core height) with 3 SRVs This case shows that 3 SRVs (the current CLTP PRA success criterion for such scenarios) is sufficient for the CLTP condition, and that RPV ED can be delayed until approximately t=7.5 mins. during an isolation ATWS with no high pressure injection.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	Time of Run	Comments
MNGPEPU7b	Isolation ATWS, HPCI Injection, Level Control at TAF, RPV ED, 1 SLC pump and 1 LPCI pump and no SPC • EPU power level • MSIV Closure ATWS at t=0 (no FW coast down flow credited) • RPT (both pumps) at t=0 • All SVs/SRVs available for initial pressure transient • HPCI auto initiates, and then control level to TAF • SLC initiated at t=12 min • RPV ED (using only 3 SRVs) when HCTL reached • 1 LPCI pump when HPCI trips and continue to control level at TAF • No RHR SPC or venting available	• Determine time allowable for early SLC initiation action • Determine acceptable time frame for SPC initiation during ATWS scenario	73 sec.	n/a	n/a Max. temp. at t=0	91 sec. MSCWLL	2.5 hr.	SLC initiated at 12 min. RPV ED due to HCTL at t=16 min. SP/T=200F at t=50 mins. Differences in time to 200F between EPU and CLTP cases due to number of HPCI cycles occurring in MAAP as the code controls level around TAF.
MNGPEPU7bx	Same as MNGPEPU7b except: • Pre-EPU (CLTP) power of 1775 MWth, and • SLC initiated at t=13.5 min. (time used in CLTP base PRA)	<Same as case above>	80 sec.	n/a	n/a Max. temp. at t=0	96 sec. MSCWLL	2.5 hr.	SLC initiated at 13.5 min. RPV ED due to HCTL at t=17 min. SP/T=200F at t=46 mins. Differences in time to 200F between EPU and CLTP cases due to number of HPCI cycles occurring in MAAP as the code controls level around TAF

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU7c	Isolation ATWS, HPCI Injection, Level Control at normal, RPV ED, 1 SLC pump and 1 LPCI pump and no SPC • EPU power level • MSIV Closure ATWS at t=0 (no FW coast down flow credited) • RPT (both pumps) at t=0 • All SVs/SRVs available for initial pressure transient • HPCI auto initiates, and then control level at Normal • SLC initiated at t=12 min • RPV ED (using only 3 SRVs) when HCTL reached • 1 LPCI pump when HPCI trips and continue to control level at Normal • No RHR SPC or venting available	• Determine time allowable for early SLC initiation action • Determine acceptable time frame for SPC initiation during ATWS scenario	92 sec.	n/a	n/a Max. temp. at t=0	117 sec. MSCWLL	2.5 hr.	SLC initiated at 12 min. RPV ED due to HCTL at t=14 min. SP/T=200F at t=48 mins. Differences in time to 200F between EPU and CLTP cases due to number of HPCI cycles occurring in MAAP as the code controls level around TAF
MNGPEPU7cx	Same as MNGPEPU7c except: • Pre-EPU (CLTP) power of 1775 MWth, and • SLC initiated at t=13.5 min. (time used in CLTP base PRA)	<Same as case above>	105 sec.	n/a	n/a Max. temp. at t=0	144 sec. MSCWLL	2.5 hr.	SLC initiated at 13.5 min. RPV ED due to HCTL at t=15 min. SP/T=200F at t=44 mins. Differences in time to 200F between EPU and CLTP cases due to number of HPCI cycles occurring in MAAP as the code controls level around TAF

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8a	MSIV Closure, no injection and no RPV ED • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • No HP or LP injection • No RPV ED • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify time available to control FW flow (operator action FW-CNTRL-Y). HRA uses time to TAF for this action. • Verify time to core damage for a loss of injection HP core damage transient	12 min.	36 min.	50 min. Core Damage	17 min. MSCWLL	2.5 hr.	This case shows that the time to core damage for an isolation transient w/o RPV injection and w/o RPV ED is t=50 min. for the EPU condition.
MNGPEPU8ax	Same as MNGPEPU8a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	16 min.	43 min.	62 min. Core Damage	23 min. MSCWLL	2.5 hr.	This case shows that the time to core damage for an isolation transient w/o RPV injection and w/o RPV ED is t=62 min. for the CLTP condition.
MNGPEPU8b	MSIV Closure, no injection and no RPV ED • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • No HP or LP injection • RPV ED (using only 3 SRVs) at TAF • SPC w/1 RHR train initiated at pool temp. 90°F[3]	• Verify time to core damage for a loss of injection LP core damage transient	12 min.	17 min.	30 min. Core Damage	12 min. MSCWLL	2.5 hr.	RPV ED initiated at t=12 min. (TAF) with 3 SRVs. This case shows that the time to core damage for an isolation transient w/o RPV injection and w/RPV ED is t=30 min. for the EPU condition.

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LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8bx	Same as MNGPEPU8b except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	16 min.	21 min.	37 min. Core Damage	17 min. MSCWLL	2.5 hr.	RPV ED initiated at t=16 min. (TAF) with 3 SRVs. This case shows that the time to core damage for an isolation transient w/o RPV injection and w/RPV ED is t=37 min. for the EPU condition.
MNGPEPU8c	SBO, with RCIC, no OSP recovery, and no DFP injection alignment • EPU power level • SBO at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • Only RCIC available for injection, RCIC manual control to keep normal RPV level • RCIC fails at t=4 hrs due to battery depletion	• Verify time to core damage for SBO w/RCIC or HPCI and battery failure at t=4 hrs, to verify that OSP Recovery at t=6 hrs is still appropriate for EPU	5.3 hr.	6.1 hr.	6.6 hr. Core Damage	5.6 hr. MSCWLL	10 hr.	This case shows that the time to core damage is t=6.6 hrs for the EPU for an SBO, with initial RCIC or HPCI, and battery depletion at t=4hrs. If RCIC were allowed to auto cycle, the time to core damage would be longer as RCIC completes a vessel filling cycle just before t=4 hrs As such, the assumption in the CLTP PRA for OSP recovery required at t=6 hrs for such scenarios is still bounded by the EPU.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8cx	Same as MNGPEPU8c except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	5.5 hr.	6.4 hr.	7.0 hr. Core Damage	5.8 hr. MSCWLL	10 hr.	This case shows that the time to core damage is t=7.0 hrs for the CLTP for an SBO, with initial RCIC or HPCI, and battery depletion at t=4 hrs. The time to core damage may vary by approximately an hour depending upon RCIC level control; if RCIC were allowed to auto cycle and RCIC filled the vessel just prior to loss of DC at t=4hrs, then the time to core damage would be about an hour longer. The assumption in the CLTP PRA is that OSP recovery is required at t=6 hrs for such scenarios.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8d	SBO, with RCIC long-term (batteries being charged), no OSP recovery, and subsequent RCIC failure on high pool temperature • EPU power level • SBO at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • Only RCIC available for injection, suction from the pool only • RCIC manual control to keep normal RPV level • No RPV ED • RCIC fails when SP/T = 200F	• Verify time to core damage for SBO w/RCIC or HPCI long-term (batteries being charged), to verify that OSP Recovery at t=12 hrs is still appropriate for EPU	9.6 hr.	10.6 hr.	11.1 hr. Core Damage	6.1 hr. HCTL 9.9 hr. MSCWLL	15 hr.	RCIC fails on high pool temperature (SP/T=220F) at t=8.1 hrs. This case shows that the time to core damage is t=11.1 hrs for the EPU for an SBO, w/RCIC or HPCI long-term (batteries being charged) but subsequent failure on high pool temperature. Although the time to core damage may be extended if RCIC were allowed to auto cycle, the EPU risk assessment assumes that level will be controlled manually. As such, the assumption in the CLTP PRA for OSP recovery required at t=12 hrs is adjusted in the EPU risk assessment to t=11 hrs. for these sequences.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8dx	Same as MNGPEPU8d except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	11.7 hr.	12.9 hr.	13.6 hr. Core Damage	7.5 hr. HCTL 12.1 hr. MSCWLL	16 hr.	RCIC fails on high pool temperature (SP/T=220F) at t=9.9 hrs. This case shows that the time to core damage is t=13.6 hrs for the CLTP for an SBO, w/RCIC or HPCI long-term (batteries being charged) but subsequent failure on high pool temperature. Similar to comment in case MNGPEPU6cx, the time to core damage may vary by approximately an hour for this case depending upon the mode of RCIC level control. The assumption in the CLTP PRA is that OSP recovery is required at t=12 hrs for such scenarios.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8e	SBO, w/SORV and HPCI • EPU power level • SBO at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • Only HPCI available for injection, HPCI manual control to keep normal RPV level • One (1) SORV • HPCI subsequently fails on low steam pressure	• Verify time to core damage for SBO w/HPCI and a SORV, to verify that OSP Recovery at t=3 hrs is still appropriate for EPU	3.2 hr.	4.6 hr.	4.9 hr. Core Damage	3.4 hr. MSCWLL	5 hr.	HPCI trips on low steam pressure at t=2.8 hrs. This case shows that the time to core damage is t=4.9 hrs for the EPU for an SBO, with HPCI and an SORV. As such, the assumption in the CLTP PRA for OSP recovery required at t=3 hrs for such scenarios is still bounded by the EPU. The EPU actually stretches the time that HPCI operates before tripping on low steam pressure.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU8ex	Same as MNGPEPU8e except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	2.8 hr.	4.3 hr.	4.7 hr. Core Damage	2.9 hr. MSCWLL	5 hr.	HPCI trips on low steam pressure at t=1.9 hrs. This case shows that the time to core damage is t=4.7 hrs for the CLTP for an SBO, with HPCI and an SORV. Similar to comment in case MNGPEPU6cx, the time to core damage may vary by approximately 30-60 minutes in this case depending upon the mode of HPCI level control. The assumption in the CLTP PRA is that OSP recovery is required at t=3 hrs for such scenarios.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU9	Transient with loss of containment heat removal • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • HPCI only injection source initially • RPV ED (using only 3 SRVs) on HCTL • 1 LPCI pump initiated at LP interlock • CRDH only injection source after SRVs re-close on high containment pressure (and RPV repressurizes) • No RHR SPC or containment venting available	• Identify time frames for containment venting, RHR SPC initiation, and ultimate containment failure due to overpressure	N/A	N/A	N/A	6.6 hr. HCTL	48 hr.	SP/T reaches 200F (HPCI, RCIC failure point in PRA) at t=6.8 hrs. SRVs closed due to Hi DW press at 31 hr. at 75 psig in the DW. Containment failure (at 118 psia) occurs at t=43 hr. Very long time available to operators in which to align SPC or initiate emergency containment vent in order to prevent loss of injection (either due to SRV re-closure, or high pool temperature) and containment overpressure failure. Operator action (CHR-DET-Y) for initiation of SPC during transients based on time to SP/T=200F.

Table E-1

LEVEL 1 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	Time of Run	Comments
MNGPEPU9x	Same as MNGPEPU9 except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	N/A	N/A	N/A	7.7 hr. HCTL	48 hr.	SP/T reaches 200F (HPCI, RCIC failure point in PRA) at t=8.0 hrs. SRVs closed due to Hi DW press at 40 hr. at 75 psig in the DW. Containment failure (at 118 psia) occurs after t=48 hrs (98 psia at t-48 hrs.). Very long time available to operators in which to align SPC or initiate emergency containment vent in order to prevent loss of injection (either due to SRV re-closure, or high pool temperature) and containment overpressure failure. Operator action (CHR-DET-Y) for initiation of SPC during transients based on time to SP/T=200F. MNGP CLTP base PRA assumes that time is approximately t=10hrs

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10a	Large Late Release Scenario, Class ID, RPV breach, no DW injection, no DW shell failure, later DW thermal failure[6] • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • No HP or LP injection for vessel injection or debris cooling • Delayed RPV ED (using only 3 SRVs) at onset of core damage • DW fails when TGDW>700°F • Credit reactor building in reducing release magnitudes • No SPC or venting	• Verify that EPU does not cause Large Late release sequences to become Large EARLY (LERF)	33 min.	1.0 hr.	1.3 hr.	41 min. MCSWLL	7.9 (curve rising sharply at end of run and will exceed CsI LARGE Magnitude threshold shortly after end of run	87	5.9	1.3	40 hr.	Performed using CLTP "no-inj-lowP.inp" MAAP 4.0.4 input deck and then increasing Rx power to EPU. This is one of the two fastest progressing Large/Late sequences (most Large/Late sequences begin to release many hours later). CsI release to environment expected to exceed CsI 10% threshold (LARGE magnitude in MNGP PRA) soon after end of run. Cont. fails (and release from containment begins) when DW/T >700F at t=8.9 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~7-8 hrs after the declaration (LATE release in MNGP PRA). Case shows that Large/Late releases do not become Large/Early for EPU.

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF(4)	Time to Reach 1/3 Core Height(5)	Time to Max Core Temp or Time to >2200 °F(1)	Time HCTL(2) Exceeded or MSCWLL(4)	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10ax	Same as MNGPEPU10a except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	38 min.	1.2 hr.	1.5 hr.	48 min. MCSWLL	4.1 (curve rising sharply at end of run and will exceed CsI LARGE Magnitude threshold shortly after end of run	87	2.9	1.4	40 hr.	Performed using CLTP "no-inj-lowP.inp" MAAP 4.0.4 input deck. This is one of the two fastest progressing Large/Late sequences (most Large/Late sequences begin to release many hours later). CsI release to environment expected to exceed CsI 10% threshold (LARGE magnitude in MNGP PRA) soon after end of run. Cont. fails (and release from containment begins) when DW/T >700F at t=10.3 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~8-9 hrs after the declaration (LATE release in MNGP PRA).

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10b	Large Late Release Scenario, Class IA, RPV breach, no DW injection, no DW shell failure, later DW thermal failure[6] • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • No HP or LP injection for vessel injection or debris cooling • No RPV ED • DW fails when TGDW>700°F • Credit reactor building in reducing release magnitudes • No SPC or venting	• Verify that EPU does not cause Large Late release sequences to become Large EARLY (LERF)	33 min.	1.0 hr.	1.3 hr.	41 min. MCSWLL	45	86	34	1.3	40 hr.	Performed using CLTP "no-inj-highP.inp" MAAP 4.0.4 input deck and then increasing Rx power to EPU. This is one of the two fastest progressing Large/Late sequences (most Large/Late sequences begin to release many hours later). CsI release to environment is 34% (HIGH magnitude in MNGP PRA). Cont. fails (and release from containment begins) when DW/T >700F at t=7.8 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~6-7 hrs after the declaration (LATE release in MNGP PRA). Case shows that Large/Late releases do not become Large/Early for EPU.

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10bx	Same as MNGPEPU10b except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	38 min.	1.2 hr.	1.5 hr.	48 min. MCSWLL	30	86	22	1.4	40 hr.	Performed using CLTP "no-inj-highP.inp" MAAP 4.0.4 input deck. This is one of the two fastest progressing Large/Late sequences (most Large/Late sequences begin to release many hours later). CsI release to environment is 22% (LARGE magnitude in MNGP PRA). Cont. fails (and release from containment begins) when DW/T >700F at t=8.9 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~7-8 hrs after the declaration (LATE release in MNGP PRA).

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10c	Medium Early Release Scenario, Class ID, RPV breach, DW shell failure[6] • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • No HP or LP injection for vessel injection or debris cooling • Delayed RPV ED (using only 3 SRVs) at onset of core damage • DW steel shell failure occurs 7 mins. after RPV breach • Credit reactor building in reducing release magnitudes • No SPC or venting	• Verify that EPU does not cause Medium Early release sequences to become LARGE Early (LERF)	12 min.	35 min.	49 min.	17 min. MCSWLL	7.1	66	4.2	1.7	40 hr.	Performed using CLTP "none-lowP-dw-early.inp" MAAP 4.0.4 input deck and then increasing Rx power to EPU. Cont. fails (and release from containment begins) when DW shell melt-thru occurs at t=3.7 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~2-3 hrs after the declaration (EARLY release in MNGP PRA). CsI release to environment is 4.2% (MEDIUM magnitude in MNGP PRA). Case shows that Medium/Early releases do not become Large/Early for EPU.

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10cx	Same as MNGPEPU10c except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	15 min.	43 min.	1.0 hr.	22 min. MCSWLL	6.2	63	3.0	2.0	40 hr.	Performed using CLTP "none-lowP-dw-early.inp" MAAP 4.0.4 input deck. Cont. fails (and release from containment begins) when DW shell melt-thru occurs at t=4.9 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~3-4 hrs after the declaration (EARLY release in MNGP PRA). CsI release to environment is 3.0% (MEDIUM magnitude in MNGP PRA).

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10d	Medium Early Release Scenario, Class IA, RPV breach, containment flooding w/DW vent[6] • EPU power level • MSIV Closure at t=0 (no FW coast down flow credited) • All SVs/SRVs available for initial pressure transient • No HP or LP injection for vessel injection or debris cooling • No RPV ED • Initiate containment flooding at time of RPV breach • Initiate DW vent at 67 psi and maintain between 57-67 psi • Credit reactor building in reducing release magnitudes • No SPC	• Verify that EPU does not cause Medium Early release sequences to become LARGE Early (LERF)	12 min.	35 min.	49 min.	17 min. MCSWLL	9.1	50	4.9	1.9	40 hr.	Performed using CLTP "floodPB-highP-dw56.inp" MAAP 4.0.4 input deck and then increasing Rx power to EPU. Cont. fails (and release from containment begins) when DW vent is initiated during containment flooding at t=2.8 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~1-2 hrs after the declaration (EARLY release in MNGP PRA). CsI release to environment is 4.9% (MEDIUM magnitude in MNGP PRA). Case shows that Medium/Early releases do not become Large/Early for EPU.

Table E-2

LEVEL 2 PRA MAAP RUNS FOR MONTICELLO EXTENDED POWER UPRATE

Case ID	MAAP Run Description	Purpose	Time to TAF[4]	Time to Reach 1/3 Core Height[5]	Time to Max Core Temp or Time to >2200 °F[1]	Time HCTL[2] Exceeded or MSCWLL[4]	CsI Rel from Cont (%)	NG Rel (%)	CsI Rel to Env (%)	RB Decon Factor	Time of Run	Comments
MNGPEPU10dx	Same as MNGPEPU10d except Pre-EPU (CLTP) power of 1775 MWth.	<Same as case above>	15 min.	43 min.	1.0 hr.	22 min. MCSWLL	5.8	35	3.3	1.7	40 hr.	Performed using CLTP "floodPB-highP-dw56.inp" MAAP 4.0.4 input deck. Cont. fails (and release from containment begins) when DW vent is initiated during containment flooding at t=3.7 hrs. Per MNGP EAL procedure A.2-101, Rev. 38, declaration of Gen. Emergency would occur in t=1-2 hrs for this sequence. Release to environment occurs ~2-3 hrs after the declaration (EARLY release in MNGP PRA). CsI release to environment is 3.3% (MEDIUM magnitude in MNGP PRA).

Notes to Tables E-1 and E-2:

(1) Core damage is defined in the MNGP PRA MAAP runs as 2200°F in the core (based on the MAAP variable TCRHOT).

(2) The suppression pool Heat Capacity Temperature Limit, HCTL, is one of the key parameters (along with low RPV water level) requiring RPV Emergency Depressurization per the EOPs.

(3) The MAAP parameter file initiates SPC no earlier than t=15 mins to account for various issues such as operator focus on other tasks. As such, the directives in these input decks that state SPC initiation at a pool temperature of 90F means that SPC initiation occurs at t=15 mins (i.e., the pool is assumed to start at 85F at t=0 per the MNGP MAAP parameter file and it reaches 90F before t=15 mins for all isolation scenarios, thus SPC alignment occurs at the earliest allowed time point of t=15 mins.).

(4) The time to TAF (Top of Active Fuel, -126" at MNGP) shown in this table is based on the MAAP variable XWSH (water level in the shroud), and is indicative of level indication available to the operator. The same variable is used in this table for MSCWLL (Minimum Steam Cooling Water Level Limit, -149" at MNGP).

(5) The time to 1/3 core height in this table is based on the MAAP variable XWCOR (2-phase water level in the core).

(6) The Level 1 MAAP runs are performed using MNGP MAAP version 4.0.6 parameter file. The Level 2 MAAP release runs are performed using the MNGP MAAP Version 4.0.4 parameter file to be consistent with the MAAP runs for the release categorizations used by MNGP in the development of the Level 2 PRA release categorizations.

< Print-outs of MAAP input decks and graphs contained in file *'7740-495 MNGP EPU AppE attch.pdf'* >

Appendix F

COP SENSITIVITY

Appendix F
COP SENSITIVITY

This sensitivity study assesses the impact on plant risk if containment accident pressure is assumed not present (e.g., postulated pre-existing primary containment failure) during the postulated accident scenarios such that inadequate LP ECCS pump NPSH occurs.

F.1 RISK ASSESSMENT APPROACH

This risk assessment is performed by modification and quantification of the at-power internal events MNGP EPU base PRA model, and using the risk assessment guidance of NRC RG 1.174.

The performance of the COP risk assessment is best described by the following major analytical steps:

- Assessment of NPSH calculations
- Estimation of pre-existing containment failure probability
- Analysis of relevant plant experience data
- Manipulation and quantification of PRA models
- Comparison to ΔCDF and ΔLERF RG 1.174 acceptance guidelines
- Performance of uncertainty and sensitivity analyses

These steps are discussed below.

F.2 ASSESSMENT OF NPSH CALCULATIONS

The purpose of this task is to develop an understanding of the MNGP EPU NPSH calculations that result in the need to credit containment accident pressure for DBA LOCA accident scenarios.

The NPSH calculations are reviewed to understand the scenarios of interest that require COP credit to determine how best to modify the PRA models.

Two general approaches to PRA modeling of COP credit exist depending upon the number and types of NPSH calculations available:

1. Use of sensitivity studies of DBA NPSH calculations
2. Use of NPSH results from Monte Carlo process

The second approach is used here.

A Monte Carlo statistical analysis was performed by GE (using the SHEX code) of the containment response and associated NPSH calculations to produce a 95/95 result for the available NPSH in a given accident scenario. The 95/95 point represents the 95% confidence level that the available NPSH is greater than the calculated Monte Carlo result with a 0.95 probability (or, that there is only a 0.05 probability that the available NPSH is lower than the 95/95 point).

The Monte Carlo NPSH results are used to define a single PRA basic event with a probability based on the Monte Carlo result. The basic event represents the probability that initial plant conditions (i.e., high initial suppression pool temperature, high UHS temperature, etc.) exist at the onset of the modeled DBA scenarios such that inadequate LP ECCS NPSH is available.

For the purpose of modeling the conditions of inadequate NPSH, the PRA is interested in the probability of the plant conditions such that NPSHa is less than NPSHr (i.e., the fraction of the NPSH spectrum below the NPSHr point). The 95/95 point for NPSHa from the Monte Carlo analysis is not directly usable (i.e., to use directly as a 0.05 probability basic event) in the PRA logic modeling unless it coincidentally equals NPSHr.

As the result of the Monte Carlo analysis is a single 95/95 NPSH point rather than a cumulative probability distribution as a function of NPSH, engineering judgment is used (based on review of the NPSH Monte Carlo results) to assign a basic event probability for each DBA LLOCA COP scenario (refer to Table F-1 for descriptions of these three scenarios) that initial plant conditions exist at the onset of the scenarios such that inadequate LP ECCS NPSH is available. The estimated probabilities are as follows:

- Scenario #1: 1.0E-1
- Scenario #2: 5.0E-1
- Scenario #3: 1.0E-1

For Scenario #1, the probability that plant conditions will result in inadequate NPSH is known to be some value higher than 5E-2 (i.e., because the 95/95 NPSHa point is below NPSHr). As the calculated NPSHa 95/95 point is comparatively close (1-2 ft.) to NPSHr in the short time frame modeled for Scenario #1, a nominal probability of 1E-1 is estimated for this basic event. The same results apply to Scenario #3 (i.e., the calculated NPSHa 95/95 point is comparatively close to NPSHr).

For Scenario #2, the calculated NPSHa 95/95 point is much lower (by a factor of 2-3) than it is for Scenarios #1 and #3. As such, a nominal probability of 5E-1 is used for Scenario #2.

As shown later with a quantitative sensitivity case, exact values for the above probabilities are not necessary in showing that the COP risk impact is "very small".

The three scenarios are summarized in Table F-1. As can be seen from Table F-1, Scenarios #1 and #3 may be modeled together as a single scenario because the impact of LPCI Loop Select Logic single-failure does not change the conclusion that COP credit is required in approximately 7 mins. and that throttling LP ECCS will preclude the need

for COP credit. Therefore, the scenario modeling in the PRA for the DBA LOCA is as follows:

- Scenario #1 / #3: (Large LOCA Initiator) x (SPC Not Initiated Within t=10 min.) x (Containment Isolation fails at t=0) x (Operators Fail to Throttle LP ECCS Flow Within t=10 min.) x (Probability that Existing Plant Conditions Result in Inadequate NPSH) x (Probability that LP ECCS Pumps Fail Due to Inadequate NPSH)
- Scenario #2: (Large LOCA Initiator) x (One Division ECCS Available) x (SPC Not Initiated Within t=10 min.) x (Containment Isolation fails at t=0) x (Probability that Existing Plant Conditions Result in Inadequate NPSH) x (Probability that LP ECCS Pumps Fail Due to Inadequate NPSH)

The modeling of these scenarios in the PRA is discussed later in Section F.5

F.3 ESTIMATION OF PRE-EXISTING CONTAINMENT FAILURE PROBABILITY

This task involves defining the size of a pre-existing containment failure pathway to be used in the analysis to defeat the COP credit, and then quantifying the probability of occurrence of the un-isolable pre-existing containment failure. The approach to this input parameter calculation will follow EPRI guidelines regarding calculation of pre-existing containment leakage probabilities in support of integrated leak rate test (ILRT) frequency extension LARs (i.e., EPRI Report 1009325, Risk Impact of Extended Integrated Leak Rate Testing Intervals, 2005). This is the same approach used in the Vermont Yankee EPU COP analyses presented to the ACRS in November and December 2005.

Containment failures that may be postulated to defeat the containment accident pressure credit include containment isolation system failures and pre-existing unisolable containment leakage pathways. The pre-existing containment failure may be one that only manifests as the containment pressurizes.

Containment isolation system failures are already modeled in the MNGP PRA containment isolation fault tree used in the Level 2 PRA. These failures include failures on demand and failures of valves to remain closed during the standard 24-hr PRA mission time. A basic event for pre-existing containment leakage was added to the MNGP containment isolation fault tree (both in the pre-EPU and EPU base models) for this assessment.

The pre-existing containment leakage probability may be obtained from EPRI 1009325, Risk Impact of Assessment of Extended Integrated Leak Rate Testing Intervals. EPRI 1009325 provides a framework for assessing the risk impact for extending integrated leak rate test (ILRT) surveillance intervals. EPRI 1009325 includes a compilation of industry containment leakage events, from which an assessment was performed of the likelihood of a pre-existing unisolable containment leakage pathway.

A total of seventy-one (71) containment leakage or degraded liner events were compiled. Approximately half (32 of the 71 events) had identified leakage rates of less than or equal to 1La (i.e., the Technical Specification containment allowed leakage rate). None of the 71 events had identified leakage rates greater than 21La. EPRI 1009325 employed industry experts to review and categorize the industry events, and then various statistical methods were used to assess the data.

The EPRI 1009325 study uses 100La as a conservative estimate of the leakage size that would represent a large early release pathway consistent with the LERF risk measure, but estimated that leakages of 600La or greater are a more realistic representation of a large early release. The COP risk assessment for the Vermont Yankee Mark I BWR plant, presented to the ACRS in November and December 2005, determined a leakage size of 27La using the conservative 10CFR50, Appendix K containment analysis approach. Earlier ILRT industry guidance (NEI Interim Guidance) conservatively recommended use of 10La to represent "small" containment leakages and 35La to represent "large" containment leakages.

This analysis is not concerned per se about the size of a leakage pathway that would represent a LERF release, but rather a leakage size that would defeat the containment accident pressure credit. Given the low likelihood of such a leakage, the exact size is not key to this risk assessment, and no detailed calculation of the exact hole size is performed here. A sensitivity study discussed later assesses the sensitivity of the results to the pre-existing leakage size assumption.

Given the above, the base analysis here assumes 20La as the size of a pre-existing containment leakage pathway sufficient to defeat the containment accident pressure credit. Such a hole size does not realistically represent a LERF release (based on EPRI 1009325) and is also believed (based on the VY hole size estimate) to be on the low end of a hole size that would preclude containment accident pressure credit. The probability of a 20La pre-existing containment leakage at any given time at power is 1.88E-03.

This low likelihood of a significant pre-existing containment leakage path is consistent with MNGP primary containment performance experience. The MNGP primary containment performance experience shows MNGP containment leakages much less than 1La.

F.4 ANALYSIS OF RELEVANT PLANT EXPERIENCE DATA

An unisolated primary containment is not the only determining factor in defeating low pressure ECCS pump NPSH. Variations in MNGP UHS and suppression pool water temperatures, suppression pool level and RHR heat exchanger "K" value were statistically analyzed. The purpose of this data assessment is to estimate realistic probabilities that UHS water temperature, suppression pool level and temperature, and heat exchanger effectiveness will exceed a given value, i.e. the probability of

exceedance. These values are used as input into the Monte Carlo simulations of the available NPSH and in the risk assessment.

F.5 MANIPULATION AND QUANTIFICATION OF PRA MODELS

This task is to make the necessary modifications to the PRA models to simulate the loss of low pressure ECCS pumps during a Large LOCA. Large LOCA initiated sequences in the PRA are modified as appropriate to mirror the DBA accident calculations requiring COP credit. Accident sequences involving Interfacing Systems LOCAs and other LOCAs Outside Containment are not adjusted in this risk assessment because such LOCAs result in deposition of decay heat directly outside the containment and not into the suppression pool.

PRA Model Modifications

The modifications made to the MNGP PRA to model the COP credit for DBA LOCA scenarios are shown in Figure F-1. Pages 1 and 2 of Figure F-1 show the DBA LOCA COP credit scenario logic developed under a sub-tree that is input into the CS and LPCI fault tree logic. Page 3 of Figure F-1 shows the pre-existing containment leakage basic event added to the containment isolation fault tree. As can be seen in Figure F-1, the new logic is ANDed with the large LOCA initiator to ensure that the logic applies to large LOCA initiated accident sequences.

The basic event stating that SPC is not initiated within t=10 minutes is conservatively assigned a 1.0 probability, and reflects the assumption in the DBA LLOCA short-term scenario.

The two basic events that model the probability that plant conditions at the time of the DBA LOCA contribute to inadequate LP ECCS are based on the discussions in Section F.2.

The human error probability basic event for operator failure to throttle LP ECCS is calculated using the same human reliability analysis methodology (i.e., NUREG/CR-4772) used in the MNGP PRA:

- Per the plant EOPs and operator training, the operators will throttle ECCS flow as necessary per NPSH curves existing on the EOP flowcharts
- The time of the initial cue to the operators for the need to throttle ECCS flow is estimated at t=5 mins. for Scenarios 1 & 3. This is the point at which available head is nearing NPSHr and which flow fluctuations may be notable to the operator.
- The end of the available time window to the operator is conservatively estimated at t=10 mins. and is the time at which pump head collapse is assumed to occur. This time is judged conservative.
- Manipulating LP ECCS pump flow is a manual action performed at the main control panels in the control room. The time required to travel to the proper panel(s) and perform the flow manipulation is estimated at 1 min.
- Therefore, the available diagnosis time to the operator is (10 min. – 5 min.) – 1 min. = 4 mins.
- Using the MNGP PRA HRA Methodology (i.e., NUREG/CR-4772), the diagnosis error contribution for a diagnosis time frame of 4 mins. is 2.5E-1; and the manipulation error rate for performing the action is 5E-3. The total HEP for failure to throttle is 2.55E-1.

In conditions of inadequate NPSH, the pumps will experience surging and cavitation but will not necessarily fail. However, this analysis conservatively assumes the low pressure ECCS pumps fail with a probability of 1.0 given inadequate NPSH and failure to throttle.

The probability of an unisolated containment at the time of the accident is modeled using the MNGP containment isolation fault tree. The probability of the pre-existing leakage basic event is discussed previously in Section F.3 and is based on an assumed hole size of 20La.

Scenario #2 involves failures that result in only one available ECCS division. Those failures are a LOOP combined with failure of one division of ECCS (the DBA single failure is assumed to be an EDG, but the PRA recognizes that it could also be a bus or ECCS equipment failures):

- The conditional probability of a LOOP given a LOCA initiator is 2.4E-2, based on USNRC Memorandum to Mark A. Cunningham, Chief from Alan S. Kuritzky, "Transmittal of Preliminary Staff and Contractor Comments on EPRI Expert Elicitation Meeting on the Probability of LOOP Given Large LOCA", June 14, 2002.
- Failure of one division of ECCS is modeled as failure of Division 1 "OR" Division 2 ECCS. Each division is modeled with an undeveloped basic event with a probability of 1E-1. This 1E-1 probability covers failure of one EDG (a contribution of approximately 5E-2), failure of the associated safety bus (a negligible contribution), and failures for one division of ECCS pumps and valves (a contribution of approximately 5E-3), and is judged conservative.

PRA Model Quantification

The Level 1 (core damage) PRA is then quantified using the standard quantification techniques of the base PRA. The impact on the Level 2 LERF accident sequences are conservatively modeled here with the assumption that the COP credit failure scenarios lead directly to a LERF release. As such, the ΔLERF is assumed to equal the calculated ΔCDF.

The size of the assumed containment hole used in the pre-existing containment leakage basic event is conservatively small (i.e., BWR PRAs typically use a 2" diameter hole in the primary containment to represent the minimum size of a LERF release pathway, and a 2" diameter hole is much greater than the 20La equivalent hole size used in the base calculation). In addition, the location of the assumed containment leakage pathway has an impact on LERF. If the containment leakage pathway is assumed to exist in the

wetwell airspace then the post-accident releases from the containment would be scrubbed by the suppression pool and thus not result in a LERF magnitude release. This analysis conservatively assumes that the containment leakage pathway is such that, given a core damage event, the conditional probability of a LERF release is 1.0. The impact of this conservative assumption on ΔLERF does not change the overall conclusion that the risk impact of COP credit is very small.

F.6 COMPARISON TO ΔCDF AND ΔLERF RG 1.174 ACCEPTANCE GUIDELINES

The revised MNGP PRA models are quantified to determine the change in the base CDF. As discussed above in Section F.5, the change in LERF is assumed to equal the change in CDF.

The RG 1.174 ΔCDF and ΔLERF risk acceptance guidelines are summarized in Figures F-2 and F-3, respectively. The boundaries between regions are not necessarily interpreted by the NRC as definitive lines that determine the acceptance or non-acceptance of proposed license amendment requests; however, increasing delta risk is associated with increasing regulatory scrutiny and expectations of compensatory actions and other related risk mitigation strategies.

The risk impact results for EPU COP credit for DBA LOCAs is:

- ΔCDF = 9.0E-9
- ΔLERF = 9.0E-9

Both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region.

These impacts are referenced with respect to the base modeling assumption that no COP credit is required for LP ECCS adequate NPSH during DBA LOCA scenarios. If

the base model where revised to include modeling of the existing COP credit already allowed at MNGP, the change in risk for the additional COP credit required by the EPU would be even smaller.

F.7 UNCERTAINTY AND SENSITIVITY ANALYSES

To provide additional information for the decision making process, this sensitivity risk assessment is supplemented by parametric uncertainty analysis and quantitative and qualitative sensitivity studies to assess the sensitivity of the calculated risk results.

Uncertainty is typically categorized into the following three types, consistent with PRA industry literature:

- Parametric
- Modeling
- Completeness

Parametric uncertainties are those related to the values of the fundamental parameters of the PRA model, such as equipment failure rates, initiating event frequencies, and human error probabilities. Typical of standard industry practices, the parametric uncertainty aspect is assessed by performing a Monte Carlo parametric uncertainty propagation analysis. Probability distributions are assigned to each parameter value in the PRA, and a Monte Carlo sampling code is used to sample each parameter and propagate the parametric distributions through to the final results.

Modeling uncertainty is focused on the structure and assumptions inherent in the risk model. The structure of mathematical models used to represent scenarios and phenomena of interest is a source of uncertainty, due to the fact that models are a simplified representation of a real-world system. Model uncertainty is addressed here by the identification and quantification of focused sensitivity studies.

Completeness uncertainty is primarily concerned with scope limitations. Scope limitations are addressed here by the qualitative assessment of the impact on the conclusions if external events and shutdown risk contributors are also considered.

F.7.1 PARAMETRIC UNCERTAINTY ANALYSIS

The MNGP PRA is not currently constructed to allow parametric uncertainty analysis; as such, parametric uncertainty analysis was not performed. However, based on knowledge of the issues, the COP risk impact, and PRA parametric uncertainty assessments, the results of a parametric uncertainty analysis would not change the conclusion that the risk impact of COP credit for DBA LOCAs is "very small" per RG 1.174.

F.7.2 MODELING UNCERTAINTY ANALYSIS

As stated previously, modeling uncertainty is concerned with the sensitivity of the results due to uncertainties in the structure and assumptions in the logic model. EPRI has developed a guideline for modeling uncertainty that takes the rational approach of identifying key sources of modeling uncertainty and then performing appropriate sensitivity calculations. This approach is taken here.

The modeling issues selected here for assessment are those related to the risk assessment of the containment accident pressure credit. This assessment does not involve investigating modeling uncertainty with regard to the overall base PRA. The modeling issues identified for sensitivity analysis are:

- Pre-existing containment leakage size and associated probability
- Calculation of containment isolation system failure
- Probability of plant conditions contributing to inadequate NPSH

- Large LOCA initiators in the PRA
- HEP for failure to throttle LP ECCS

Sensitivity Case 1: Pre-Existing Containment Leakage Size/Probability

The base case analysis assumes a pre-existing containment leakage pathway leakage size of 20La that would result in defeat of the necessary containment accident pressure credit.

A larger pre-existing leak size of 100La, consistent with the EPRI 1009325 recommended assumption for a "large" leak, is used in this sensitivity to defeat the necessary COP credit. From EPRI 1009325, the probability of a pre-existing 100La containment leakage pathway at any given time at power is 2.47E-04.

Sensitivity Case 2: Calculation of Containment Isolation System Failure

The base case quantification uses the containment isolation system fault tree logic to represent failure of the containment isolation system. The fault tree specifically analyzes primary containment penetrations greater than 2" diameter. This modeling sensitivity case expands the scope of the containment isolation fault tree to include additional smaller lines as potential defeats of COP credit. This sensitivity is quantified by multiplying by a factor of 10 the probability contribution in the containment isolation fault tree from isolation failures of penetration lines in response to a containment isolation signal.

Sensitivity Case 3: Probability of Plant Conditions Contributing to Inadequate NPSH

The basic event probabilities for the different scenarios that plant conditions at the time of the DBA LOCA contribute to inadequate LP ECCS are based on the discussions in Section F.2. As previously discussed, precise estimates of these probabilities are not necessary to show that the risk impact of COP credit for LP ECCS NPSH is very small. This fact is shown by this sensitivity. This sensitivity is performed assuming that plant conditions (e.g., high initial suppression pool temperature, high UHS temperature, etc.) contributing to inadequate NPSH exist 100% of the time.

Sensitivity Case 4: Large LOCA Initiators in the PRA

The MNGP PRA has a single "Large LOCA" initiator in the PRA, and this initiator was used to represent the DBA LOCA scenarios. However, in addition to the "Large LOCA" initiator, the MNGP PRA also contains an initiator for "RPV Rupture" and LOCA-induced scenarios (i.e., Transient initiators with failure of SRVs to actuate). This sensitivity case includes the "RPV Rupture" initiator and the LOCA-induced scenarios in the COP credit risk assessment. The impact on the base results is negligible.

Sensitivity Case 5: LP ECCS Throttling HEP

The base analysis uses a human error probability, HEP, of 2.55E-1 for failure to throttle LP ECCS to avoid pump failure due to inadequate NPSH. This HEP is based on the plant specific timings from the thermal hydraulic calculations and the human reliability analysis methodology used in the MNGP PRA. This sensitivity study conservatively assumes that the HEP for failure to throttle LP ECCS is 1.0.

Summary of Modeling Uncertainty Results

The results of these sensitivity studies are as follows:

Case	ΔCDF	ΔLERF
Sensitivity Case 1	1.2E-9	1.2E-9
Sensitivity Case 2	1.4E-8	1.4E-8
Sensitivity Case 3	8.4E-8	8.4E-8
Sensitivity Case 4	9.0E-9	9.0E-9
Sensitivity Case 5	3.3E-8	3.3E-8

The above sensitivity studies do not change the base conclusions that the risk impact of COP credit for a DBA LOCA is "very small" per RG 1.174.

F.7.3 COMPLETENESS UNCERTAINTY ANALYSIS

As stated previously, completeness uncertainty is addressed here by the qualitative assessment of the impact on the conclusions if special events, external events and shutdown risk contributors are also considered.

ATWS

The risk impact of COP credit for low pressure ECCS pump NPSH during ATWS scenarios can be assessed with the following representative ATWS scenario:

- Initiator: Isolation event
- Failure to scram
- Successful RPV level/power control
- Containment isolation failure at t=0
- Only one division of ECCS available
- Operators fail to throttle ECCS pumps

An isolation event is one that results in isolation of the RPV from the main condenser heat sink. Based on NUREG/CR-6928, the industry average frequency for such an event is approximately 2E-1/yr. Based on the various isolation initiating events (e.g., MSIV Closure, Loss of Condenser Vacuum, etc.) modeled in the MNGP PRA, the frequency of such an event at MNGP is approximately 3E-1/yr. The frequency of 3E-1/yr is used in this analysis.

The probability of scram failure in the MNGP PRA is 5.9E-6. This probability is consistent with other current BWR industry PRAs.

The sum of ARI, RPT, SLC, and operator level control and ADS inhibit action failures is generally in the range of 0.1 to 0.2 for industry BWR PRAs , which would result in a probability of successful level/power control in the 0.8 to 0.9 range. This analysis conservatively assumes the probability of successful level/power control is 1.0. Failure of level/power control would result in a scenario which would lead to core damage regardless of COP credit issues; therefore, such scenarios are not part of this assessment.

The probability of containment isolation failure at t=0 is approximately 2E-3.

As discussed earlier in the base case analysis of this risk assessment, the failure probability for one division of ECCS is approximately 5E-3.

The same human error probability of 2.55E-1 used in the base analysis for failure to throttle the ECCS pumps is assumed here.

The risk impact for such a scenario is calculated as:

$$3E\text{-}1 \times 5.9E\text{-}6 \times 1.0 \times 2E\text{-}3 \times 5E\text{-}3 \times 2.55E\text{-}1 = 4.5E\text{-}12/yr$$

Even if this representative ATWS scenario were to require only that the containment be unisolated (i.e., failure of one division of ECCS not assumed and throttling not a success path), the accident sequence frequency would still be a non-significant 3.5E-9/yr.

Postulating this additional scenario would not change the conclusion that the risk impact of COP credit is "very small" per RG 1.174.

SBO

The risk impact of COP credit for low pressure ECCS pump NPSH during SBO scenarios can be assessed with the following representative SBO scenario:

- Initiator: Loss of Offsite Power
- Failure of all EDGs
- One SBO capable injection source successfully operates
- Containment isolation failure at t=0
- Offsite AC power recovered at t=4hrs (the MNGP SBO coping period)
- Alignment of SPC at t=4hrs

Note that the above is an extension of the SBO Rule sequence (i.e., MNGP does not require COP for the SBO 4-hr coping period).

Based on NUREG/CR-6928, the industry average frequency for loss of offsite power is approximately 4E-2/yr. The LOOP initiator frequency in the MNGP PRA is 2.28E-2/yr. The frequency of 4E-2/yr is used in this analysis.

As discussed previously for the base case analysis, the failure probability of one EDG is approximately 5E-2. Failure of all EDGs is estimated here using a common cause failure approach and assuming a 5% failure of all EDGs given failure of one EDG. The

5% common cause failure probability is conservative (industry average is in the 2-3% range). Therefore, the failure of all EDGs is estimated at 5E-2 x 0.05 = 2.5E-3.

This analysis assumes that the probability of a SBO capable injection source (e.g., RCIC) successfully operating for the SBO coping period is 1.0.

The probability of containment isolation failure at t=0 is approximately 2E-3.

Based on NUREG/CR-6890, Reevaluation of Station Blackout Risk at Nuclear Power Plants, the industry average exceedance probability for successfully recovering offsite AC at 4 hours following a LOOP at power is approximately 0.84. Failure of AC power recovery would result in a scenario which would lead to core damage regardless of COP credit issues; therefore, such scenarios are not part of this assessment.

This analysis assumes that the probability of alignment of ECCS pumps to the suppression pool immediately following offsite AC recovery is 1.0. This assessment also assumes that throttling of the pumps will not prevent the inadequate NPSH condition and that the pumps fail with a probability of 1.0 once they are aligned to the pool.

This SBO scenario conservatively does not credit other injection systems (e.g., RCIC from the CST or DFP; alternate RPV injection sources) that would be available after 4 hrs when offsite AC power is recovered.

The risk impact for such a scenario is calculated as:

$$4E\text{-}2 \times 2.5E\text{-}3 \times 1.0 \times 2E\text{-}3 \times 0.84 \times 1.0 = 1.7E\text{-}7/yr$$

Postulating this additional scenario would not change the conclusion that the risk impact of COP credit is "very small" per RG 1.174 for the Core Damage Frequency risk metric.

The Large Early Release Frequency risk metric is just above the border of the "very small" and "small" region. Relaxation of the excess conservatisms in the LERF modeling (e.g., recognizing that loss of low pressure ECCS at t=4hrs does not directly result in a LERF release) would show that LERF risk metric is also clearly in the "very small" region of RG 1.174.

Seismic

The change in plant risk due to seismic-induced large LOCA COP scenarios is non-significant and likely undetectable with current state of the technology seismic PRA. The COP credit scenarios require one or more RHR pumps to be in operation (i.e., the PRA already models core damage accident sequences in which loss of all RHR pumps causes loss of LP ECCS – due to the need to initiate emergency containment venting) and the containment to fail.

A seismic event severe enough to fail the primary containment will also fail, with a much higher likelihood, the RHR system. Another aspect is in the modeling of like component failures in a seismic PRA. In a seismic PRA, like components located on the same elevation (e.g., RHR pumps) are commonly modeled as all failed given one fails. As such, if a seismic event fails an RHR pump (with some probability that varies depending upon the seismic magnitude), a seismic PRA will fail all the RHR pumps. As such, the likelihood of a seismic scenario that fails the containment yet fails only 2 or 3 out of the four RHR pumps is a very low likelihood scenario. As a final point on this issue, very high magnitude earthquakes become moot for this issue, as they would result in failure of key buildings and structures and lead directly to core damage.

As such, seismic issues do not impact the decision making for containment accident pressure credit.

Internal Fires

COP credit for the DBA LOCA scenario is necessary, among other aspects, due to the large heat addition to the suppression pool during the blowdown. An internal fire induced large LOCA type scenario (i.e., a scenario with large heat addition to the suppression pool and no high pressure injection sources available) can be postulated as follows:

- Initiator: Fire in main control panel initiates ADS [OR] fire-induced isolation event with subsequent multiple stuck open relief valves
- Containment isolation failure at t=0
- Plant conditions at time of event contribute to inadequate NPSH
- Operators fail to throttle ECCS pumps

The fire induced initiator can be estimated at 1E-4/yr. A fire in the main control panel that initiates ADS is in the 1E-6/yr to 1E-4/yr range using current industry fire initiator techniques. A fire induced isolation transient with subsequent multiple SORVs would also be in the 1E-6/yr to 1E-4/yr range (i.e., the sum of all fire-induced isolation transients would be in the 1E-2/yr to 1E-1/yr range, and the probability of multiple SORVs given an isolation transient is approximately 1E-4 to 1E-3). Therefore, the sum of both these two fire scenarios is estimated at 1E-4/yr.

The probability of containment isolation failure at t=0 is approximately 2E-3. This analysis does not assume that this fire scenario also results in fire-induced containment isolation failure. A fire in the control room that causes both a fire-induced ADS actuation and fire-induced containment isolation failure would involve fires initiating in separate control panels at the same time (an extremely low likelihood scenario). A postulated fire scenario in which a fire initiates in one panel and then the operators fail to suppress the fire such that it spreads to multiple panels would be modeled in a fire

PRA as a control room evacuation scenario and would lead to core damage with a high conditional probability regardless of COP credit impacts.

The same probability of 0.1 used for Scenario #1 for plant conditions at the time of the event that contribute to inadequate NPSH can be reasonably used here.

Likewise, the same human error probability of 2.55E-1 used in the analysis for failure to throttle the ECCS pumps can also be assumed here. Use of this HEP assumes that the timing for the need for COP credit in this scenario occurs as fast for this fire-induced SORV event as it does for the DBA LOCA.

The risk impact for such a scenario is calculated as: 1E-4 x 2E-3 x 1E-1 x 2.55E-1 = 5.1E-9/yr

Although not a DBA LOCA, postulating this additional scenario would not change the conclusion that the risk impact of COP credit for a DBA LOCA is "very small" per RG 1.174.

Other External Hazards

In addition to seismic events and internal fires, the other following external hazard categories exist:

- High Winds/Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Other External Hazards

The NRC IPEEE Program has generally determined that these other external hazard categories are not significant risk contributors. As such, these other external hazards

are judged not to significantly impact the decision making for containment accident pressure credit.

Shutdown Risk

The credit for containment accident pressure is not required for accident sequences occurring during shutdown. As such, shutdown risk does not influence the decision making for containment accident pressure credit.

F.8 COP RISK ASSESSMENT CONCLUSIONS

The risk impact results for COP credit for LP ECCS NPSH for DBA LOCAs is:

- ΔCDF = 9.0E-9
- ΔLERF = 9.0E-9

Both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region. These impacts are referenced with respect to the base modeling assumption that no COP credit is required for LP ECCS adequate NPSH during DBA LOCA scenarios. If the base model where revised to include modeling of any existing COP credit already allowed at the plant, the change in risk for the additional COP credit required by an EPU (or other LAR) would be even smaller.

Sensitivity studies show that even assuming plant conditions (e.g., high suppression pool temperature, high UHS temperature, etc.) contributing to inadequate NPSH exist 100% of the time results in a "very small" calculated risk impact.

The results for COP credit for DBA LOCA scenarios are orders of magnitude below the upper threshold of the RG 1.174 "very small" risk increase region. Even if COP credit were assumed required for DBA LOCAs, special events, and external events, the

conservative and simplified calculations in this analysis shows that the overall impact (i.e., summing the impacts of COP credit for all such accidents) would still remain within the 'very small" risk increase region of RG 1.174 for Core Damage Frequency and just above the border of the "very small" and "small" region for Large Early Release Frequency. Relaxation of the excess conservatisms in the LERF modeling in this analysis (e.g., recognizing that loss of low pressure ECCS at t=4hrs does not result in a LERF release) would show that LERF risk metric is also clearly in the "very small" region even when COP credit impacts for DBA LOCAs, special events, and external events are summed.

Table F-1

Summary of Understanding of MNGP DBA LLOCA NPSH Issues
(Assuming no COP Exists)

DBA LLOCA Scenario	IE	Single Failure	@ t = 0 min.			@ t = 10 min.			Time COP Required	Time of "Head Collapse"	Comment
			LP ECCS Pumps Injecting	ECCS Throttled	# Loops of SPC	LP ECCS Pumps Injecting	ECCS Throttled	# Loops of SPC			
#1	DBA LOCA	LPCI Loop Select Logic	6 (4 LPCI, 2 CS)	No	0	n/a	n/a	n/a	t=420 sec (7 min.) (NPSHa ~1-2 ft. below NPSHr)	t=10 min. (judged conservative)	• Throttling LP ECCS prior to t=10 min. will restore adequate NPSH • Scenario #1 and #3 can be modeled together as need for COP credit occurs at approximately same time, throttling will preclude need, and whether or not LPCI loop select logic fails does not impact this result
#2	DBA LOCA	One Division Emergency AC	3 (2 LPCI, 1 CS)	No	0	1 (1 CS)	Yes	1 (1 RHR pump, 1 Hx, 1 RHRSW pump)	t=8160 sec (135 min.) (NPSHa ~2-6 ft. below NPSHr)	t=13560 s (226 min.)	• Need for COP credit occurs in late time frame • LP ECCS already throttled (i.e., throttling LP ECCS does not preclude need for COP credit)
#3	DBA LOCA	Containment Isolation	6 (4 LPCI, 2 CS)	No	0	2 (2 CS)	Yes	2 (2 RHR pumps per loop, 1 Hx per loop, 2 RHRSW pumps per Hx)	t=440 sec (7.3 min.) (NPSHa ~1-2 ft. below NPSHr)	t=10 min. (judged conservative)	• Throttling LP ECCS prior to t=10 min. will restore adequate NPSH • Scenario #1 and #3 can be modeled together as need for COP credit occurs at approximately same time, throttling will preclude need, and whether or not LPCI loop select logic fails does not impact this result

Figure F-1 (1 of 3)

DBA LOCA Scenario Modifications Made to MNGP PRA for COP Sensitivity

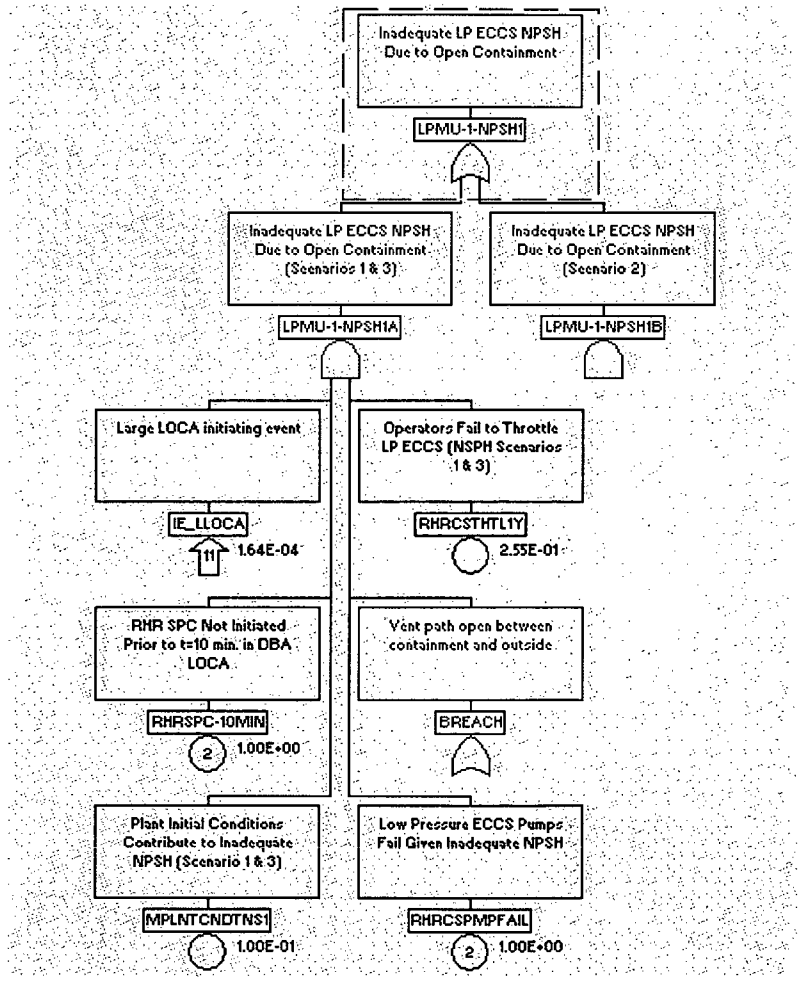


Figure F-1 (2 of 3)

DBA LOCA Scenario Modifications Made to MNGP PRA for COP Sensitivity

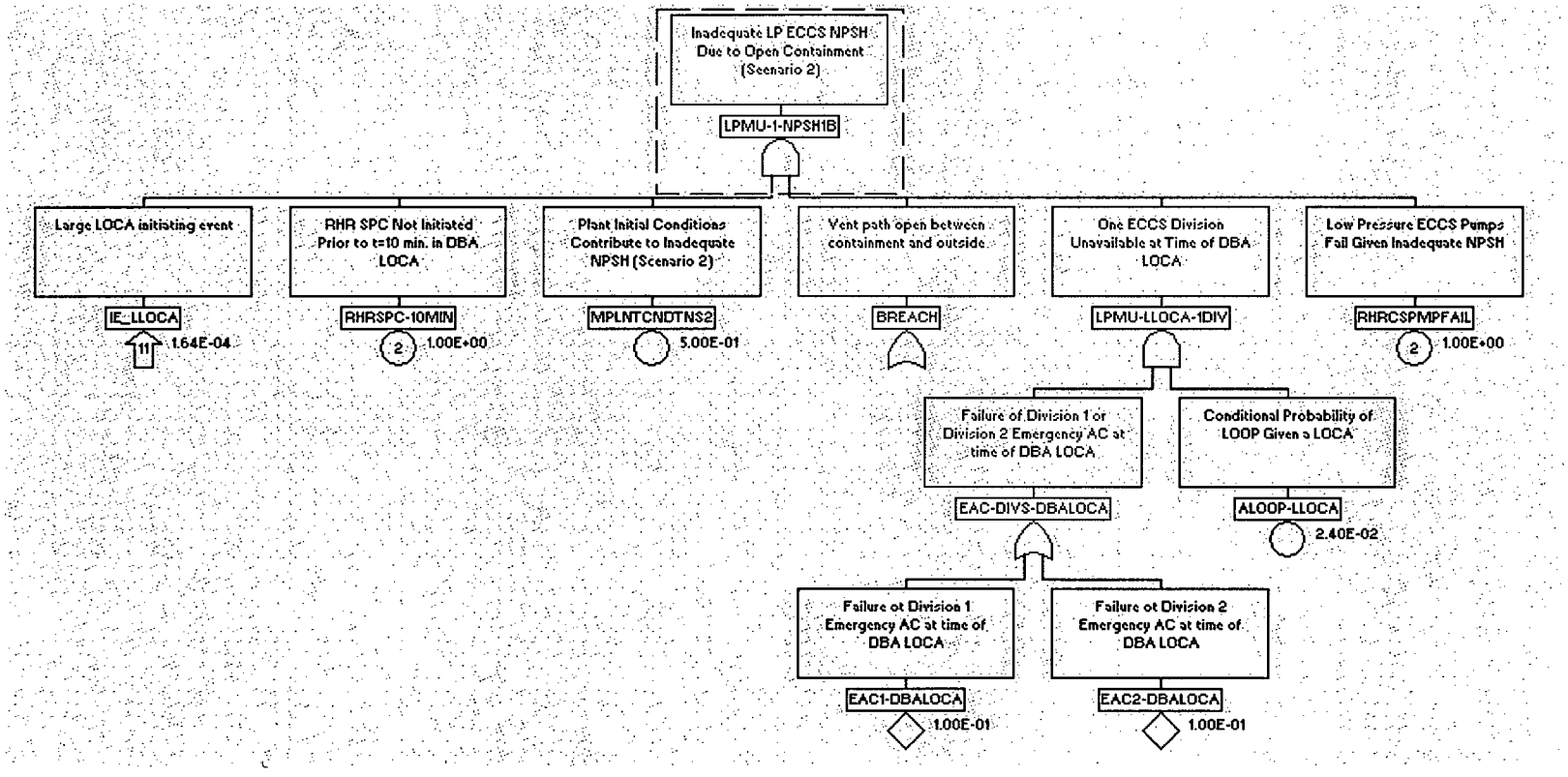


Figure F-1 (3 of 3)

DBA LOCA Scenario Modifications Made to MNGP PRA for COP Sensitivity

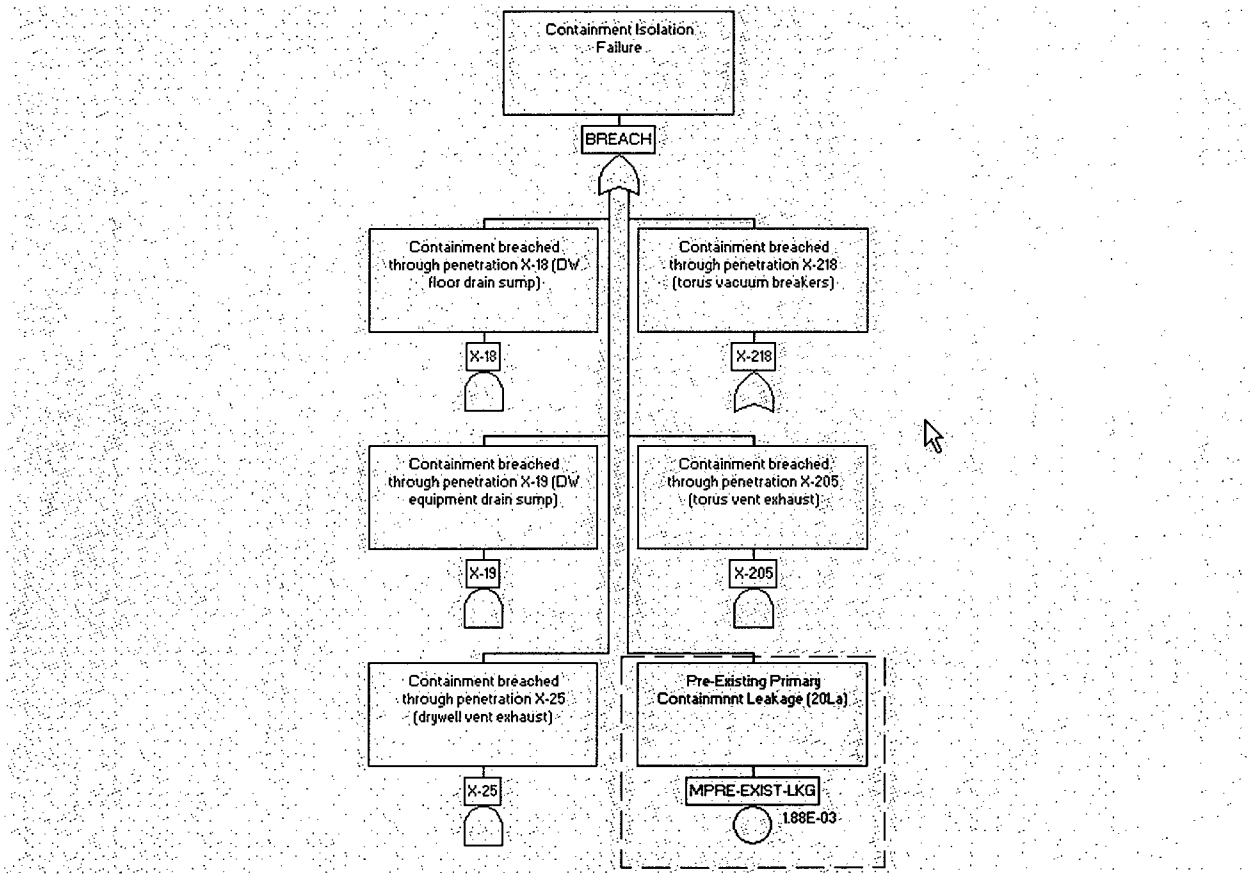


Figure F-2

RG 1.174 Delta CDF Risk Acceptance Guidelines

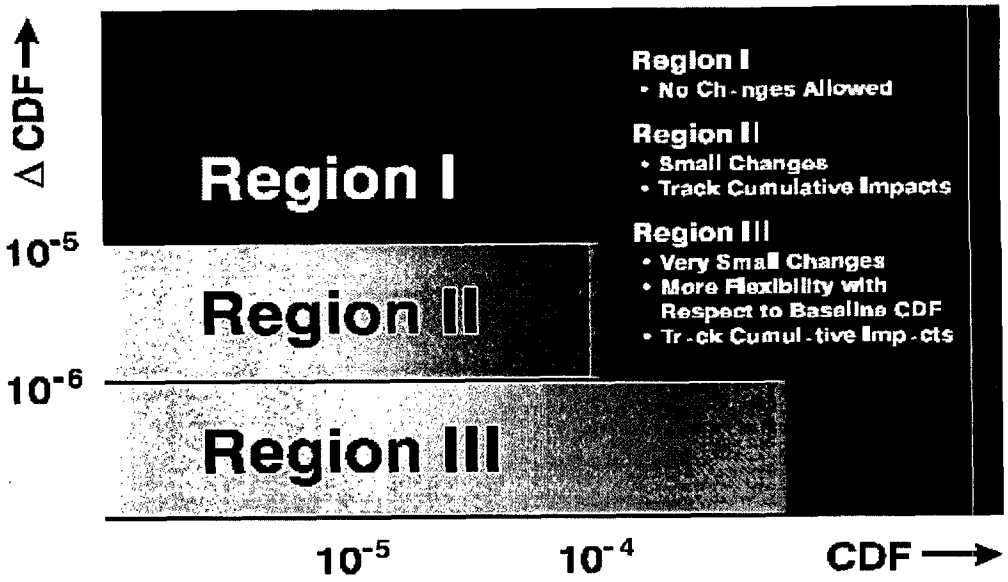
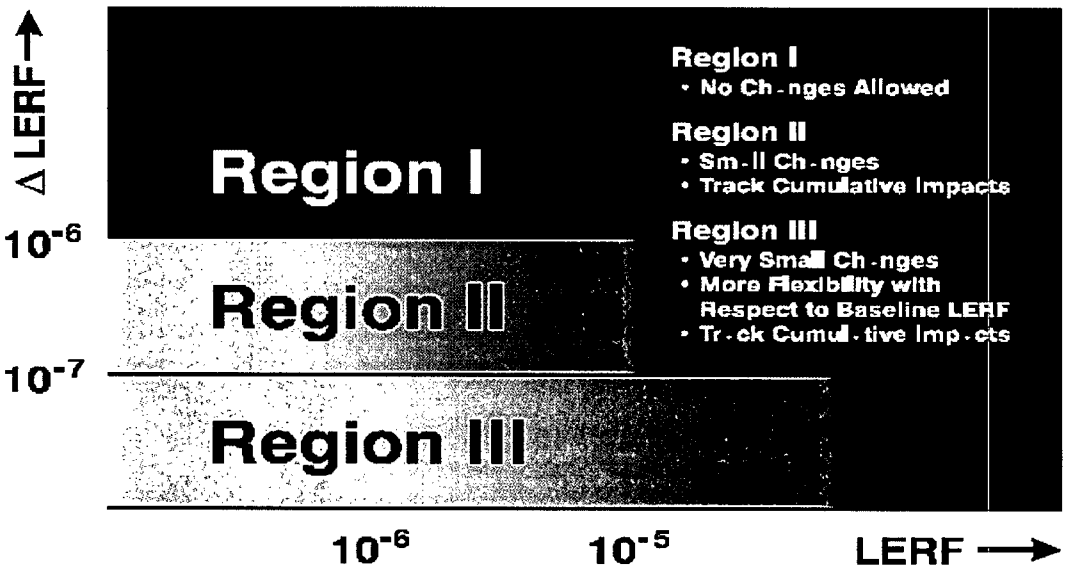


Figure F-3

RG 1.174 Delta LERF Risk Acceptance Guidelines



Enclosure 16 to L-MT-08-052

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

ENCLOSURE 16

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Enclosure 16 documents questions posed by NRC reviewers during their review of NMC's (now NSPM) EPU LAR submittal dated March 31, 2008.

This table is provided to aid the NRC in review of NSPM's EPU resubmittal. It should be noted that NSPM acceptance review responses to NRC questions associated with the steam dryer contained in this enclosure may have been superseded by information contained in Enclosure 11.

The NSPM response in L-MT-08-042 that addresses the question from the EEEB concerning equipment qualification has changed. The revised response is provided in Enclosure 17 to the EPU LAR resubmittal.

ENCLOSURE 16

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
1) RERB – ENVIRONMENTAL REVIEW BRANCH (ML081490281)		
Per yesterday meeting with applicant I would like to request document "2006 MNGP ALARA Report " that was cited in Enclosure 4 to L-MT-08-018 "MNGP extended power uprate Environmental Assessment" under table 7.2.1-1 "Exposure history (in REM) from 2006 MNGP ALARA Report", p. 50 of 69.	L-MT-08-039	ML081490639
In sections 6.1.6.1 and 6.2.2 of Enclosure 4, the Environmental Assessment provides a description of the Higgins' eye pearlymussel, a freshwater mollusk, which is a federally endangered species that is located in the Mississippi River in the vicinity of the Monticello Nuclear Generating Plant (MNGP). Its range has been reduced to 50% of historic levels, and is limited to the Mississippi River and three of its tributaries. The pearlymussel is susceptible to entrainment and impingement in its early life stages, including the male gamete and larval (glochidia) stages, both of which are found in river currents. In paragraph 6 of section 6.2.8 (Impingement and Entrainment) sentence one states "Extended power uprate does not effect the impingement and entrainment of organisms ...." However, with an increase of the average annual water intake from the current water withdrawal rate of 509 cubic feet/second (cfs) to the maximum annual average surface water appropriation limit of 645 cfs (Section 6.2.2- Surface Water Appropriation), which is greater than a 25% increase in water withdrawal, a strong inference can be made that this increase in water withdrawal will correspondingly lead to a greater than 25% increase in pearlymussel early life stage mortality within the vicinity of MNGP. This would contradict the statement in section 6.2.8 quoted above that no organisms will be affected by impingement and entrainment. While current permits allow for an increase in average water withdrawal, this will still result in an increase in the average annual water withdrawal from the Mississippi River, and a corresponding increase in the mortality of the federally protected juvenile Higgins' eye pearlymussel. Do you have data to evaluate what impact this increased water withdrawal will have on the population of the Higgins' eye pearlymussel?	L-MT-08-039	ML081490639

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Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
Section 6.2.4 (Increase in Circulating Water Discharge Temperature) describes the thermal impacts associated with an increased discharge temperature of 4.5 degrees F, stating that "The slight discharge canal temperature increase will not result in one half of the surface width of the river temperature exceeding the 90 degree F maximum...", and "... water temperatures downstream are not high enough to harm aquatic species or impede fish migration even in summer months." In section 6.2.6 (Mississippi River Thermal Plume) it is stated "... roughly 30 to 70 percent of the river is unaffected by the heated discharge. This also means that up to 70% of the river width is affected by current heat discharges. And section 6.2.7 (Cold Shock) notes that compliance with State water quality standards was not possible under extreme summer flows. The thermal plume has been noted to extend six kilometers downstream of the plant. With an increase of 4.5 degrees F for thermal discharges, it appears that there can be increases in the length of the thermal plume, increases to the percent of the river affected by the heated discharge beyond the current 70°h, and an increase in non-compliance with State water quality standards, which contradicts several of your findings in section 10.0 (Conclusions). Please address these concerns.	L-MT-08-039	ML081490639
2) CPTB – COMPONENT PERFORMANCE AND TESTING BRANCH (ML081490281)		
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.	L-MT-08-039	ML081490639
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.	L-MT-08-039	ML081490639
Describe activities and lessons learned programs that are dedicated to the enhancement of MOVs and AOVs performance/design basis review, and testing programs.	L-MT-08-039	ML081490639
Provide an evaluation of EPU impact on the functional	L-MT-08-039	ML081490639

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Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
design of safety-related pumps, and EPU impact on the IST program for pumps and valves.		
Precedents approved -SPU Amendment Request for Millstone 3. (Section 2.2.4) (ADAMS #ML072000386) -SPU Amendment Request for Comanche Peak, (Section 2.2.4) (ADAMS #ML072490131)	L-MT-08-039	ML081490639
3) CSGB – SG Tube Integrity and Chemical Engineering Branch (ML081490281)		
Protective Coating Systems (Paints) - Organic Materials Protective Coating Systems (Paints) - Organic Materials The applicant should identify the conditions (temperature, pressure, radiological dose) used to qualify Service Level I protective coatings in containment for current operating conditions and assess whether they remain bounding for DBA conditions following the extended power uprate.	L-MT-08-039	ML081490639
Flow-Accelerated Corrosion The applicant should provide a sample list of components for which wall thinning is predicted and measured by ultrasonic testing or other methods in order to assess the accuracy of the FAC predictions from CHECWORKS. This list should also include the initial wall thickness (nominal), current (measured) wall thickness, and a comparison of the measured wall thickness to the thickness predicted by the CHECWORKS FAC model. The applicant should identify those systems that are expected to experience the greatest increase in wear as a result of power uprate and the effect of individual process variables (i.e., moisture content, temperature, oxygen, and flow velocity) on each system identified. For the most susceptible systems and components, the applicant should provide the total predicted increase in wear rate due to FAC as a result of power uprate conditions.	L-MT-08-039	ML081490639
4) Electrical Engineering Branch (EEEB) (ML081490281)		
In Section 2.3 of the LAR under the section titled 'Outside Containment', the licensee stated the following: "The total integrated doses (normal plus accident) for EPU conditions were evaluated and determined not to adversely affect qualification of most of the EQ equipment located outside of containment. Equipment not qualified to the new environmental conditions at EPU will be reanalyzed, re-qualified, or replaced prior to implementation of EPU."	L-MT-08-039 L-MT-08-042 (The response provided in L-MT-042 has been revised and is provided in Enclosure 17.)	ML081490639 ML081570467 (The response provided in L-MT-042 has been revised and is provided in Enclosure 17.)

ENCLOSURE 16

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
In order for the Electrical Engineering Branch (EEEB) to start its review, the full EQ analysis must be completed. This includes any reanalysis, re-qualification, or replacement of equipment. The licensee must also describe how the equipment was evaluated (e.g., calculations, assessments, etc.) and show how the equipment remains bounded (i.e., provide the original design parameters and the updated values including the supporting calculations).		
For each topic in Section 2.3 of the LAR, the licensee consistently concludes that systems, structures, and components continue to remain bounded by existing analyses. In order for EEEB to start its review, the licensee must demonstrate how the analyses for the SSCs remain bounding (i.e., provide the original design parameters and the updated values including the supporting calculations). Additionally, the licensee also must provide more detailed information as to how the SSCs were evaluated.	L-MT-08-039	ML081490639
In Section 2.3 of the LAR (Specifically Sections 2.3.3 and 2.3.4), the licensee stated that some equipment may change. In order for EEEB to start its review, the licensee must provide assurance that all required plant modifications are accounted for in its EPU application.	L-MT-08-039	ML081490639
In Section 2.3 of the LAR, the licensee consistently notes that conditions do not change significantly as a result of EPU. In order for EEEB to start its review, the licensee must quantify the changes in conditions as a result of the proposed EPU.	L-MT-08-039	ML081490639
5) Reactor Systems Branch - SRXB (ML081490281)		
The SRXB issue with Rod Drop Accident is as follows: Appendix B to SRP Section 4.2, Revision 3, provides new acceptance criteria for the "reactivity initiated accident," i.e., the Control Rod Drop Accident. The acceptance criteria are given in terms of peak radial average fuel enthalpy and fuel rod internal pressure for low-power events with respect to high cladding temperature. At greater than 5% thermal power, the criterion is based on CPR. For pellet clad metal interaction, fuel failure criteria are expressed in terms of radial average fuel enthalpy and fuel hydrogen content. The acceptance	L-MT-08-039	ML081490639

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Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
criteria are below the previously accepted 170 cal/g for fuel failure, and significantly below the design limit of 280 cal/g.		
6) Mechanical & Civil Engineering Branch - EMCB (ML081490281)		
The analysis does not account for Finite Element mesh bias and uncertainty errors consistent with those accepted by the staff in previous applications. The FE bias and uncertainty errors were established from the bench-marking of Hope Creek FE analysis to the shaker test results. Why should Monticello's FE analysis be any different?	L-MT-08-040	ML081550504
The dryer was considered structurally adequate despite the fact that the minimum alternating stress ratio is less than 2, which the staff and ACRS consider as the threshold for acceptance due to the limited validation of the ACM Code. Why didn't the applicant consider dryer structural modifications and improvements to increase the minimum stress ratio to a magnitude higher than 2?	L-MT-08-040	ML081550504
The application does not include information on operating history, location of flaws and cracks that currently exist in the steam dryer, and the root causes for such cracks. Furthermore, the application does not address the effect of EPU on the integrity of the dryer in the presence of existing cracks.	L-MT-08-040	ML081550504
Insufficient details were provided regarding the establishment of the Main Steam Lines time histories that are used to define the dryer loads. How were they established? What is their length? How did the applicant determine that they are conservative?	L-MT-08-040	ML081550504
The application does not address the strong spectral (PSD) peak around 100 Hz (for outer hood nodes 7 and 99). The application should clarify the source and nature of the strong peak at these nodes.	L-MT-08-040	ML081550504
The application does not include the procedure employed for noise signal removal. No information was provided on whether only fictitious tones due to ACM error are removed from dryer loads? It is unclear why the alternating stress ratios dropped so significantly by 50% when noise is removed.	L-MT-08-040	ML081550504
The application does not include information on the mode shapes of the dryer at and near peak frequencies including 25-26 Hz, 154 Hz, and 162 Hz	L-MT-08-040	ML081550504
A bump up factor of 1.39 was used to scale stresses and loads from CLTP to EPU. This factor appears non conservative considering that a bump up factor at about 2.1 would more appropriately capture the potential valve resonance frequency near 162 Hz.	L-MT-08-040	ML081550504

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Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
7) Accident Analysis & Dose - AADB (ML081490281)		
Request to review accident analyses calculations as per the telephone conference between NRC and NMC on May 15, 2008.	L-MT-08-036	ML081430494
8) Reactor Inspection Branch		
Onsite Radiation Levels Provide the radiation levels prior to EPU and at EPU for the areas described in Table 2.10-1 and 2.10-2. Describe the methodology used to determine EPU radiation levels.	L-MT-08-041	ML081550640
Onsite Radiation Levels Describe the radiation surveys to be performed as part of the startup testing plan.	L-MT-08-041	ML081550640
Onsite Radiation Levels Describe the contribution and effects of hydrogen water chemistry (HWC) (N-16) to the radiation doses (both pre-EPU and post-EPU) to members of the public onsite.	L-MT-08-041	ML081550640
Off-Site Radiation Levels Provide the dose value contributions for the primary sources of normal operation offsite doses (all effluent releases, gamma shine, storage and transfer of radioactive materials) to a member of the public at EPU. Describe the methodology to determine these doses.	L-MT-08-041	ML081550640
General For all percentages used to describe the changes in dose and radiation levels at EPU described in Section 2.10, provide actual radiation and dose values.	L-MT-08-041	ML081550640
9) Piping & Non Destructive Examination Branch – (CPNB) (ML081500797)		
Identify the materials of construction for the reactor coolant pressure boundary (RCPB) piping and safe-ends. Discuss and explain the effect of the requested power uprate on the RCPB piping and safe-end materials and its impact on the potential degradation mechanisms.	L-MT-08-043	ML081640435
Identify the RCPB piping and safe-end components that are susceptible to intergranular stress-corrosion cracking (IGSCC). Discuss any augmented inspection programs that have been implemented and the adequacy of the augmented inspection programs in light of the EPU.	L-MT-08-043	ML081640435
Identify all flawed components including overlay repaired welds that have been accepted for continued service by analytical evaluation based on the American Society for Mechanical Engineers, Section XI rules. Discuss the adequacy of such analyses considering the effect of the EPU on the flaws.	L-MT-08-043	ML081640435
Identify the mitigation processes being applied at Monticello to reduce the RCPB component's susceptibility to IGSCC, and discuss the effect(s) of the requested EPU	L-MT-08-043	ML081640435

ENCLOSURE 16

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
on the effectiveness of these mitigation processes. For example, if hydrogen water chemistry (HWC) was applied at the plant, it would be necessary to perform the electrochemical potential measurements at the most limiting locations to ensure that the applied hydrogen injection rate is adequate to maintain the effectiveness of HWC (since oxygen content in the coolant is expected to increase due to the increased radiolysis of water from extended power uprate).		
Hope Creek, ML070460243, dated 2123107,lst page of attachment Identify the materials of construction for the reactor coolant pressure boundary (RCPB) piping/safe-ends. Discuss and explain the effect of the requested power uprate on the RCPB piping/safe-end materials.	L-MT-08-043	ML081640435
Identify the RCPB piping/safe-end components that are susceptible to intergranular stress corrosion cracking (IGSCC). Discuss any augmented inspection programs that have been implemented and the adequacy of the augmented inspection programs in light of the EPU.	L-MT-08-043	ML081640435
Identify all flawed components including overlay repaired welds that have been accepted for continued service by analytical evaluation based on American Society of Mechanical Engineers (ASME), Section XI rules. Discuss the adequacy of such analysis considering the effect of the EPU on the flaws.	L-MT-08-043	ML081640435
Identify the mitigation processes being applied at Hope Creek to reduce the RCPB component's susceptibility to IGSCC, and discuss the effect of the requested EPU on the effectiveness of these mitigation processes. For example, if hydrogen water chemistry (HWC) was applied at the plant, it would be necessary to perform the electrochemical potential measurements at the most limiting locations to ensure that the applied hydrogen injection rate is adequate to maintain the effectiveness of HWC since oxygen content in the coolant is expected to increase due to increased radiolysis of water resulting from extended power uprate.	L-MT-08-043	ML081640435
Vermont Yankee, ML033640138, dated 12/30/03, page 5 of attachment Section 3.5.1 of Attachment 4 of your submittal dated September 10, 2003, provides the results of the structural evaluation of the reactor coolant pressure boundary (RCPB) piping. Provide the basis for the disposition of the first system listed in this section.	L-MT-08-043	ML081640435
Identify the materials of construction for the Reactor Recirculation System piping and discuss the effect of the requested EPU on the material. If other than type "A" (per NUREG 0313) material exist, discuss augmented	L-MT-08-043	ML081640435

ENCLOSURE 16

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
inspection programs and discuss the adequacy of augmented inspection programs in light of the EPU.		
Section XI of the American Society of Mechanical Engineers (ASME) Code allows flaws to be left in service after a proper evaluation of the flaws is performed in accordance with the ASME, Section XI rules. Indicate whether such flaws exist in the Reactor Recirculation System piping and evaluate the effect of the EPU on the flaws.	L-MT-08-043	ML081640435
Discuss flaw mitigation steps that have been taken for the RCPB piping and discuss changes, if any, that will be made to the mitigation process as a result of the EPU.	L-MT-08-043	ML081640435
Browns Ferry, ML043440045, dated 12/30/04, questions 1-4 Explain why the reactor coolant pressure boundary (RCPB) piping materials are not affected by the power uprate.	L-MT-08-043	ML081640435
Identify the materials of construction for the Reactor Recirculation System piping and discuss the effect of the requested extended power uprate (EPU) on the material. If other than type "A" (per NUREG 0313) materials exist, discuss any augmented inspection programs and discuss the adequacy of augmented inspection programs in light of the EPU.	L-MT-08-043	ML081640435
Section XI of the American Society of Mechanical Engineers (ASME) Code allows flaws to be left in service after a proper evaluation of the flaws is performed in accordance with the ASME, Section XI rules. Indicate whether such flaws exist in the Reactor Recirculation System piping and evaluate the effect of the EPU on the flaws.	L-MT-08-043	ML081640435
Discuss flaw mitigation steps that have been taken for the RCPB piping and discuss changes, if any, that will be made to the mitigation process as a result of the EPU.	L-MT-08-043	ML081640435
10) Electrical Engineering Branch - EEEB 2nd round of questions (ML081620031)		
Provide the staff with the USAR section number that describes the AC load Study.	L-MT-08-043	ML081640435
The licensee will provide statements that the margins discussed in the acceptance review response for the batteries will be met during the development of the modifications.	L-MT-08-043	ML081640435
For the EQ analyses, clearly state that it has been completed and that NMC has identified the equipment that is impacted by EPU conditions.	L-MT-08-043	ML081640435
The licensee states the SBO analysis has been revised for EPU conditions, but does not explain what the changes	L-MT-08-043	ML081640435

ENCLOSURE 16

Table of Docketed NRC Acceptance Review Questions and NMC Response Letters Associated with the March 31, 2008 Monticello EPU LAR Submittal

Monticello EPU LAR Acceptance Review Question	NMC Letter No.	Accession No.
are. The licensee agreed to develop a table that outlines the changes in the SBO analysis from CLTP to the EPU. The table should include the standard acceptance criteria as well as changes in assumptions.		

Enclosure 17 to L-MT-08-052

NSPM Response to Review Items Documented in the June 26, 2008 NRC Non-Acceptance Letter

ENCLOSURE 17

NSPM Responses to NRC Review Items Documented in Non-Acceptance Letter

Enclosure 17 provides Northern States Power Company's, a Minnesota corporation (NSPM), response to review items identified in the NRC non-acceptance letter dated June 26, 2008 (ML 081770338). The formatting of NRC items and NSPM responses utilized in this enclosure consist of NRC review items presented first followed by an NSPM response.

In addition, this enclosure includes a revised response to an EEEB acceptance review Item documented in NSPM letter L-MT-08-042 (ML 081570467) regarding EQ analysis. The revised NSPM response is provided at the end of this enclosure. NSPM's response is supplemented by Task Report 1004, Rev. 1 (T1004), "Environmental Qualification" which follows the NSPM revised response.

NRC Review Area: **Steam Dryer Structural Integrity**

NRC Review Item:

(1) Noise Removal

The NRC has previously accepted the noise removal approach using low main steamline flow data for removing non-physical tones from dryer loads in a previous EPU application. The NRC staff may accept some noise removal for the nonphysical load near 100 Hz in MNGP, provided the licensee presents a quantitative substantiation of the removal, using a similar approach accepted by the NRC staff in a previous application. [ ]. As shown in the current submittal, the minimum alternate stress ratio is 1.79, which is significantly less than 2.0 (see Browns Ferry meeting summary dated June 5, 2008; ADAMS Accession No. ML081260712). When the above items are considered in the steam dryer analysis, the minimum alternating stress ratio (SR-a) is expected to be further reduced. In summary, the NRC staff expects the EPU amendment application to reflect a complete and final stress analysis of the dryer considering previously identified bias errors and uncertainties in the acoustic circuit model (ACM) as well as justifiable noise removal addressed above.

NSPM Response:

Please refer to revised Enclosure 11 of the EPU LAR resubmittal.

ENCLOSURE 17

NSPM Responses to NRC Review Items Documented in Non-Acceptance Letter

NRC Review Item:

Possible Dryer Modification

The MNGP steam dryer is a first generation square hood type dryer, and some degree of conservatism should be applied due to limited validation of ACM. For those plants with no dryer instrumentation (i.e., dryer analysis based purely on main steamline strain gage data), the acceptability of the steam dryer performance relies on the extent of the available margins in the predicted stress. The NRC staff and Advisory Committee on Reactor Safeguards consider a minimum alternating stress ratio of 2.0 as the acceptance threshold due to the limited validation of the ACM code. As stated in (1) above, the minimum alternating stress ratio for the MNGP steam dryer is less than 2.0. Based on the revised stress analysis results as noted in (1) above, the licensee should re-compute the projected minimum alternating stress ratio at the proposed EPU conditions applying the appropriate bump-up factors (EPU/current licensed thermal power) established from scale model testing to capture acoustic resonance effects for projected EPU conditions. If the minimum alternating stress ratio is less than 2.0, the licensee should identify relevant structural modifications of the steam dryer to achieve a minimum alternating stress ratio of ≥ 2.0 at EPU conditions.

NSPM Response:

Please refer to revised Enclosure 11 of the EPU LAR resubmittal.

NRC Review Item:

Evaluation of Locations with Existing Cracks

As stated in NMC's May 30, 2008, supplement, there are existing indications noted at welds V390, V1090, V10270, at the dryer support bracket guide channel, and at the access hole cover plate in drain channel-F. Using the revised stress analysis results, the licensee should determine the alternating stresses that are present at all identified indications (including intra-granular stress corrosion cracks at the access hole cover plate) in the MNGP steam dryer at current license thermal power, and evaluate the effect of EPU operation on the integrity of the dryer in the presence of the existing cracks. The licensee should implement either dryer design improvements and/or modifications, or perform analytical evaluations to confirm that the structural integrity of the dryer has not been compromised for EPU operation. The current dryer analysis in the application, as supplemented, does not address this topic, and hence is considered to be incomplete for detailed review.

ENCLOSURE 17

NSPM Responses to NRC Review Items Documented in Non-Acceptance Letter

NSPM Response:

Please refer to the attached calculation (Attachment 1) for a detailed evaluation of crack growth potential and impact on the normal modes of the structure in the frequency range of concern for FIV loading. NSPM will inspect the steam dryer during the next refueling outage to confirm the conclusions of this evaluation.

NRC Review Area: **Equipment Qualification**

NRC Review Item:

The licensee provided the supplemental information in the May 28, June 5, and June 12, 2008, letters. Based on its review of the supplemental information, the staff concludes that the original EPU application, as supplemented, is incomplete. During a June 17, 2008, teleconference, the NRC staff requested the licensee to clarify its June 12, 2008, response. Specifically, the NRC staff requested the licensee to clarify the following information that was contained in that submittal:

> The note explains that the process for final resolution of the identified EPU impact may include:
>
> - additional equipment-specific analysis to be documented in the equipment-specific qualification file, or
> - replacement or modification of a specific piece of equipment.
>
> The process for final resolution of the identified EPU impacts (additional equipment-specific analysis, replacement or modification) is controlled in accordance with the MNGP Equipment Qualification (EQ) Program requirements.
>
> A summary was included in the submittal. The summary concludes that analyses to determine the EPU impact are complete. It also states that the equipment-specific resolutions will be completed as controlled by the MNGP EQ Program requirements. Final resolution of identified impacts will be documented in the related equipment-specific qualification file prior to implementation of EPU in accordance with 10 CFR 50.49.

The licensee stated that with exception to a few components, the documentation portion of the equipment qualification process is all that remained. The NRC staff continued the discussion by requesting the licensee to identify the components that are still being evaluated for environmental qualification. In response to this request, the licensee stated that it is still in the process of performing an environmental qualification analysis on certain transmitters, flow switches, and motor control center buckets. The NRC staff expects EPU

amendment applications to reflect a complete and final environmental qualification analysis. Based on the information provided by the licensee, the NRC staff does not have adequate assurance that the licensee has fully completed its environmental qualification analysis in accordance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

The NRC expects a complete and final environmental qualification analysis prior to commencing the detailed review process to avoid multiple rounds of requests for additional information and subsequent revisions and re-reviews. Based on the above findings, the NRC staff concludes that the MNGP EPU amendment application, as supplemented, has several notable deficiencies as set forth above.

NSPM Response:

The environmental qualification analysis on "certain transmitters, flow switches, and motor control center buckets" as noted in previous correspondence as "in process" has been completed (with a clarification on the MCC bucket scope discussed below). A discussion of the results follows.

The evaluation and associated modification with respect to the "transmitters" is complete and these transmitters are planned for replacement. The modification scope includes certain transmitters for residual heat removal (RHR) heat exchanger d/p, RHR injection flow, RHR containment spray flow, core spray pump flow, and torus wide range level. Only the torus wide range level transmitters are required to be replaced as a result of the evaluation, the other transmitters will be replaced to enhance margin. The replacements will occur through the normal EQ, design, and work control processes and will be completed prior to operating at EPU power levels.

The evaluation with respect to the "flow switches" is complete and the flow switches are planned for replacement. The scope includes certain flow switches for the standby gas treatment system (SBGTS). The replacements will occur through the normal EQ, design, and work control processes and will be completed prior to operating at EPU power levels.

The current analysis work being performed relative to "motor control center buckets" is not EPU-related and is a current issue being addressed via the site corrective action program. This scope of work is related to a steam line HELB event, and per the CLTR, a Constant Pressure Power Uprate has no effect on the steam pressure or enthalpy at the postulated break locations. Therefore, EPU has no effect on the mass and energy releases from an HELB in a steam line and no plant-specific evaluation is required for steam line breaks to support EPU.

Additional detailed, equipment-specific, analyses (both EPU and non-EPU related) have been completed since the withdrawal of the original EPU LAR. These additional analyses

have not identified any additional required equipment changes or modifications beyond the scope already communicated via this submittal and previous correspondence.

In conclusion, all required EPU related EQ analyses are complete. Impacted equipment will be replaced or modified, as appropriate, and the final resolution of identified impacts will be documented in the related equipment-specific qualification file prior to implementation of EPU in accordance with 10 CFR 50.49.

NRC Review Area: **Instrument Setpoint Methodology**

NRC Item:

Setpoint Calculation Methodology

The licensee needs to provide documentation (including sample calculations) of the methodology used for establishing the limiting setpoint (or NSP) and the limiting acceptable values for the as-found and as-left setpoints as measured in periodic surveillance testing as described below. Indicate the related analytical limits and other limiting design values (and the sources of these values) for each setpoint.

NSPM Response:

The response to this question is contained in NMC (now NSPM) letter, "Response to Requests for Additional Information for License Amendment Request for Power Range Neutron Monitoring System Upgrade" (ML 082620582), dated September 16, 2008 (Reference 1).

NRC Item:

Safety Limit (SL)-Related Determination

Provide a statement as to whether or not the setpoint is a limiting safety system setting for a variable on which an SL has been placed as discussed in 10 CFR 50.36(c)(1)(ii)(A). Such setpoints are described as "SL-Related" in the discussions that follow. In accordance with 10 CFR 50.36(c)(1)(ii)(A), the following guidance is provided for identifying a list of functions to be included in the subset of limiting safety system settings (LSSSs) specified for variables on which SLs have been placed as defined in Standard Technical Specifications (STS) Sections 2.1.1, "Reactor Core SLs," and 2.1.2, "Reactor Coolant System Pressure SLs." This subset includes automatic protective devices in Technical Specifications (TSs) for specified variables on which SLs have been placed that: (1) initiate a reactor trip; or (2) actuate safety systems. As such, these variables provide

protection against violating reactor core safety limits, or reactor coolant system pressure boundary safety limits.

Examples of instrument functions that might have LSSSs included in this subset in accordance with the plant-specific licensing basis are pressurizer pressure reactor trip (pressurized-water reactors), rod block monitor withdrawal blocks (boiling-water reactors), feedwater and main turbine high water level trip (boiling-water reactors), and end-of-cycle recirculation pump trip (boiling-water reactors). For each setpoint, or related group of setpoints that the licensee determined not to be SL-related, explain the basis for this determination.

NSPM Response:

Xcel Energy has evaluated the proposed changes to the Technical Specifications for EPU in Enclosure 2 herein and concluded that these changes do not involve a LSSS required to protect a Safety Limit. The following changes were part of the scope of the evaluation.

Average Power Range Monitors Simulated Thermal Power – High

The proposed EPU changes maintain the MELLLA CLTP slope of the APRM Simulated Thermal Power scram lines in terms of absolute core power versus recirculation drive flow. The method of adjusting the flow biased setpoints is in accordance with that described in the CLTR which has been previously accepted by the staff.

None of the MNGP safety analyses credit operation of the Flow Biased - Simulated Thermal Power High Scram for event mitigation for EPU. There are no analytical limits or event dependencies associated with these setpoints. The setpoints are not involved in any trip initiations or safety system actuations that are designed to protect safety limits.

The safety analysis credits the scram that is initiated by the Neutron Flux - High instrument, which bounds the flow biased setpoints. This particular setpoint is SL-related and is discussed in Enclosure 1, Section 2.C of Reference 1.

The following is an excerpt from Section 7.3.5.2.3 of the MNGP USAR which discusses safety analyses. The EPU changes and safety analyses are consistent with this section.

"*The effectiveness of the APRM high flux scram signals in preventing fuel damage following single component failures or single operational errors is demonstrated in the transient analyses... In all such failures, no fuel damage occurs...These analyses assumed a scram at the power corresponding to the scram clamp regardless of the starting point ...*"

Given the above, the proposed changes to the subject APRM setpoints are not credited in any safety analysis and are not SL-related. Therefore, MNGP has determined that it is not

a limiting safety system setting for a variable on which an SL has been placed and is not SL-related.

Turbine Stop Valve Closure and Turbine Control Valve Fast Closure SCRAM Bypass

The above scrams are required to be enabled above a certain reactor power level as discussed in the associated Technical Specification bases. The power level is applied as a specific operating condition for these scrams in Technical Specification table 3.3.1.1-1. The power level requirement is verified in accordance with surveillance requirement SR 3.3.1.1.13. The proposed change represents a change to a required instrument operating condition, and should not be regarded as a change to an instrument setpoint.

The specific operating condition enforces an assumption for P_{bypass} in the EPU transient analysis. P_{Bypass} is being scaled down (45% to 40% of rated thermal power) to maintain the power level at EPU conditions. In terms of the EPU transient analysis, P_{bypass} is the reactor power level below which turbine stop valve position and the turbine control valve fast closure scrams are assumed to be bypassed. This power level assumption affects the determination and confirmation of off-rated (power dependent) limits within the transient analysis. The value of P_{bypass} is 40% of 2004 MWt in the EPU transient analysis for the equilibrium GE 14 core, which is consistent with the proposed changes.

The instrument loop which initiates the logic to cause these scrams to be enabled consists of a turbine pressure switch and a relay connected to enable logic. The dynamic response of this instrument during a design basis event is not modeled within the MNGP safety analysis in order to verify P_{bypass} or to demonstrate margin of any kind. The instrument setpoint is fundamentally derived from the relationship of turbine first stage pressure (in psig) with a corresponding power level. The proposed changes to the required operating condition do not affect the power/ first stage pressure relationship. The actual instrument settings are being revised to conform to the design of the replacement high pressure turbine. The absolute power value of the analytical limit for this instrument is not changing; it is being scaled lower from 30% at CLTP to 26.6% at EPU to account for the relative change in percent power. The EPU startup testing will include a validation that the subject scrams are enabled prior to reaching the proposed P_{bypass} power level value of 40%.

Given the above, the proposed change is confined to the required operating condition for the turbine scrams above. This operating condition corresponds to P_{bypass} in the EPU transient analysis. The change represents a change to an assumption in a safety analysis and is not a setpoint change per se. Therefore, MNGP has determined that it is not a limiting safety system setting for a variable on which an SL has been placed and is not SL-related.

ENCLOSURE 17

NSPM Responses to NRC Review Items Documented in Non-Acceptance Letter

NRC Item:

Setpoints Determined to be SL-Related

The NRC letter to the Nuclear Energy Institute (NEI) Setpoint Methods Task Force dated September 7, 2005 (Accession No. ML052500004) describes Setpoint-Related TS (SRTS) that are acceptable to the NRC for instrument settings associated with SL-related setpoints. Specifically: Part "A" of the Enclosure to the letter provides limiting conditions for operation notes to be added to the TS, and Part "B" includes a check list of the information to be provided in the TS Bases related to the proposed TS changes.

a. Describe whether and how you plan to implement the SRTS suggested in the September 7, 2005, letter. If you do not plan to adopt the suggested SRTS, then explain how you will ensure compliance with 10 CFR 50.36 by addressing items 3b and 3c, below.

b. As-found setpoint evaluation: Describe how surveillance test results and associated TS limits are used to establish operability of the safety system. Show that this evaluation is consistent with the assumptions and results of the setpoint calculation methodology. Discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be "inoperable" or "operable but degraded". If the criteria for determining operability of the instrument being tested are located in a document other than the TS (e.g. plant test procedure), explain how the requirements of 10 CFR 50.36 are met.

c. As-left setpoint control: Describe the controls employed to ensure that the instrument setpoint is, upon completion of surveillance testing, consistent with the assumptions of the associated analyses. If the controls are located in a document other than the TS (e.g. plant test procedure) explain how the requirements of 10 CFR 50.36 are met.

NSPM Response:

The proposed changes to the Technical Specifications submitted for EPU do not involve changes to instrument settings associated with SL-related setpoints.

Please note that MNGP's response to a similar question in Reference 1 provides a response to the methodology portion of this question.

ENCLOSURE 17

NSPM Responses to NRC Review Items Documented in Non-Acceptance Letter

NRC Review Item:

Setpoints Not Determined to be SL-related

Describe the measures to be taken to ensure that the associated instrument channel is capable of performing its specified safety functions in accordance with applicable design requirements and associated analyses. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completion of periodic surveillance is consistent with your setpoint methodology. Also, discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be "inoperable" or "operable but degraded". If the controls are located in a document other than the TS (e.g., plant test procedure), describe how it is ensured that the controls will be implemented.

NSPM Response:

The MNGP Setpoint Control Program establishes calibration settings, tolerances, and Allowable Values in setpoint calculations that utilize the GE Setpoint Methodology (NEDC-31336P-A) that has been reviewed and approved by NRC. The GE setpoint methodology is used to establish the setpoints for the flow-biased APRM setpoints. The setpoint controls and corrective action processes are described in Enclosure 1, Section 3 of Reference 1.

ENCLOSURE 17

NSPM Responses to NRC Review Items Documented in Non-Acceptance Letter

References

1. Letter form NMC (Timothy J. O'Connor) to Document Control Desk, "Response to Requests for Additional Information for License Amendment Request for Power Range Neutron Monitoring System Upgrade (ML 082620582)

ATTACHMENT 1

FLAW EVALUATION AND VIBRATION ASSESSMENT OF EXISTING MONTICELLO STEAM DRYER FLAWS FOR EXTENDED POWER UPRATE

Report No. 0800760.401
Revision 0
Project No. 0800760
October 2008

Flaw Evaluation and Vibration Assessment of Existing Monticello Steam Dryer Flaws for Extended Power Uprate

Prepared for:

Nuclear Management Company
Monticello, MN
Contract No 1005 Release 4 Amendment 3

Prepared by:

Structural Integrity Associates, Inc.
Centennial, CO

Prepared by:	D. V. Sommerville	Date:	10/24/2008
Reviewed by:	M.L. Herrera, P.E. & S. Tang	Date:	10/24/2008
Approved by:	K.K. Fujikawa, P.E.	Date:	10/24/2008

Structural Integrity Associates, Inc.

REVISION CONTROL SHEET

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Section	Pages	Revision	Date	Comments
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3	3-1			
4	4-1 – 4-12			
5	5-1 – 5-7			
6	6-1 – 6-2			
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1.0 INTRODUCTION & SCOPE

The Monticello Steam Dryer was inspected using the guidelines provided in BWRVIP-139 [1] during the 2005 and 2007 refueling outages. The dryer internals were inspected for the first time in the 2007 refueling outage. Reportable indications were identified in five locations during these examinations [2, 3]. The locations with identified indications were not inspected prior to the 2005 refueling outage.

Nuclear Management Company (NMC) has previously performed flaw evaluations [4, 5] of the indications identified in the steam dryer for operation at current licensed thermal power (CLTP). NMC desires the previous flaw evaluations to be updated to incorporate the effect of operation at extended power uprate (EPU) conditions. Further, the effect of these flaws on the dynamic characteristics of the steam dryer is evaluated in order to assess if the uncracked steam dryer finite element model (FEM) used to perform the EPU stress analysis should be modified to address cracking.

This report documents flaw evaluations and a vibration assessment of the steam dryer indications identified in Section 3.0 considering EPU operating conditions. The methods utilized in the previous flaw evaluations are utilized here, where appropriate, to maximize consistency between the current and previous work. Further, plant specific EPU flow induced vibration (FIV) stresses are used for the current evaluation; plant specific stresses were not available for the previous evaluations.

2.0 STEAM DRYER INDICATIONS

The indications identified in the Monticello dryer are reported in References [2,3]. For clarity, the Inspection Notification Reports (INR) for each indication are provided in Attachments 1 through 5. Figures 2-1 through 2-7 illustrate the general location of each indication. Table 2-1 summarizes the dimensions of each flaw as reported in the inspection reports.

Table 2-1. Summary of Monticello Steam Dryer Indications.

Location	Size		Notes
	2005 [2]	2007 [3]	
V3 90°	1.375"	1.375"	One indication is located at the top of Weld V3 90°. This indication extends for approximately 1.375 inches on the outside of the end panel weld across the top and down the inside of the weld to Dryer Bank "B" for approximately 1 inch. The indication is contained within the weld material.
V10 90°	1.375"	1.375" 0.25"	Two indications have been identified at this location in 2007. One is located at the top of Weld V10 90° and extends for approximately 1.375 inches in length. The second indication is located on the opposite side the plate from the first indication and is approximately 0.25 inch long. Both indications are contained within the weld material. The 2005 inspection report does not suggest the region in which the second flaw is located was inspected during that time.
V10 270°	< 1"	< 1"	One indication is located at the top of the weld at the junction of the end panel and the dryer bank. This indication is less than 1 inch in length. The indication is contained within the weld material.
215° Dryer Support Bracket Guide Channel	0.75"	0.75"	One indication is located approximately 4 feet from the bottom of 215° Dryer Support Bracket Guide Channel. The indication is oriented horizontally across from a possible arc strike around the corner of the channel and into the left toe of a vertical weld on the face of the dryer. The indication is on the right side of the Guide Channel and the length is approximately ¾ inch.
Drain Channel "F" Access Hole Cover Plate	Not Inspected	No sizing provided	The 2007 dryer internal examination results indicate cracking sporadically around the circumference of the access hole cover plate. No flaw dimensions are provided.

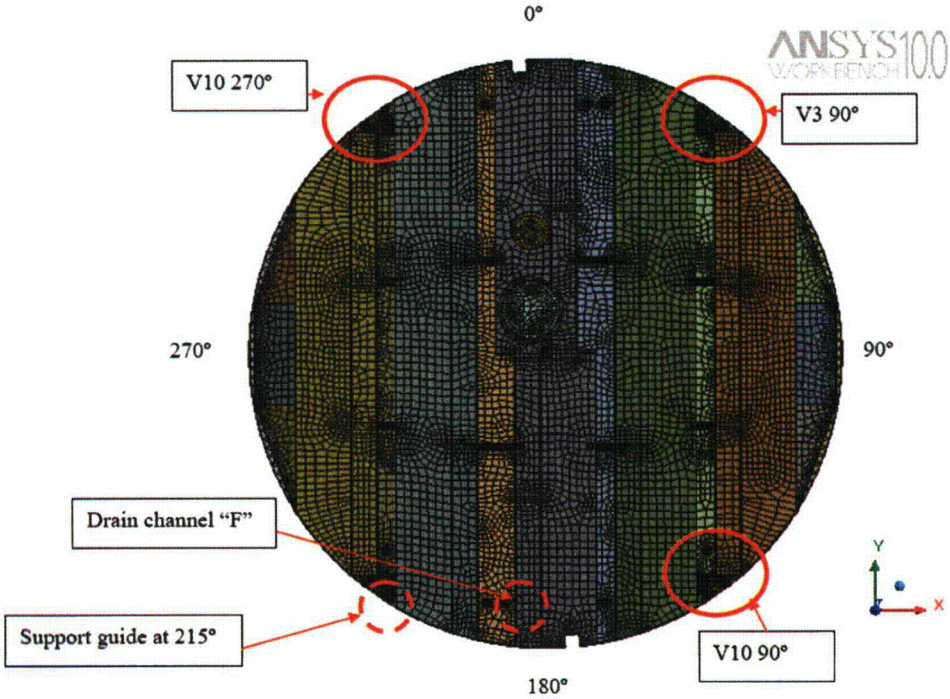
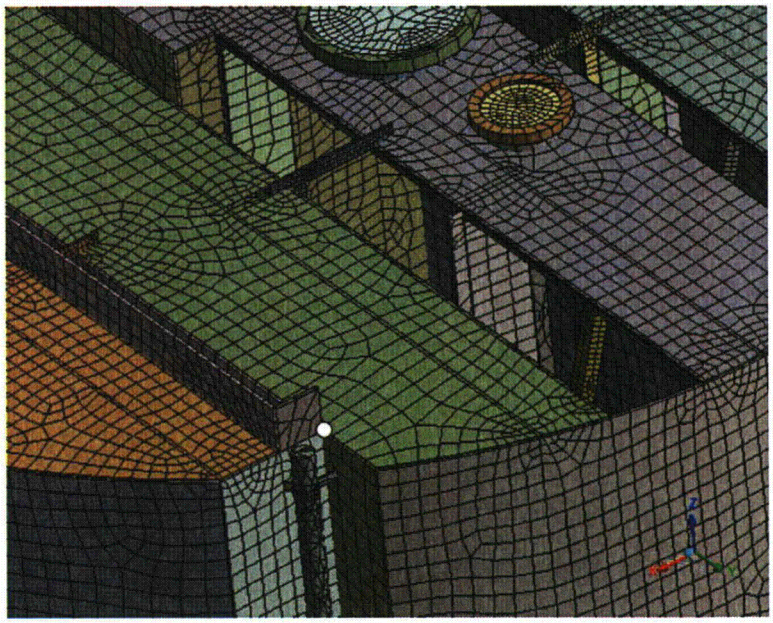
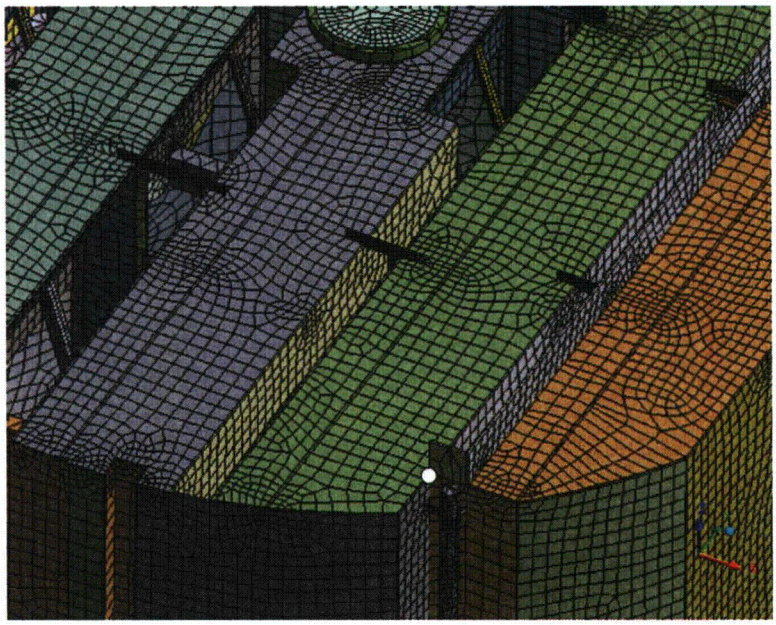


Figure 2-1. Steam Dryer Indications Identified on Top View of Dryer Finite Element Model.



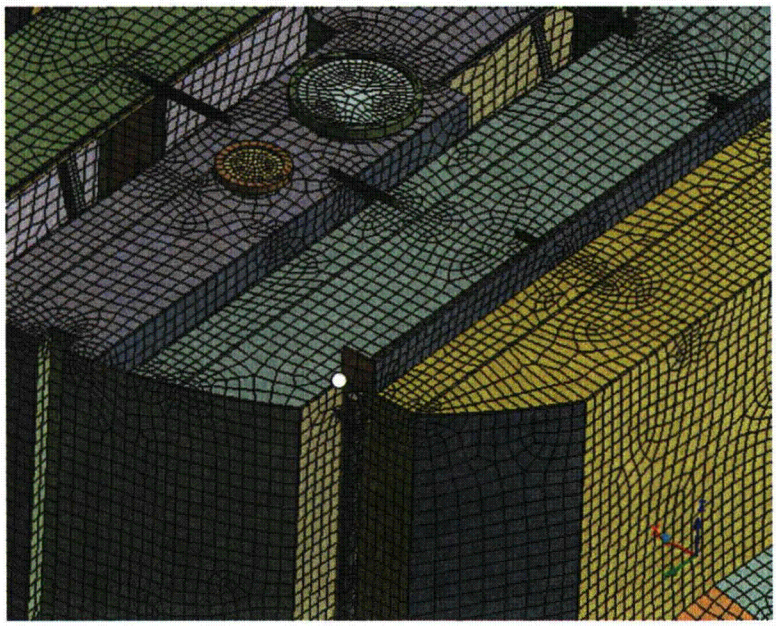
Note: 1. Dryer lifting lug has been removed to better visualize indication location.
2. Indication location is identified by white circle.

Figure 2-2. Location of V3 90° Indication.



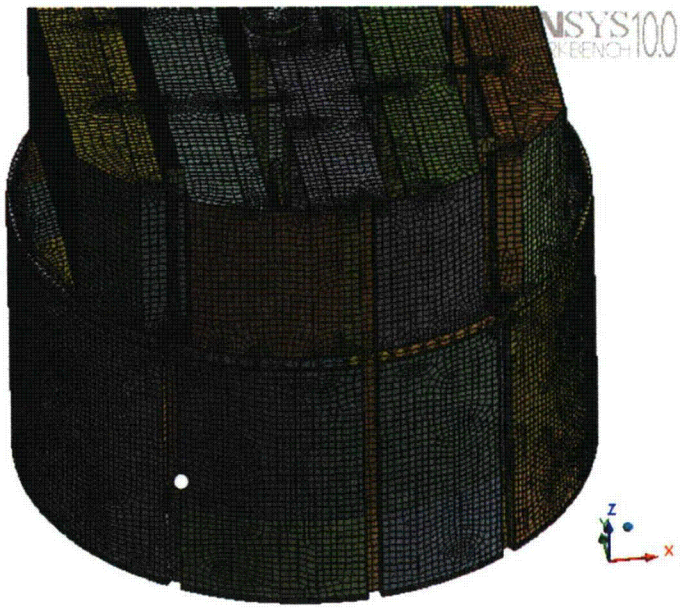
Note: 1. Dryer lifting lug has been removed to better visualize indication location.
2. Indication location is identified by white circle.

Figure 2-3. Location of V10 90° Indication.



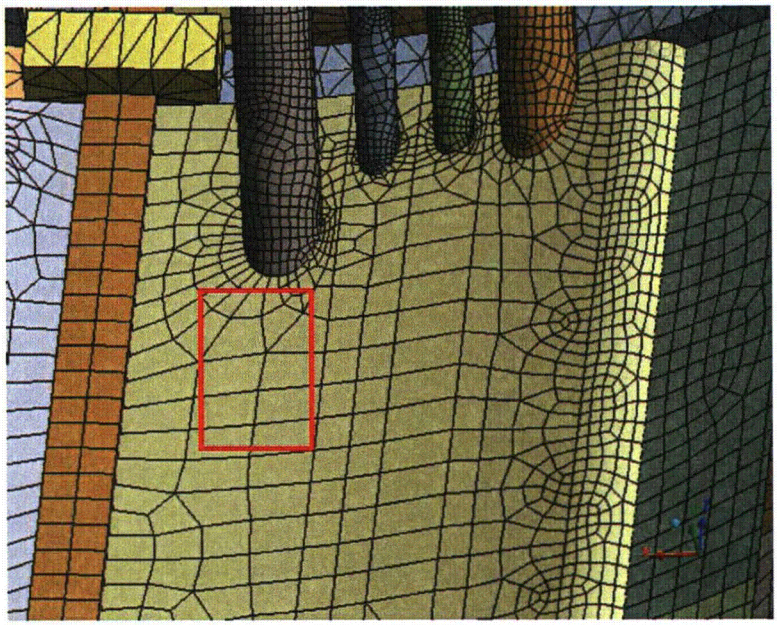
Note: 1. Dryer lifting lug has been removed to better visualize indication location.
2. Indication location is identified by white circle.

Figure 2-4. Location of V10 270° Indications.



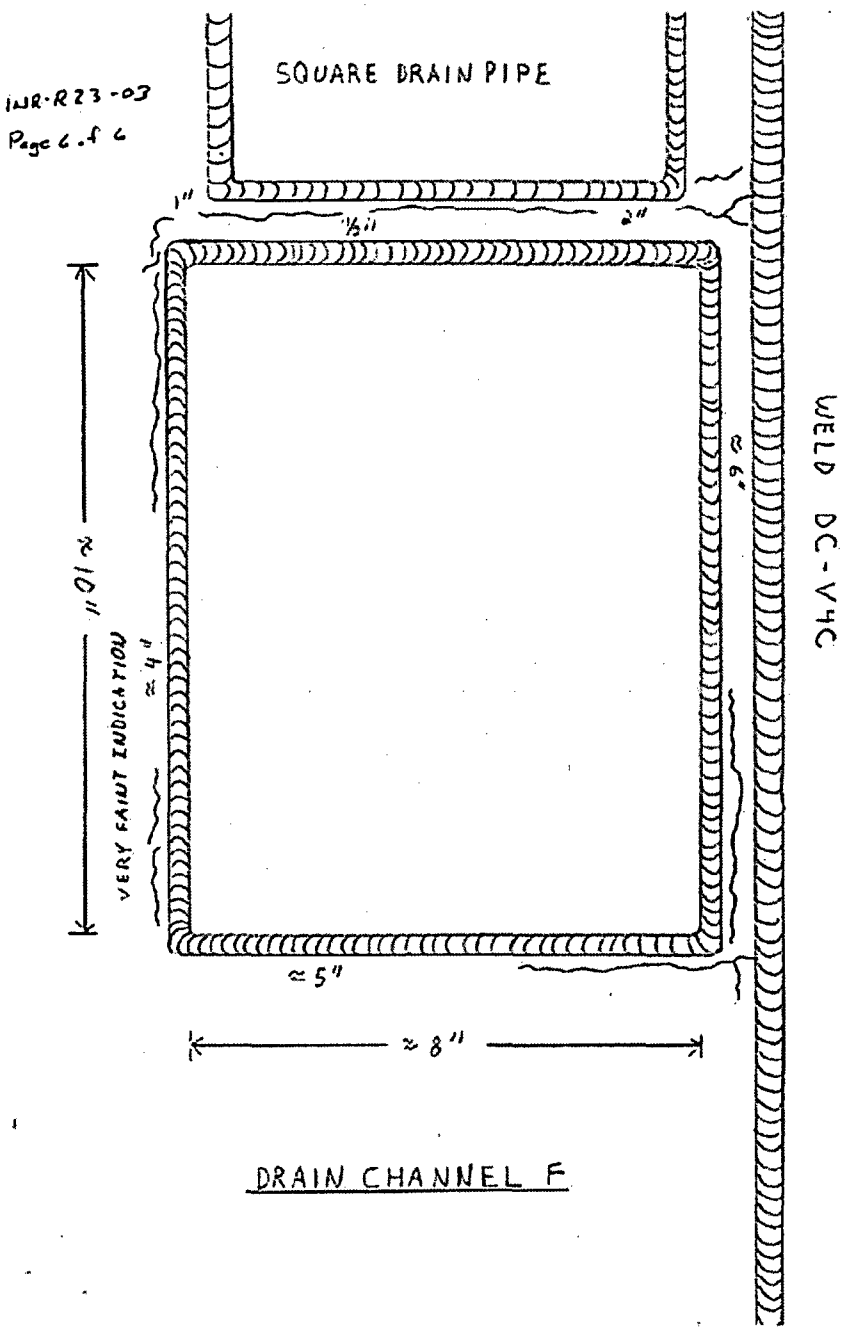
Note: 1. Approximate indication location is identified by white circle.

Figure 2-5. Location of Dryer Support Guide Channel Indication at 215°.



Note: 1. View orientation is from inside dryer looking outward.
2. Approximate indication locations are identified by red box.

Figure 2-6. Approximate Location of Drain Channel "F" Indications.



Note: 1. View orientation is from inside dryer looking outward.
2. Indications observed to follow weld HAZ.

Figure 2-7. Schematic of Drain Channel "F" Indications [5].

3.0 INPUT DATA

The following data are used as inputs to this evaluation:

- Indication location, orientation, size [2, 3, 4, 5]
- CLTP flaw evaluations [4, 5]
- FIV steam dryer stresses for EPU operation [6]
- Steam Dryer Reactor Internal Pressure Difference (RIPD) for EPU Operation [7]

The indication locations and sizes are obtained from the previous IVVI reports as well as the previous flaw evaluations. The FIV steam dryer stresses for EPU operation are obtained from Continuum Dynamics, Inc (CDI). All inputs except for the EPU FIV steam dryer stresses and previous Structural Integrity flaw evaluation were obtained from NMC via Design Information Transmittal EPU-0284 [8].

4.0 FLAW EVALUATION METHODS AND ASSUMPTIONS

This section describes the general methods and assumptions used to perform the steam dryer flaw evaluations. Each indication will be addressed separately below. In general, the methods of BWRVIP-139 [1] are utilized where specific guidance is applicable. Further, the methods and assumptions of the previous flaw evaluations [4, 5] are incorporated in the current evaluation where they remain applicable.

4.1 V3 90°, V10 90°, V10 270° Indications

This section describes the methods and assumptions used for the flaw evaluation of the four vertical indications in the V3 and V10 welds at 90° and 270°.

4.1.1 Assumptions

The following assumptions are used for the subject flaw evaluation:

1. All indications are contained within the weld material and no branching is evident; therefore, the flaws are assumed to be fatigue cracks rather than intergranular stress corrosion cracking (IGSCC).
2. The flaw configuration is adequately modeled as an edge crack in a semi-infinite plate.
3. The length of the vertical weld compared to the crack length is sufficiently large that a finite thickness correction factor is not included in the linear elastic fracture mechanics (LEFM) solution.
4. The mode I stress intensity factor (K_I) is expected to be small; therefore, a plastic zone size correction is not included in the LEFM solution.
5. The cracks in the fillet welds on either side of the vertical plate are independent flaws. The fillet welds are not full penetration welds; therefore, each fillet weld is evaluated as an independent structure.
6. The alternating stress intensity factor used for calculation of fatigue crack growth (FCG), ΔK_I, is obtained from the range of alternating stress intensity contributed by flow induced vibration (FIV) loading only.
7. The subject geometry is geometrically similar to a thin plate; therefore, the stress state will be characterized by a plane stress condition. This will result in one of the three

principal stresses being close to zero. In this case it is conservative to assume the crack driving force is bounded by the ASME B&PV Code defined stress intensity acting as a membrane stress along the entire surface of the crack face. The stress intensity will always be equal or larger to the largest principal stress component for this configuration.

8. System thermal cycles, seismic and hydraulic loads contribute an insignificant number of cycles during the next operating period; therefore, they make a negligible contribution to FCG compared to FIV loading and are not calculated here.
9. Deadweight, static thermal loads, differential pressure, and weld residual stresses contribute to the mean K_I rather than ΔK_I; therefore, they are considered only in the selection of a conservative R-ratio and not specifically considered in calculation of a mean K_I. Assuming an R-ratio of 1 incorporates the maximum effects of mean stress on the expected FCG of the steam dryer indications.

4.1.2 Methods

The flaw evaluation of the subject indications is performed using the following methods:

1. The EPU FIV range of alternating stress intensities output from the existing uncracked finite element model (FEM) of the Monticello steam dryer for each of the three crack locations are reviewed. A composite stress distribution, which bounds the individual stress distributions for all locations, is assembled from the top, middle, and bottom element output locations for each of the three indication locations (V3 90°, V10 90°, and V10 270°).
2. The composite stress distribution is conservatively scaled by a weld factor of 1.8 to incorporate peak stress effects in the weld.
3. Subsequent stress intensity factor and FCG calculations are performed for a bounding flaw size and are applicable for all flaw locations.
4. The range of stress intensity factor experienced as a result of the EPU FIV loading is calculated using two separate methods:

Method 1: The ΔK_I can be determined from the superposition of a constant and linearly varying stress distribution extracted from the composite stress distribution at any desired crack length. The total ΔK_I is determined using the K_I solutions shown on pages 193 and 205 of Reference [9]:

Constant Stress: $$K_I = 1.1215 \cdot P_m \cdot \sqrt{\pi \cdot a} \quad (1a)$$

Linear Stress Variation: $$K_I = 0.439 \cdot P_b \cdot \sqrt{\pi \cdot a} \quad (1b)$$

Superposition of both: $$K_I = (1.1215 \cdot P_m + 0.439 \cdot P_b) \cdot \sqrt{\pi \cdot a} \quad (1c)$$

Where: P_m is the constant stress distribution defined as equal to the total stress at the crack tip, ksi

P_b is the linearly varying stress distribution defined as equal to the difference between the stress at the free surface and Pm, ksi

a is the crack length, in

Figures 4-1 and 4-2 illustrate the flaw configurations for which Equations (1a) and (1b) are applicable.

Method 2: The total ΔK_I can be determined by integrating the K_I solution derived for a point load applied at an arbitrary location along an edge crack in a semi-infinite plane. Page 197 of Reference [9] gives the K_I solution for this configuration as:

$$K_I = \frac{2}{\sqrt{\pi \cdot a}} \cdot P \cdot \frac{1.3 - 0.3 \cdot \left(\frac{b}{a}\right)^{5/4}}{\sqrt{1 - \left(\frac{b}{a}\right)^2}} \quad (2a)$$

Where: P is a force per unit thickness applied at a distance b from the free surface, kip/in

b is the point of load application measured with respect to the free surface, in

a is the crack length, in

Figure 4-3 illustrates the flaw configuration for which Equation (2a) is applicable.

Defining P to be a function of b, P(b), and noting that the product of the stress distribution along the crack face σ(b) and a differential length of crack face, db, is equal to the force per unit thickness, P(b), defined in Equation (2a):

$$P(b) = \sigma(b) \cdot db \qquad (2b)$$

Inserting Equation (2b) into Equation (2a) and integrating along the length of the crack face gives a solution for K_I for an arbitrary stress distribution along an edge crack in a semi-infinite plane:

$$K_I = \frac{2}{\sqrt{\pi \cdot a}} \cdot \int_0^a \sigma(b) \cdot \frac{1.3 - 0.3 \cdot \left(\frac{b}{a}\right)^{5/4}}{\sqrt{1 - \left(\frac{b}{a}\right)^2}} \cdot db \qquad (2c)$$

Where: σ(b) is the equation describing the stress distribution along the crack face, ksi

b is the location along the crack face measured with respect to the free surface, in

db is the differential element of the crack length, in

a is the crack length, in

Either method can be used to assess the range of alternating stress intensity at the crack tip. Equation (2c) will provide a more accurate estimate because it incorporates a more accurate expression for the tractions on the crack face.

5. The FCG growth expected during the next operational cycle is determined using the methods contained in Article C-3000 of the ASME B&PV Code, Section XI. Note that since the flaw is fully contained in the weld metal and is not considered to be IGSCC, SCC growth is not considered.

4.2 215° Dryer Support Bracket Guide Channel Indication

This section describes the methods and assumptions used for the flaw evaluation of the indication in the dryer support bracket guide channel at 215°.

4.2.1 Assumptions

The following assumptions are used for the subject flaw evaluation:

1. The flaw exists in the base metal and is oriented perpendicular to the weld and heat affected zone (HAZ); therefore, it has the characteristics of a fatigue crack.
2. The flaw configuration is adequately modeled as a center crack in an infinite plate.
3. The mode I stress intensity factor (K_I) is expected to be small; therefore, a plastic zone size correction is not included in the LEFM solution.
4. The alternating stress intensity factor used for calculation of FCG, ΔK_I, is obtained from the range of alternating stress intensity contributed by FIV loading only.
5. The subject geometry is a thin plate; therefore, the stress state will be characterized by a plane stress condition. This will result in one of the three principal stresses being close to zero. In this case it is conservative to assume the crack driving force is bounded by the ASME B&PV Code defined stress intensity acting as a membrane stress along the entire surface of the crack face. The stress intensity will always be equal or larger to the largest principal stress component for this configuration. Further, for plates, the through-wall stress distribution will exhibit tensile stresses on one side and compressive stresses on the opposite side. This stress distribution suggests that the flaw would likely not grow through-wall.
6. System thermal cycles, seismic and hydraulic loads contribute an insignificant number of cycles during the next operating period; therefore, they make a negligible contribution to FCG compared to FIV loading and are not calculated here.

7. Deadweight, static thermal loads, differential pressure, and weld residual stresses contribute to the mean K_I rather than ΔK_I; therefore, they are considered only in the selection of a conservative R-ratio and not specifically considered in calculation of a mean K_I. Assuming an R-ratio of 1 incorporates the maximum effects of mean stress on the expected FCG of the steam dryer indications.

4.2.2 Methods

The flaw evaluation of the subject indication is performed using the following methods:

1. The EPU FIV range of alternating stress intensities output from the existing uncracked FEM of the Monticello steam dryer for a region 12" wide by 6" high around the flaw location are reviewed. A bounding range of alternating stress intensity is selected.
2. The range of stress intensity factor experienced as a result of the EPU FIV loading is calculated using a center cracked panel solution for a uniform membrane stress distribution [9]:

   Membrane Stress: $$K_I = \sigma \cdot \sqrt{\pi \cdot a} \qquad (3)$$

   Where: σ is a uniform stress distribution, ksi
   a is the crack half length, in

   Figure 4-4 illustrates the flaw configuration for which Equation (3) is applicable.

3. The FCG growth expected during the next operational cycle is determined using the methods contained in Article C-3000 of the ASME B&PV Code, Section XI. Note that since the flaw exists in the base metal and is not considered to be IGSCC, no SCC growth must be calculated.

4.3 Drain Channel "F" Access Hole Cover Plate Indications

This section describes the methods and assumptions used for the flaw evaluation of the indications in the drain channel access hole cover plate.

4.3.1 Assumptions

The following assumptions are used for the subject flaw evaluation:

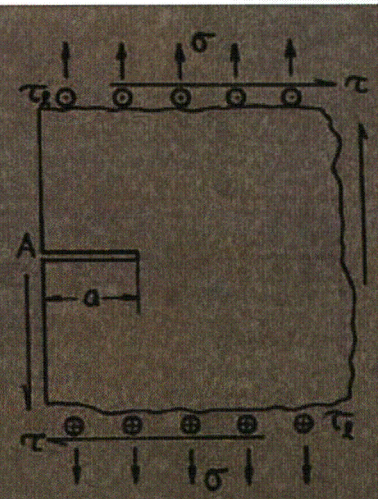
1. The flaws exist only in the weld HAZ and remain around the perimeter of the access hole cover plate; therefore, they are expected to caused by IGSCC.
2. The cover plate does not appear on fabrication drawings of the steam dryer; therefore the exact dimensions are not known. The assumed dimensions and extent of cracking used for the previous flaw evaluation [5] and shown in Figure 2-7 are used here as well.
3. The flaw configuration is adequately modeled as a center crack in an infinite plate.
4. The mode I stress intensity factor (K_I) is expected to be small; therefore, a plastic zone size correction is not included in the LEFM solution.
5. The alternating stress intensity factor used for calculation of FCG, ΔK_I, is obtained from the alternating stress intensity contributed by FIV loading only.
6. The subject geometry is a thin plate; therefore, the stress state will be characterized by a plane stress condition. This will result in one of the three principal stresses being close to zero. In this case it is conservative to assume the crack driving force is bounded by the ASME B&V Code defined stress intensity acting as a membrane stress along the entire surface of the crack face. The stress intensity will always be equal or larger to the largest principal stress component for this configuration. Further, for plates, the through-wall stress distribution will exhibit tensile stresses on one side and compressive stresses on the opposite side. This stress distribution suggests that the flaw would likely not grow through-wall.
7. System thermal cycles, seismic and hydraulic loads contribute an insignificant number of cycles during the next operating period; therefore, they make a negligible contribution to FCG compared to FIV loading and are not calculated here.
8. Deadweight, static thermal loads, differential pressure, and weld residual stresses contribute to the mean K_I rather than ΔK_I; therefore, they are considered only in the

selection of a conservative R-ratio and not specifically considered in calculation of a mean K_I. Assuming an R-ratio of 1 incorporates the maximum effects of mean stress on the expected FCG of the steam dryer indications.

4.3.2 Methods

The flaw evaluation of the subject indications is performed using the following methods:

1. The EPU FIV range of alternating stress intensities output from the existing uncracked FEM of the Monticello steam dryer for a region approximately equivalent to the assumed dimensions of the cover plate are reviewed. A bounding range of alternating stress intensity is selected.
2. The bounding range of alternating stress intensity is conservatively scaled by a weld factor of 1.8 to incorporate peak stress effects.
3. The range of stress intensity factor experienced as a result of the EPU FIV loading is calculated using the center cracked panel solution for a uniform stress distribution given by Equation (3).
4. Both the IGSCC growth and FCG expected during the next operational cycle are evaluated:
    a. The expected FCG growth is determined using the methods contained in Article C-3000 of the ASME B&PV Code, Section XI.
    b. The expected IGSCC crack growth for each indication is calculated assuming a 100% capacity factor, a two year fuel cycle, and the bounding IGSCC growth rate of 5E-5 in/hr per crack tip.
5. The ability of the cover plate remaining ligament to react the operational loads is assessed using plastic collapse as the failure mechanism and the EPU faulted RIPD [7] as the bounding load. This approach is consistent with the previous flaw evaluation [5].



Stress Intensity Factors

$$\begin{Bmatrix} K_I \\ K_{II} \end{Bmatrix} = 1.1215 \begin{Bmatrix} \sigma \\ \tau \end{Bmatrix} \sqrt{\pi a}$$

$$K_{III} = \tau_\ell \sqrt{\pi a}$$

Displacements at A

$$\begin{Bmatrix} 2v \\ 2u \end{Bmatrix} = 1.454 \frac{4}{E'} \begin{Bmatrix} \sigma \\ \tau \end{Bmatrix} a = \frac{5.816}{E'} \begin{Bmatrix} \sigma \\ \tau \end{Bmatrix} a$$

$$2w = \frac{2\tau_\ell a}{G}$$

Methods and References: K_I Integral Transform-Singular Integral Equation (Wigglesworth 1957; Kotter 1965a, Bueckner 1966; Sneddon 1971a; Benthem 1972); Successive Stress Relaxation-Integral Equation (Irwin 1958b, 1960a; Lachenbruch 1961; Nishitani 1971a; Tada 1972a)

K_{II} From similarity of free boundary corrections between Mode I and Mode II (Tada 1973)

K_{III} Westergaard Stress Function, etc.

Accuracy: K_I, K_{II} Within one unit of last digit

K_{III} Exact

Displacements were derived by Paris' equation (Paris 1957) (see Appendix B)

Figure 4-1. Stress Intensity Factor Solution for an Edge Cracked Semi-Infinite Plate with Uniform Stress [9].

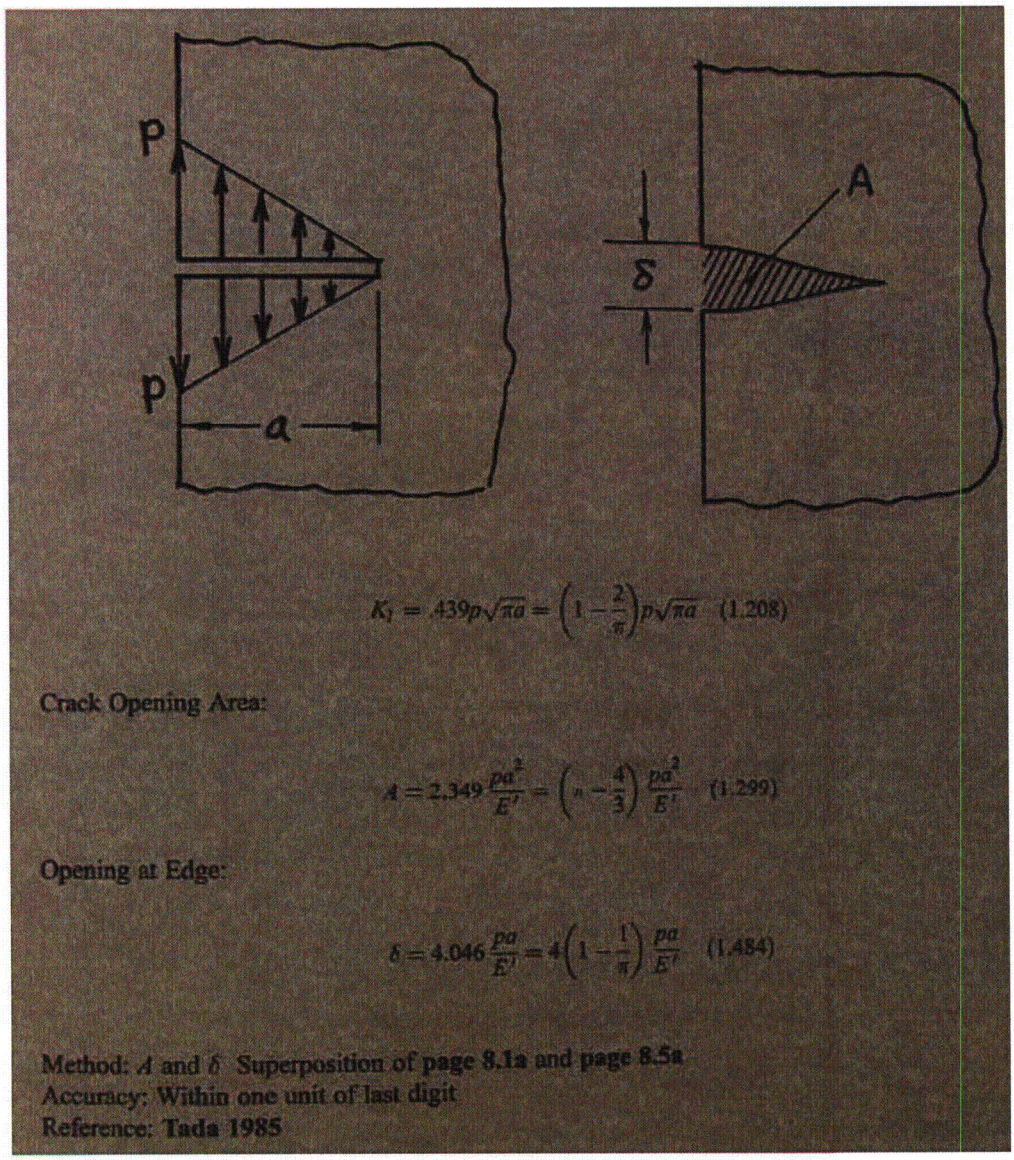


Figure 4-2. Stress Intensity Factor Solution for an Edge Cracked Semi-Infinite Plate with Linearly Varying Crack Face Traction [9].

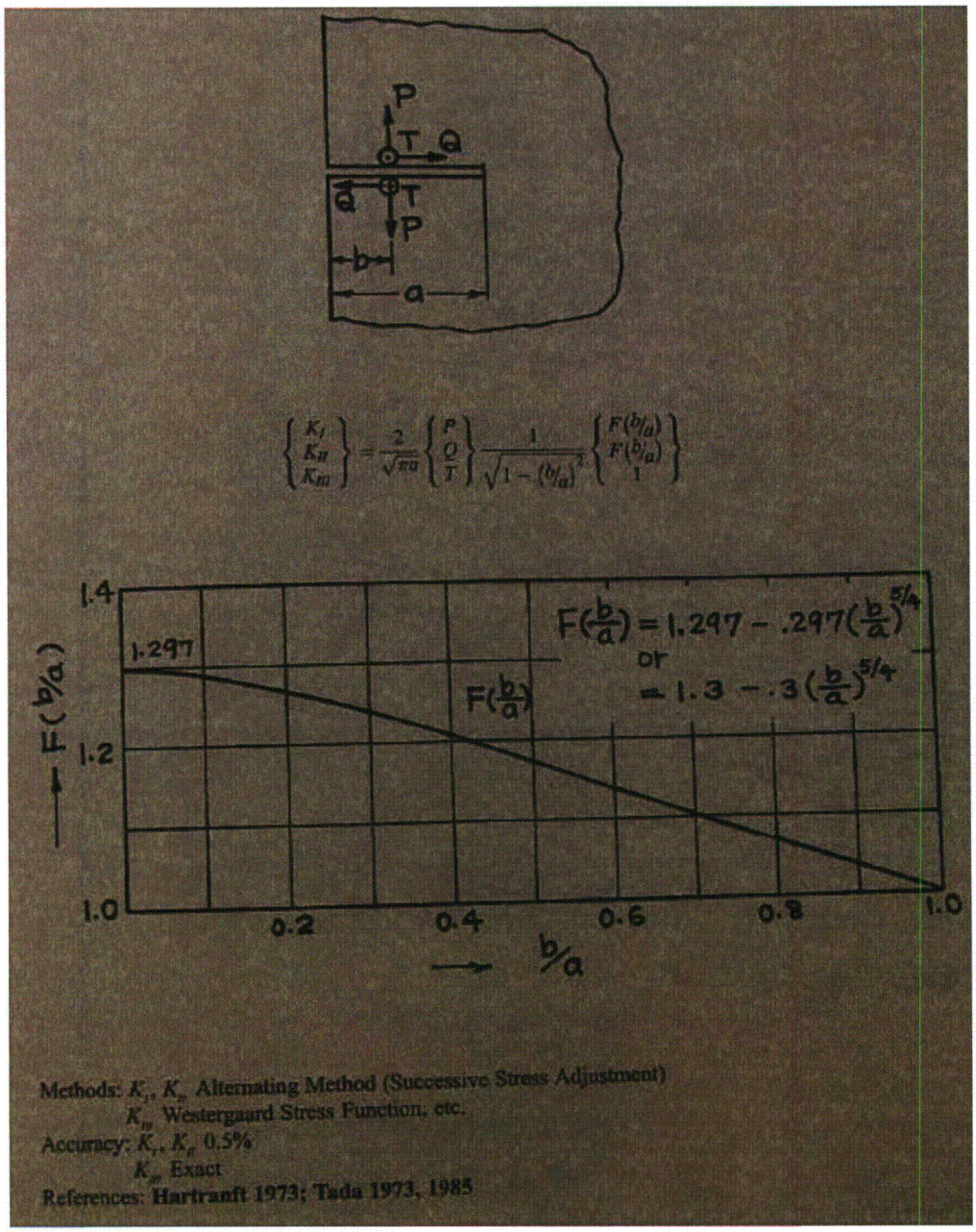
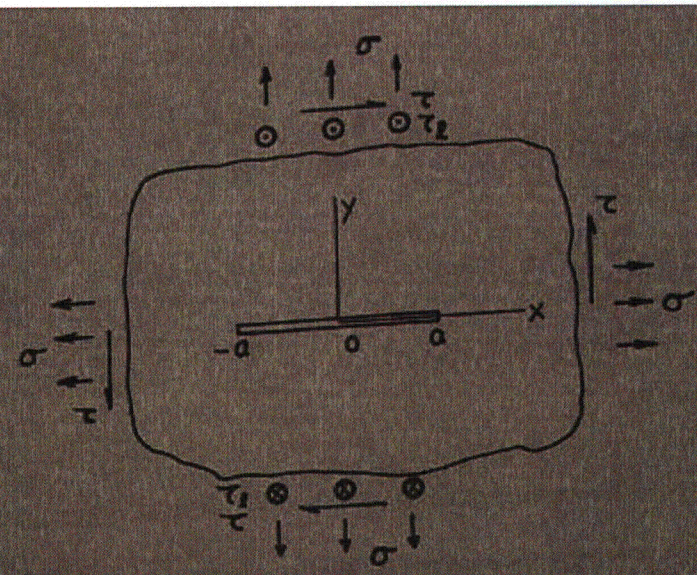


Figure 4-3. Stress Intensity Factor Solution for an Edge Cracked Semi-Infinite Plate with Point Load Applied at an Arbitrary Distance Along Crack Face [9].



$$z = x + iy$$

$$\begin{Bmatrix} Z_I(z) \\ Z_{II}(z) \\ Z_{III}(z) \end{Bmatrix} = \begin{Bmatrix} \sigma \\ \tau \\ \tau_\ell \end{Bmatrix} \frac{z}{\sqrt{z^2 - a^2}}$$

$$\begin{Bmatrix} \bar{Z}_I(z) \\ \bar{Z}_{II}(z) \\ \bar{Z}_{III}(z) \end{Bmatrix} = \begin{Bmatrix} \sigma \\ \tau \\ \tau_\ell \end{Bmatrix} \sqrt{z^2 - a^2}$$

$$\begin{Bmatrix} K_I \\ K_{II} \\ K_{III} \end{Bmatrix} = \begin{Bmatrix} \sigma \\ \tau \\ \tau_\ell \end{Bmatrix} \sqrt{\pi a}$$

$$\operatorname*{Im}_{|x| \le a} \begin{Bmatrix} \bar{Z}_I(x) \\ \bar{Z}_{II}(x) \\ \bar{Z}_{III}(x) \end{Bmatrix} = \begin{Bmatrix} \sigma \\ \tau \\ \tau_\ell \end{Bmatrix} \sqrt{a^2 - x^2}$$

Methods: Westergaard Stress Function, etc. or by Stress Concentration Factor
Accuracy: Exact
References: Griffith 1920; Westergaard 1939; Irwin 1957, 1958a; Paris 1965, etc.

Figure 4-4. Stress Intensity Factor Solution for a Center Cracked Panel [9].

5.0 FLAW EVALUATION RESULTS

This section presents the results of the flaw evaluations for the flaws summarized in Section 3.0.

5.1 V3 90°, V10 90°, V10 270° Indications

Appendix A tabulates the range of alternating nodal stress intensity output from the Monticello steam dryer uncracked FEM for the EPU FIV load case for the V3 90°, V10 90°, and V10 270° flaw locations. Each table in this Appendix lists the stress intensity output from every dryer component connected to the origin of the observed flaw for the shell elements used to model the steam dryer at the top, middle, and bottom surfaces of the element. The nodal stress intensities are averaged within the element but not across components. The maximum stress intensities occurred at the bottom surfaces of the elements at each flaw location.

Figures 5-1 through 5-3 plot the distribution of nodal stress intensity at each flaw location with respect to the global Z-axis. Figures 5-1 through 5-3 summarize the data contained in Appendix A. The global Cartesian coordinate system origin is placed at the center of the steam dryer at the elevation of the top of the upper support ring. The Z-axis is oriented along the vertical axis of the RPV, the X-axis is oriented parallel to the line bisecting the pairs of main steam nozzles, and the Y-axis is orthogonal to the X and Z-axes. Stresses for the first 12" away from the origin of each crack are shown so that the behavior of the range of alternating stress intensity factor could be determined for each flaw for dimensions greater than those reported in the INR. This is done to assess the expected FCG rate.

Figure 5-4 illustrates the bounding composite stress intensity distribution formed from the stress intensity distributions given in Figures 5-1 through 5-3. Also shown on Figure 5-4 is the stress intensity distribution, is conservatively scaled by a weld factor of 1.8 to include peak stress effects. Figure 5-5 is the scaled bounding composite stress intensity distribution expressed with respect to an origin placed at the free surface of the edge crack. Also shown on this figure are the stress intensity distributions used with Equation (1c) and the polynomial stress intensity distribution used with Equation (2c). Table 5-1 lists the nodal stress intensity values plotted in Figures 5-4 and 5-5.

Table 5-2 summarizes the range of alternating stress intensity factors calculated for the indications in the V3 and V10 welds for various crack lengths. The numerical integration performed to evaluate Equation (2c) was performed using a db=0.001 in. The small differential crack face length was chosen to reduce numerical error in the integration. The longest flaw at this location, shown in Table 2-1, is 1.375". All crack lengths considered in the LEFM calculation (2.6", 5", 10.3") are longer than the maximum observed flaw length in the subject welds. For a conservative flaw length of 2.6", the range of alternating stress intensity factor is less than 3 ksi-in$^{0.5}$. The flaw lengths of 2.6" and 10.3" were chosen based upon the mesh density of the uncracked model and were a convenient dimension to determine the stress intensity distributions needed for Equation (1c). The ΔK_I for a 10.3" flaw is also shown to be less than 3 ksi-in$^{0.5}$ using the Method 2 solution.

These results indicate that the stress distribution in this component produces a stable K field that does not exhibit a significant increase in crack driving force as a flaw increases in length. Further, review of the FCG growth correlations for Austenitic stainless steel in an air environment given in Figure C-8410-1 shows that, for an R ratio of 0.9 (the largest given in this figure), insignificant fatigue crack growth occurs for a $\Delta K_I < 3$ ksi-in$^{0.5}$. The analytical results taken into consideration with the previous field experience for these indications as well as similar indications in other steam dryers support the conclusion that the indications observed in the V3 and V10 vertical welds will not propagate further during the next operational cycle. Although further crack growth is considered to be unlikely, SI recommends that these indications, as well as the equivalent location in the uncracked weld V3-270°, be inspected during the next refueling outage to monitor any possible change in crack length.

5.2 215° Dryer Support Bracket Guide Channel Indication

Table 5-3 summarizes the maximum range of alternating stress intensity and ΔK_I from the 12" x 6" region around the guide channel indication. The maximum ΔK_I predicted for this flaw location for a flaw length of 0.75" (a=0.375") is 0.43 ksi-in$^{0.5}$. Review of the FCG growth correlations for Austenitic stainless steel in an air environment given in Figure C-8410-1 shows that, for an R ratio of 0.9 (the largest given in this figure), insignificant fatigue crack growth occurs for a $\Delta K_I < 3$ ksi-in$^{0.5}$.

These analytical results taken into consideration with the previous field experience for this indication as well as similar indications in other steam dryers support the conclusion that the indication observed in Guide Channel will not propagate further during the next operational cycle. Although further crack growth is considered to be unlikely, SI recommends that this indication be inspected during the next refueling outage to monitor any possible change in crack length.

5.3 Drain Channel "F" Access Hole Cover Plate Indications

Recognizing that the flaws observed in this component exhibit characteristics of IGSCC, a crack growth contribution from IGSCC is calculated for the next operational cycle and given below as:

$$\Delta a_{IGSCC} = 2yr \cdot 365.25 \frac{days}{yr} \cdot 24 \frac{hr}{day} \cdot 5E-5 \frac{in}{hr} \cdot 2 \frac{tips}{flaw} = 1.75 \frac{in}{flaw}$$

Table 5-3 summarizes the maximum range of alternating stress intensity and ΔK_I from the ~10" x 8" region evaluated for the indications identified in the Access Hole Cover Plate. For a conservative flaw length of 10" (a=5") the predicted $\Delta K_I = 1.72$ ksi-in$^{0.5}$. Note that the largest flaw is 8" assuming that multiple flaws on the top of the cover plate coalesce during the next cycle. Review of the FCG growth correlations for Austenitic stainless steel in an air environment given in Figure C-8410-1 shows that, for an R ratio of 0.9 (the largest given in this figure), insignificant fatigue crack growth occurs for a $\Delta K_I < 3$ ksi-in$^{0.5}$.

Given the indication dimensions shown in Figure 2-7 there is expected to be approximately 10" of uncracked ligament remaining around the perimeter of the access hole cover plate at the end of the next operational cycle. Also given that the EPU RIPDs listed in Reference [7] are bounded by the CLTP RIPDs considered in the previous flaw evaluation for this indication [5], the conclusions regarding the ability of the remaining ligament to react the applied operating load without collapse remain valid. The access hole cover plate is not expected to fail and will not become a loose part during the next operational cycle.

Table 5-1. Tabulation of Nodal Stress Intensity Distributions Given in Figures 5-5 and 5-6.

z, in	z-corrected[1], in	V3-90°, psi	V10-90°, psi	V10-270°, psi	Overall Composite, psi	Overall Comp x SCF, psi
51.3	10.3	77	83	129	129	232
53.8	7.7	97	97	103	103	185
56.4	5.1	218	150	61	218	392
58.9	2.6	240	175	118	240	432
61.5	0.0	527	597	609	609	1096

1. Origin shifted from global X,Y,Z to crack origin.

Table 5-2. Summary of Alternating Stress Intensity Factor for Various Crack Lengths in V3 & V10 Welds.

Method	a, in	$\Delta K_{I\,M}$[1], ksi-in$^{0.5}$	$\Delta K_{I\,B}$[2], ksi-in$^{0.5}$	$\Delta K_{I\,Total}$, ksi-in$^{0.5}$
1 (See Equation 1c)	2.6	1.38	0.83	2.21
	10.3	1.49	2.16	3.65
2 (See Equation 2c)	2.6	-	-	2.18
	5	-	-	2.22
	10.3	-	-	2.29

1. M refers to constant stress distribution
2. B refers to linearly varying stress distribution

Table 5-3. Summary of Alternating Stress Intensity Factor for Guide Bracket and Access Hole Cover Plate Indications.

Component	Crack Length, in	σ, psi	$\Delta K_{I\,B}^{2}$, ksi-in$^{0.5}$
215° Dryer Support Bracket Guide Channel	0.75	400	0.43
Access Hole Cover Plate[1,2]	10	435	1.72

1. Weld factor of 1.8 is conservatively applied to FEM stress output.
2. Crack half length is 5".

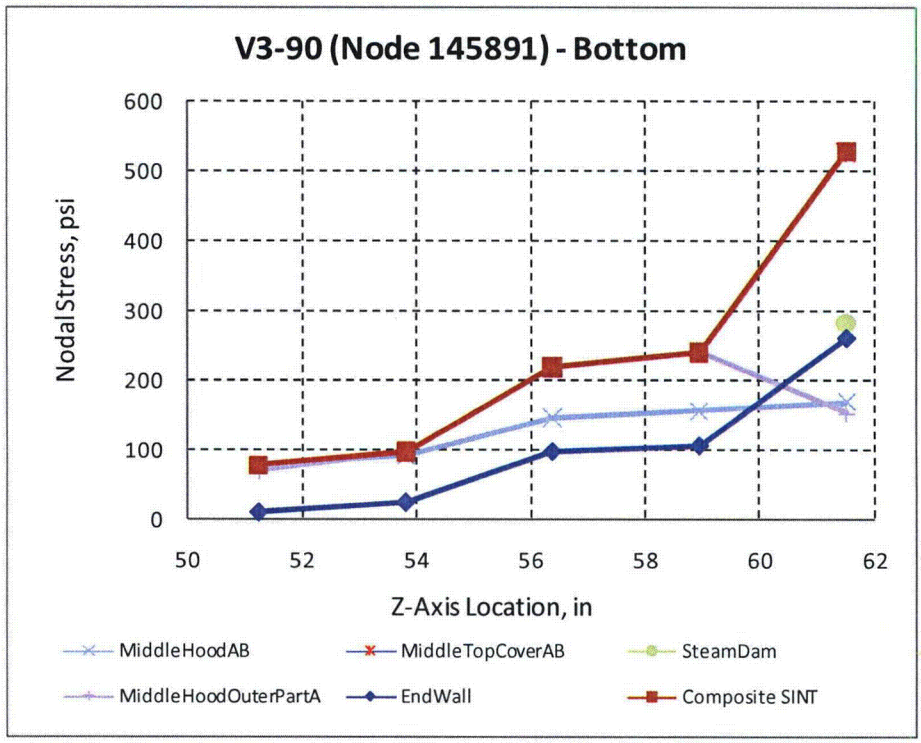


Figure 5-1. Component Stress Intensity Distributions for V3-90° Flaw Location.

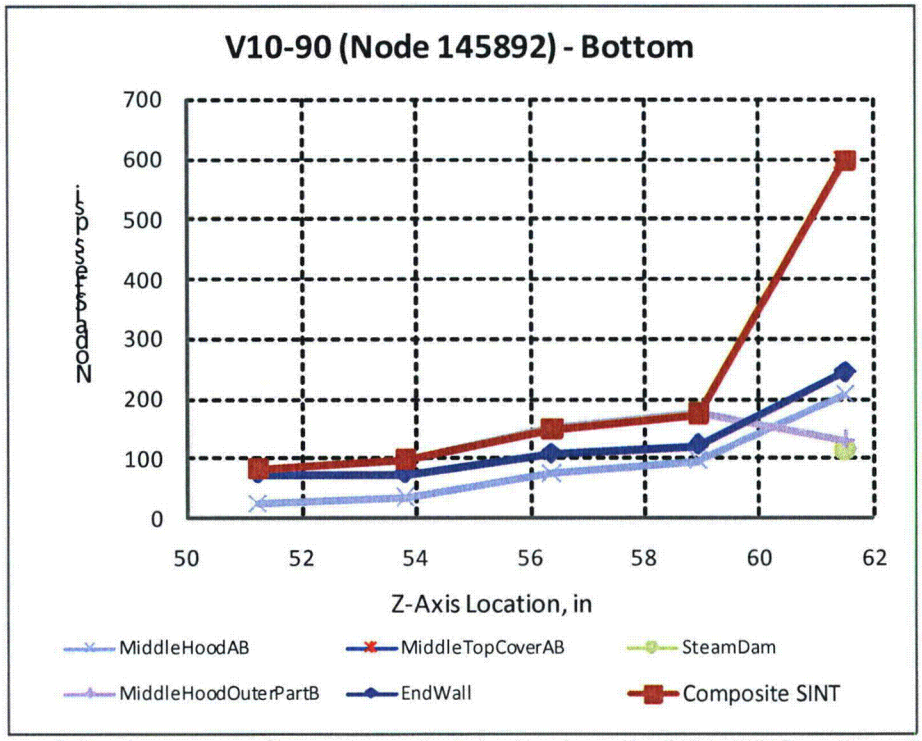


Figure 5-2. Component Stress Intensity Distributions for V10-90° Flaw Location.

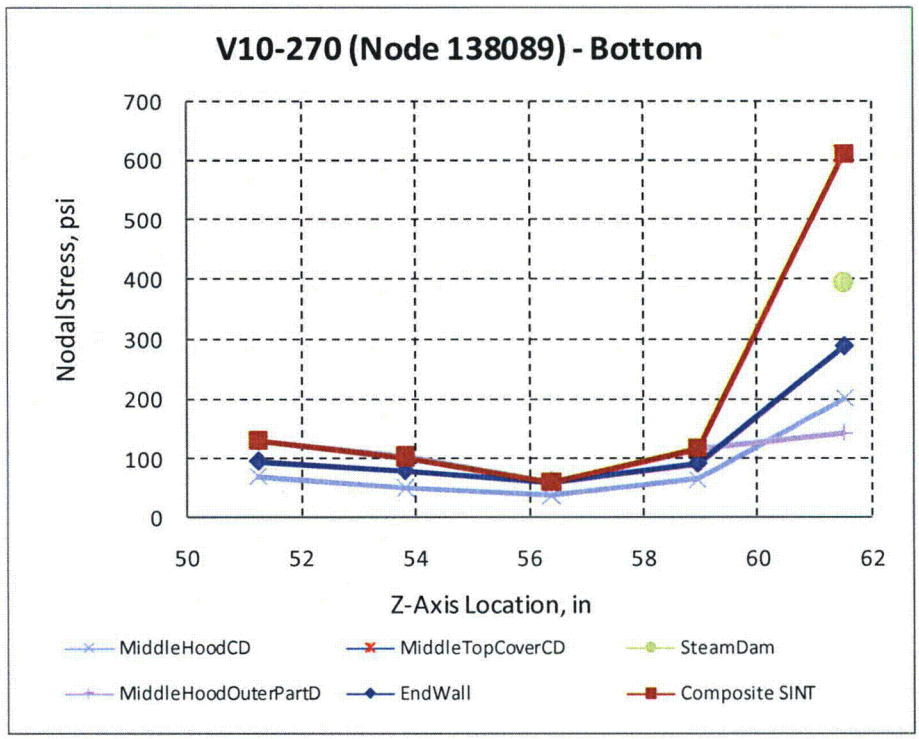


Figure 5-3. Component Stress Intensity Distributions for V10-270° Flaw Location.

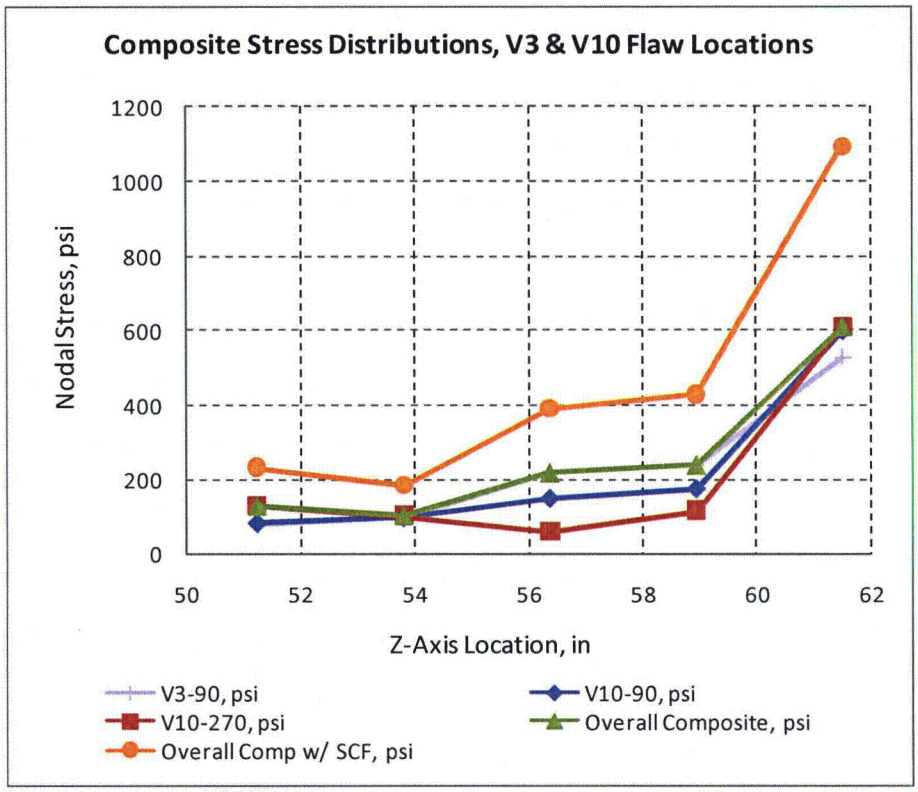


Figure 5-4. Composite Stress Intensity Distributions for all Flaw Locations.

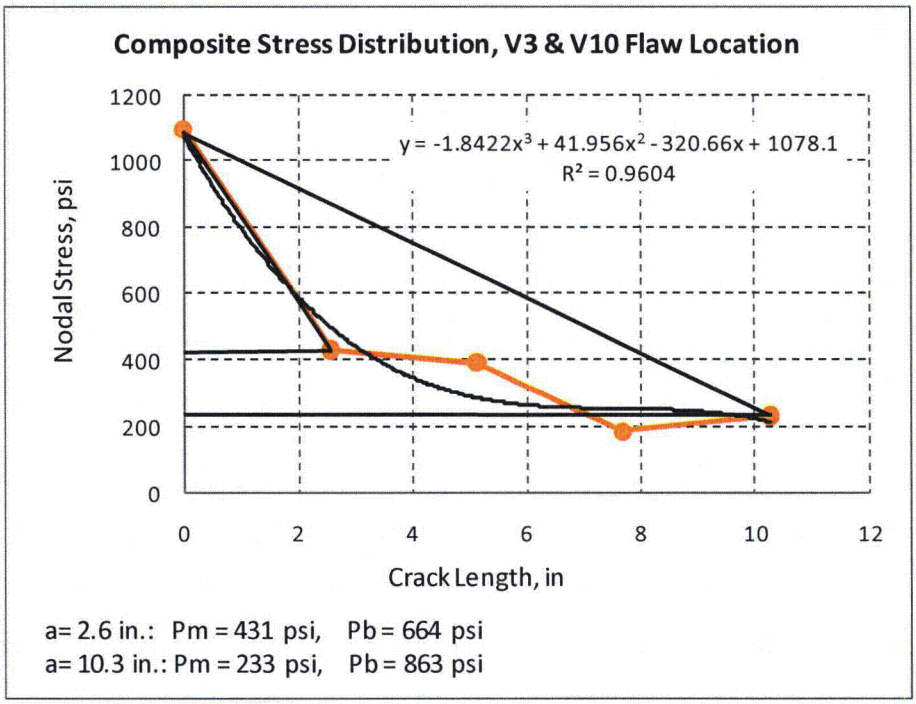


Figure 5-5. Scaled Composite Stress Intensity Distributions for all Flaw Locations, corrected to Flaw Origin and Curve-fit.

6.0 VIBRATION ASSESSMENT

This section documents the vibration assessment performed for the cracking observed during the 2005 and 2007 inspections. Each flaw is addressed separately below.

6.1 V3 90°, V10 90°, V10 270° Indications

Observation of the INRs, contained as Attachments 1 through 4, demonstrates the following:

1. The cracks are very short (less than 2" long).
2. The vertical plates are attached to the inner hoods with a double sided fillet weld
3. The cracking exists in only one of the two fillet welds for two of the three plates. The third plate shows cracking in both fillet welds for ¼ inch only.

The short crack lengths are considered to be insignificant with respect to the overall plate dimensions; therefore, existence of cracking is also considered to introduce an insignificant effect on the boundary conditions applied to the plate. Further, even if one fillet weld is cracked, the second fillet weld restrains motion of the plate preserving the boundary condition. For the one plate that exhibits cracking in both fillet welds, it should be noted that the weld configuration and orientation is such that lateral vibration of the plates would tend to close the crack imposing a boundary condition on the plate similar to that contributed to an uncracked weld. The existence of short cracking in the V3 and V10 welds does not affect the significant modes of vibration of the affected plates, since the vibration frequency is governed by the short length of the plate and the indication is along the long edge of the plate..

6.2 215° Dryer Support Bracket Guide Channel Indication

The flaw in this component is approximately ¾" long. Existence of a single small crack in a long channel with length significantly greater than the crack length will cause an insignificant effect on the global stiffness of the component and consequently have no effect on the significant modes of vibration of the channel.

6.3 Drain Channel "F" Access Hole Cover Plate Indications

Figure 2-7 shows cracking oriented normal to and parallel with the weld seam at the boundary of the plate. Cracks running along the plate effectively removes the boundary condition at the edge of the plate for the length of the crack; however, the cracks are bounded at each side by uncracked plate material which ensures that the cracked section of plate is effectively restrained along the original boundary. For short cracking, as observed at Monticello, any effect on mode shape and frequency of the plate would only be expected at very high modes with associated natural frequencies above the frequency band of concern for FIV.

For the cracks oriented normal to the weld seam, it is possible that the loss of membrane stiffness at the crack location could affect the normal modes of the plate for longer cracks located at locations of high modal displacement. However, note that the cover plate is located close to the boundary of the plate. The plate boundary requires that the modal displacements all along the boundary be zero; therefore, for short cracks close to the plate boundary the modal displacements will be negligible. The loss of stiffness local to the crack and close to the boundary will not affect the normal modes of the plate.

7.0 CONCLUSIONS

Given the inspection history of the subject indications, the operating experience for this and other BWR steam dryers with similar indications, and the LEFM results documented in this report, the following conclusions are made:

1. The subject indications are not expected to exhibit further fatigue crack growth.

2. The IGSCC indications in the access hole cover plate are predicted to experience further IGSCC growth; however, the ligament remaining at the end of the next operational cycle is adequate to react the applied loading and prevent collapse of this component.

3. None of the indications considered in this evaluation have the potential to create loose parts during the next operational cycle.

4. All indications should be inspected during the next refueling outage to establish current flaw dimensions.

5. The uncracked vertical weld, V3-270°, should be inspected during the next refueling outage.

Further, the short cracks observed in the Monticello steam dryer will not affect the vibration response of the steam dryer sufficiently such that the FEM created for the EPU stress analysis needs to be modified to incorporate cracking.

8.0 REFERENCES

1. BWRVIP-139: BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines, EPRI, Palo Alto, CA, 2005. 1011463.
2. Monticello RF0-22 Steam Dryer In-Vessel Visual Inspection Final Report. AREVA. 2005, SI File No. 0800760.204.
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5. Engineering Evaluation 10451 contained in Design Information Transmittal EPU-0284, SI File No. 0800760.201.
6. Monticello Steam Dryer Extended Power Uprate FIV stresses, SI File No. 0800760.203.
7. Peters, S. "Task T0304: Reactor Internal Pressure Differences, Fuel Lift Margin, CRGT Lift Force, Acoustic and Flow Induced Loads," GEH, GE-NE-0060-9039-TR-R0. August 2007. **GEH Proprietary Information.** SI File No. 0800760.201P.
8. Nuclear Management Company Design Information Transmittal EPU-0284. September 30, 2008. Monticello Unit 1, SI File No. 0800760.201.
9. Tada, Hiroshi, and Paul C. Paris, George R. Irwin. The Stress Analysis of Cracks, 3rd Edition. New York: ASME, 2000.

APPENDIX A

MONTICELLO STEAM DRYER FINITE ELEMENT MODEL EPU FIV RANGE OF ALTERNATING STRESS INTENSITY OUTPUT AT ALL FLAW LOCATIONS

Notes:

1. ICOMP is the component number
2. BMU (bottom, middle, upper) indicates which level the stress is recorded at
3. xyz - physical location of the node, in
4. nx, ny, nz - shell normal (pointing from bottom to top of shell) for the given component
5. sxx -szx - the range of alternating stress components, psi
6. SINT - the range of alternating stress intensity, psi
7. X and Y coordinate values not shown in Tables A-2 through A-4 because all data is extracted along a vertical line parallel to the Z axis.

Table A-1. ANSYS Component Identification for V3 and V10 EPU FIV Load Case Stress Intensity Output

ICOMP			Thickness, in	Component Name
V3 90°	V10 90°	V10 270°		
31	31	33	0.5	MiddleHoodAB
35	35	37	0.5	MiddleTopCoverAB
49	48	52	0.5	SteamDam
216	215	224	0.5	MiddleHoodOuterPartA
221	220	230	0.5	EndWall

Table A-2. V3 90° FEM Stress Intensity Output – EPU FIV Load Case

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
144457	31	1	51.3	1	0	0	0.0	76.2	28.2	0.0	7.2	0.0	77.3
145704	31	1	53.8	1	0	0	0.0	89.6	35.8	0.0	9.8	0.0	91.3
141663	31	1	56.4	1	0	0	0.0	145.5	54.9	0.0	2.1	0.0	145.6
141662	31	1	58.9	1	0	0	0.0	155.2	53.1	0.0	5.6	0.0	155.5
145891	31	1	61.5	1	0	0	0.0	-86.9	-89.3	0.0	79.6	0.0	167.7
145891	35	1	61.5	0	0	1	-525.0	-150.1	0.0	-25.5	0.0	0.0	526.7
145891	49	1	61.5	0	-1	0	-241.3	0.0	-107.2	0.0	0.0	-83.4	281.2
144457	216	1	51.3	1	0	0	0.0	69.5	26.4	0.0	-3.7	0.0	69.8
145704	216	1	53.8	1	0	0	0.0	96.7	37.7	0.0	4.7	0.0	97.1
141663	216	1	56.4	1	0	0	0.0	218.0	92.1	0.0	7.2	0.0	218.4
141662	216	1	58.9	1	0	0	0.0	238.1	92.7	0.0	-14.5	0.0	239.6
145891	216	1	61.5	1	0	0	0.0	-74.2	-93.0	0.0	-68.8	0.0	153.0
144457	221	1	51.3	0	1	0	5.8	0.0	5.4	0.0	0.0	5.3	10.9
145704	221	1	53.8	0	1	0	24.4	0.0	15.2	0.0	0.0	2.0	24.8
141663	221	1	56.4	0	1	0	92.5	0.0	45.1	0.0	0.0	16.3	97.6
141662	221	1	58.9	0	1	0	85.9	0.0	21.3	0.0	0.0	41.2	105.9
145891	221	1	61.5	0	1	0	41.2	0.0	58.1	0.0	0.0	129.3	259.2

Table A-2. V3 90° FEM Stress Intensity Output – EPU FIV Load Case, cont.

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
144457	31	2	51.3	1	0	0	0.0	-2.1	3.6	0.0	3.2	0.0	8.6
145704	31	2	53.8	1	0	0	0.0	-0.4	8.0	0.0	3.9	0.0	11.5
141663	31	2	56.4	1	0	0	0.0	1.3	13.9	0.0	6.1	0.0	17.6
141662	31	2	58.9	1	0	0	0.0	6.1	18.5	0.0	12.1	0.0	27.3
145891	31	2	61.5	1	0	0	0.0	4.3	19.9	0.0	19.7	0.0	42.4
145891	35	2	61.5	0	0	1	-69.6	0.2	0.0	-1.5	0.0	0.0	69.8
145891	49	2	61.5	0	-1	0	-98.7	0.0	-10.2	0.0	0.0	-51.9	136.3
144457	216	2	51.3	1	0	0	0.0	-2.9	3.5	0.0	-6.1	0.0	13.7
145704	216	2	53.8	1	0	0	0.0	-1.3	7.8	0.0	-5.8	0.0	14.7
141663	216	2	56.4	1	0	0	0.0	0.4	15.3	0.0	-6.3	0.0	19.5
141662	216	2	58.9	1	0	0	0.0	4.8	23.3	0.0	-5.9	0.0	25.1
145891	216	2	61.5	1	0	0	0.0	17.5	24.8	0.0	-16.0	0.0	37.6
144457	221	2	51.3	0	1	0	3.1	0.0	5.8	0.0	0.0	7.8	15.8
145704	221	2	53.8	0	1	0	6.7	0.0	12.9	0.0	0.0	7.3	17.7
141663	221	2	56.4	0	1	0	15.2	0.0	20.2	0.0	0.0	13.7	31.6
141662	221	2	58.9	0	1	0	27.2	0.0	4.0	0.0	0.0	31.1	66.3
145891	221	2	61.5	0	1	0	-95.0	0.0	7.2	0.0	0.0	115.6	252.8

Table A-2. V3 90° FEM Stress Intensity Output – EPU FIV Load Case, cont.

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
144457	31	3	51.3	1	0	0	0.0	-80.4	-21.0	0.0	-0.8	0.0	80.5
145704	31	3	53.8	1	0	0	0.0	-90.4	-19.9	0.0	-2.0	0.0	90.5
141663	31	3	56.4	1	0	0	0.0	-142.8	-27.1	0.0	10.2	0.0	143.7
141662	31	3	58.9	1	0	0	0.0	-143.0	-16.0	0.0	18.7	0.0	145.7
145891	31	3	61.5	1	0	0	0.0	95.6	129.0	0.0	-40.1	0.0	155.7
145891	35	3	61.5	0	0	1	385.9	150.4	0.0	22.4	0.0	0.0	388.0
145891	49	3	61.5	0	-1	0	44.0	0.0	86.7	0.0	0.0	-20.4	94.8
144457	216	3	51.3	1	0	0	0.0	-75.4	-19.3	0.0	-8.4	0.0	76.6
145704	216	3	53.8	1	0	0	0.0	-99.3	-22.0	0.0	-16.2	0.0	102.5
141663	216	3	56.4	1	0	0	0.0	-217.2	-61.6	0.0	-19.9	0.0	219.7
141662	216	3	58.9	1	0	0	0.0	-228.6	-46.0	0.0	2.6	0.0	228.7
145891	216	3	61.5	1	0	0	0.0	109.1	142.7	0.0	36.8	0.0	166.4
144457	221	3	51.3	0	1	0	0.5	0.0	6.2	0.0	0.0	10.3	21.4
145704	221	3	53.8	0	1	0	-10.9	0.0	10.6	0.0	0.0	12.6	33.1
141663	221	3	56.4	0	1	0	-62.2	0.0	-4.8	0.0	0.0	11.2	64.3
141662	221	3	58.9	0	1	0	-31.6	0.0	-13.4	0.0	0.0	21.0	45.8
145891	221	3	61.5	0	1	0	-231.3	0.0	-43.7	0.0	0.0	101.8	276.9

Max	61.5
Min	51.25
Range	10.25

Table A-3. V10 90° FEM Stress Intensity Output – EPU FIV Load Case

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
142894	31	1	51.3	1	0	0	0.0	15.8	8.6	0.0	12.0	0.0	25.0
142893	31	1	53.8	1	0	0	0.0	30.5	17.9	0.0	9.5	0.0	35.5
142899	31	1	56.4	1	0	0	0.0	72.4	33.5	0.0	8.4	0.0	74.1
142898	31	1	58.9	1	0	0	0.0	95.2	29.3	0.0	-7.0	0.0	95.9
145892	31	1	61.5	1	0	0	0.0	-147.2	-91.3	0.0	-80.7	0.0	204.7
145892	35	1	61.5	0	0	1	-596.2	-168.8	0.0	17.1	0.0	0.0	596.9
145892	48	1	61.5	0	-1	0	55.2	0.0	84.8	0.0	0.0	-40.1	112.8
142894	215	1	51.3	1	0	0	0.0	77.7	27.5	0.0	16.5	0.0	82.6
142893	215	1	53.8	1	0	0	0.0	94.9	38.4	0.0	11.5	0.0	97.2
142899	215	1	56.4	1	0	0	0.0	148.7	73.7	0.0	8.3	0.0	149.6
142898	215	1	58.9	1	0	0	0.0	174.4	62.3	0.0	10.9	0.0	175.4
145892	215	1	61.5	1	0	0	0.0	-62.4	-73.1	0.0	62.1	0.0	130.1
142894	220	1	51.3	0	-1	0	68.0	0.0	23.1	0.0	0.0	15.8	73.0
142893	220	1	53.8	0	-1	0	70.0	0.0	28.4	0.0	0.0	11.9	73.1
142899	220	1	56.4	0	-1	0	101.5	0.0	46.0	0.0	0.0	18.8	107.3
142898	220	1	58.9	0	-1	0	108.8	0.0	29.8	0.0	0.0	32.7	120.5
145892	220	1	61.5	0	-1	0	59.7	0.0	64.4	0.0	0.0	121.1	242.3

Table A-3. V10 90° FEM Stress Intensity Output – EPU FIV Load Case, cont.

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
142894	31	2	51.3	1	0	0	0.0	-3.1	2.5	0.0	0.5	0.0	5.7
142893	31	2	53.8	1	0	0	0.0	-1.8	7.0	0.0	0.0	0.0	8.9
142899	31	2	56.4	1	0	0	0.0	-0.9	12.7	0.0	-2.2	0.0	14.2
142898	31	2	58.9	1	0	0	0.0	2.5	17.8	0.0	-6.9	0.0	20.7
145892	31	2	61.5	1	0	0	0.0	-5.2	19.9	0.0	-14.1	0.0	37.8
145892	35	2	61.5	0	0	1	-116.1	-16.9	0.0	-0.7	0.0	0.0	116.1
145892	48	2	61.5	0	-1	0	-133.2	0.0	-11.4	0.0	0.0	-71.3	187.6
142894	215	2	51.3	1	0	0	0.0	-2.5	2.6	0.0	4.5	0.0	10.3
142893	215	2	53.8	1	0	0	0.0	-1.9	7.0	0.0	4.5	0.0	12.7
142899	215	2	56.4	1	0	0	0.0	-0.4	13.1	0.0	4.5	0.0	16.2
142898	215	2	58.9	1	0	0	0.0	1.2	22.0	0.0	6.6	0.0	24.7
145892	215	2	61.5	1	0	0	0.0	13.9	27.0	0.0	18.3	0.0	39.8
142894	220	2	51.3	0	-1	0	3.1	0.0	4.6	0.0	0.0	3.2	7.1
142893	220	2	53.8	0	-1	0	4.8	0.0	9.8	0.0	0.0	2.7	11.0
142899	220	2	56.4	0	-1	0	7.1	0.0	16.8	0.0	0.0	9.1	22.3
142898	220	2	58.9	0	-1	0	20.6	0.0	3.4	0.0	0.0	26.3	55.3
145892	220	2	61.5	0	-1	0	-121.0	0.0	-0.7	0.0	0.0	111.8	253.8

Table A-3. V10 90° FEM Stress Intensity Output – EPU FIV Load Case, cont.

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
142894	31	3	51.3	1	0	0	0.0	-22.1	-3.6	0.0	-11.0	0.0	28.7
142893	31	3	53.8	1	0	0	0.0	-34.2	-3.8	0.0	-9.4	0.0	36.8
142899	31	3	56.4	1	0	0	0.0	-74.2	-8.2	0.0	-12.8	0.0	76.6
142898	31	3	58.9	1	0	0	0.0	-90.3	6.3	0.0	-6.9	0.0	97.5
145892	31	3	61.5	1	0	0	0.0	136.9	131.2	0.0	52.5	0.0	186.6
145892	35	3	61.5	0	0	1	364.0	135.0	0.0	-18.6	0.0	0.0	365.5
145892	48	3	61.5	0	-1	0	-321.6	0.0	-107.6	0.0	0.0	-102.6	362.8
142894	215	3	51.3	1	0	0	0.0	-82.8	-22.2	0.0	-7.5	0.0	83.7
142893	215	3	53.8	1	0	0	0.0	-98.7	-24.3	0.0	-2.5	0.0	98.8
142899	215	3	56.4	1	0	0	0.0	-149.5	-47.4	0.0	0.7	0.0	149.5
142898	215	3	58.9	1	0	0	0.0	-172.0	-18.3	0.0	2.4	0.0	172.0
145892	215	3	61.5	1	0	0	0.0	90.1	127.1	0.0	-25.5	0.0	140.1
142894	220	3	51.3	0	-1	0	-61.8	0.0	-14.0	0.0	0.0	-9.4	63.6
142893	220	3	53.8	0	-1	0	-60.5	0.0	-8.7	0.0	0.0	-6.4	61.3
142899	220	3	56.4	0	-1	0	-87.3	0.0	-12.4	0.0	0.0	-0.6	87.3
142898	220	3	58.9	0	-1	0	-67.6	0.0	-23.0	0.0	0.0	19.8	75.1
145892	220	3	61.5	0	-1	0	-301.7	0.0	-65.8	0.0	0.0	102.4	339.9

Max	61.5
Min	51.25
Range	10.25

Table A-4. V10 270° FEM Stress Intensity Output – EPU FIV Load Case

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
139141	33	1	51.3	-1	0	0	0.0	-40.0	-12.0	0.0	32.5	0.0	70.7
139142	33	1	53.8	-1	0	0	0.0	-32.0	-14.3	0.0	24.2	0.0	51.6
139143	33	1	56.4	-1	0	0	0.0	-33.0	-18.3	0.0	9.6	0.0	37.7
139144	33	1	58.9	-1	0	0	0.0	-59.9	-2.6	0.0	-16.3	0.0	65.9
138089	33	1	61.5	-1	0	0	0.0	153.2	48.8	0.0	-83.3	0.0	199.2
138089	37	1	61.5	0	0	1	608.0	153.7	0.0	-16.1	0.0	0.0	608.6
138089	52	1	61.5	0	-1	0	350.6	0.0	98.3	0.0	0.0	-112.6	393.5
139141	224	1	51.3	-1	0	0	0.0	-111.0	-32.6	0.0	42.1	0.0	129.4
139142	224	1	53.8	-1	0	0	0.0	-83.0	-31.8	0.0	38.1	0.0	103.3
139143	224	1	56.4	-1	0	0	0.0	-53.9	-32.2	0.0	12.9	0.0	59.9
139144	224	1	58.9	-1	0	0	0.0	-117.7	-26.7	0.0	5.2	0.0	118.0
138089	224	1	61.5	-1	0	0	0.0	99.2	42.8	0.0	65.4	0.0	142.3
139141	230	1	51.3	0	1	0	-76.3	0.0	-19.9	0.0	0.0	38.1	95.5
139142	230	1	53.8	0	1	0	-59.6	0.0	-22.9	0.0	0.0	32.8	78.9
139143	230	1	56.4	0	1	0	-47.8	0.0	-28.6	0.0	0.0	20.6	60.9
139144	230	1	58.9	0	1	0	-74.0	0.0	-16.5	0.0	0.0	35.4	91.2
138089	230	1	61.5	0	1	0	32.3	0.0	-38.9	0.0	0.0	139.1	287.2

Table A-4. V10 270° FEM Stress Intensity Output – EPU FIV Load Case, cont.

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
139141	33	2	51.3	-1	0	0	0.0	4.1	2.1	0.0	0.3	0.0	4.1
139142	33	2	53.8	-1	0	0	0.0	1.2	-4.8	0.0	-0.4	0.0	6.1
139143	33	2	56.4	-1	0	0	0.0	-1.5	-12.8	0.0	-2.3	0.0	13.2
139144	33	2	58.9	-1	0	0	0.0	-6.0	-18.9	0.0	-6.9	0.0	21.9
138089	33	2	61.5	-1	0	0	0.0	2.0	-21.0	0.0	-14.5	0.0	37.0
138089	37	2	61.5	0	0	1	168.4	22.7	0.0	4.3	0.0	0.0	168.5
138089	52	2	61.5	0	-1	0	193.2	0.0	14.2	0.0	0.0	-93.1	258.3
139141	224	2	51.3	-1	0	0	0.0	2.8	1.9	0.0	7.6	0.0	15.3
139142	224	2	53.8	-1	0	0	0.0	0.7	-5.0	0.0	7.9	0.0	16.8
139143	224	2	56.4	-1	0	0	0.0	-3.4	-12.7	0.0	9.9	0.0	21.8
139144	224	2	58.9	-1	0	0	0.0	-4.8	-23.1	0.0	14.0	0.0	33.4
138089	224	2	61.5	-1	0	0	0.0	-16.3	-27.5	0.0	26.6	0.0	54.4
139141	230	2	51.3	0	1	0	-3.1	0.0	0.2	0.0	0.0	4.8	10.1
139142	230	2	53.8	0	1	0	-0.2	0.0	-5.4	0.0	0.0	6.3	13.5
139143	230	2	56.4	0	1	0	5.9	0.0	-13.4	0.0	0.0	11.2	29.6
139144	230	2	58.9	0	1	0	-13.4	0.0	2.2	0.0	0.0	40.0	81.6
138089	230	2	61.5	0	1	0	166.5	0.0	6.2	0.0	0.0	136.9	317.3

Table A-4. V10 270° FEM Stress Intensity Output – EPU FIV Load Case, cont.

Node	ICOMP	BMU	Z, in	nx	ny	nz	Sxx, psi	Syy, psi	Szz, psi	Sxy, psi	Syz, psi	Szx, psi	SINT, psi
139141	33	3	51.3	-1	0	0	0.0	48.1	16.3	0.0	-31.9	0.0	71.2
139142	33	3	53.8	-1	0	0	0.0	34.4	4.6	0.0	-25.0	0.0	58.2
139143	33	3	56.4	-1	0	0	0.0	30.0	-7.2	0.0	-14.2	0.0	46.8
139144	33	3	58.9	-1	0	0	0.0	48.0	-35.2	0.0	2.5	0.0	83.3
138089	33	3	61.5	-1	0	0	0.0	-149.2	-90.7	0.0	54.2	0.0	181.6
138089	37	3	61.5	0	0	1	-271.3	-108.3	0.0	24.7	0.0	0.0	275.0
138089	52	3	61.5	0	-1	0	35.8	0.0	-69.9	0.0	0.0	-73.7	181.3
139141	224	3	51.3	-1	0	0	0.0	116.7	36.5	0.0	-26.9	0.0	124.9
139142	224	3	53.8	-1	0	0	0.0	84.3	21.7	0.0	-22.3	0.0	91.5
139143	224	3	56.4	-1	0	0	0.0	47.2	6.8	0.0	6.8	0.0	48.3
139144	224	3	58.9	-1	0	0	0.0	108.2	-19.5	0.0	22.7	0.0	135.5
138089	224	3	61.5	-1	0	0	0.0	-131.7	-97.8	0.0	-12.1	0.0	135.6
139141	230	3	51.3	0	1	0	70.1	0.0	20.3	0.0	0.0	-28.6	83.1
139142	230	3	53.8	0	1	0	59.1	0.0	12.1	0.0	0.0	-20.3	66.7
139143	230	3	56.4	0	1	0	59.7	0.0	1.8	0.0	0.0	1.8	59.7
139144	230	3	58.9	0	1	0	47.2	0.0	21.0	0.0	0.0	44.7	93.1
138089	230	3	61.5	0	1	0	300.7	0.0	51.4	0.0	0.0	134.8	367.1

Max	61.5
Min	51.25
Range	10.25

Table A-5. Guide Channel FEM Stress Intensity Output – EPU FIV Load Case

node	icomp	BMU	x	y	z	nx	ny	nz	sxx	syy	szz	sxy	syz	szx	SI
138631	357	1	-55.75	-83.32	-57.63	-0.545	-0.838	0.000	-214.16	-90.68	-77.45	139.36	-1.25	1.92	304.86
138631	357	2	-55.75	-83.32	-57.63	-0.545	-0.838	0.000	2.67	1.13	15.33	-1.74	-2.21	3.39	16.61
138631	357	3	-55.75	-83.32	-57.63	-0.545	-0.838	0.000	219.50	92.94	108.11	-142.83	-3.16	4.85	312.60
138635	357	1	-55.75	-83.32	-54.91	-0.545	-0.838	0.000	-243.83	-103.25	-88.27	158.67	-1.58	2.42	347.11
138635	357	2	-55.75	-83.32	-54.91	-0.545	-0.838	0.000	3.52	1.49	17.54	-2.29	-2.02	3.11	18.55
138635	357	3	-55.75	-83.32	-54.91	-0.545	-0.838	0.000	250.86	106.23	123.35	-163.24	-2.46	3.79	357.18
138636	357	1	-55.75	-83.32	-52.20	-0.545	-0.838	0.000	-271.63	-115.01	-99.99	176.75	-1.40	2.15	386.66
138636	357	2	-55.75	-83.32	-52.20	-0.545	-0.838	0.000	4.30	1.82	18.24	-2.80	-1.47	2.26	18.81
138636	357	3	-55.75	-83.32	-52.20	-0.545	-0.838	0.000	280.23	118.65	136.46	-182.34	-1.53	2.36	398.91
116917	357	1	-53.60	-84.72	-55.07	-0.524	-0.852	0.000	-104.58	-42.18	-31.64	66.39	9.46	-14.96	149.38
116917	357	2	-53.60	-84.72	-55.07	-0.524	-0.852	0.000	3.34	1.34	13.28	-2.12	-1.35	2.10	13.95
116917	357	3	-53.60	-84.72	-55.07	-0.524	-0.852	0.000	111.27	44.87	58.20	-70.63	-12.16	19.15	161.09
116869	357	1	-53.60	-84.72	-57.76	-0.524	-0.852	0.000	-94.37	-38.05	-31.80	59.90	10.19	-16.13	135.88
116869	357	2	-53.60	-84.72	-57.76	-0.524	-0.852	0.000	2.18	0.88	8.75	-1.39	-0.85	1.30	9.15
116869	357	3	-53.60	-84.72	-57.76	-0.524	-0.852	0.000	98.72	39.82	49.31	-62.67	-11.88	18.72	143.70
116916	357	1	-53.59	-84.72	-52.37	-0.524	-0.852	0.000	-113.19	-45.65	-31.13	71.85	8.45	-13.35	160.71
116916	357	2	-53.59	-84.72	-52.37	-0.524	-0.852	0.000	4.42	1.77	16.79	-2.79	-1.34	2.08	17.34
116916	357	3	-53.59	-84.72	-52.37	-0.524	-0.852	0.000	122.02	49.18	64.70	-77.43	-11.12	17.51	175.04

116868	357	1	-51.43	-86.05	-55.32	-0.502	-0.865	0.000	35.74	12.56	25.52	-21.18	12.62	-21.10	63.99
116868	357	2	-51.43	-86.05	-55.32	-0.502	-0.865	0.000	2.62	0.95	10.01	-1.58	1.29	-2.20	10.90
116868	357	3	-51.43	-86.05	-55.32	-0.502	-0.865	0.000	-30.50	-10.67	-5.50	18.03	-10.04	16.70	52.83
116867	357	1	-51.42	-86.06	-52.65	-0.502	-0.865	0.000	45.51	16.03	37.07	-27.00	11.17	-18.65	74.23
116867	357	2	-51.42	-86.06	-52.65	-0.502	-0.865	0.000	4.20	1.51	15.91	-2.51	0.77	-1.32	16.13
116867	357	3	-51.42	-86.06	-52.65	-0.502	-0.865	0.000	-37.11	-13.02	-5.25	21.97	-9.63	16.02	58.41
116837	357	1	-49.20	-87.34	-55.64	-0.480	-0.878	0.000	133.12	42.18	60.78	-74.90	9.42	-16.66	178.36
116837	357	2	-49.20	-87.34	-55.64	-0.480	-0.878	0.000	1.38	0.45	6.33	-0.79	4.79	-8.55	20.11
116837	357	3	-49.20	-87.34	-55.64	-0.480	-0.878	0.000	-130.37	-41.28	-48.12	73.33	0.16	-0.44	171.59
121460	361	1	-62.21	-78.61	-54.90	-0.628	-0.778	0.000	-8.74	-5.47	-20.83	6.91	2.82	-3.58	23.15
121460	361	2	-62.21	-78.61	-54.90	-0.628	-0.778	0.000	0.91	0.57	-15.53	-0.72	3.03	-3.84	19.62
121460	361	3	-62.21	-78.61	-54.90	-0.628	-0.778	0.000	10.56	6.60	-10.23	-8.35	3.23	-4.10	29.31
137542	361	1	-60.63	-79.84	-57.63	-0.613	-0.790	0.000	-15.82	-9.51	-30.14	12.27	3.36	-4.33	33.72
137542	361	2	-60.63	-79.84	-57.63	-0.613	-0.790	0.000	1.17	0.71	-20.75	-0.91	3.84	-4.95	25.86
137542	361	3	-60.63	-79.84	-57.63	-0.613	-0.790	0.000	18.17	10.92	-11.36	-14.09	4.31	-5.56	42.84
140799	361	1	-60.63	-79.84	-54.91	-0.613	-0.790	0.000	-16.38	-9.84	-26.19	12.70	3.60	-4.64	32.08
140799	361	2	-60.63	-79.84	-54.91	-0.613	-0.790	0.000	1.00	0.60	-16.93	-0.78	4.03	-5.20	22.73
140799	361	3	-60.63	-79.84	-54.91	-0.613	-0.790	0.000	18.38	11.05	-7.67	-14.25	4.46	-5.75	39.85
142172	361	1	-60.63	-79.84	-52.20	-0.613	-0.790	0.000	-15.92	-9.57	-21.99	12.34	3.63	-4.68	29.92

142172	361	2	-60.63	-79.84	-52.20	-0.613	-0.790	0.000	0.86	0.52	-13.32	-0.67	4.03	-5.20	19.72
142172	361	3	-60.63	-79.84	-52.20	-0.613	-0.790	0.000	17.64	10.60	-4.65	-13.68	4.43	-5.71	35.93
125905	381	1	-60.14	-76.43	-55.54	0.619	0.786	0.001	8.81	5.47	11.32	-6.94	3.33	-4.23	18.38
125905	381	2	-60.14	-76.43	-55.54	0.619	0.786	0.001	-0.44	-0.26	2.99	0.34	2.99	-3.80	10.35
125905	381	3	-60.14	-76.43	-55.54	0.619	0.786	0.001	-9.68	-6.00	-5.34	7.62	2.65	-3.37	17.23
125904	381	1	-59.74	-76.74	-57.91	0.619	0.786	0.001	10.49	6.43	10.84	-8.21	2.38	-3.04	18.79
125904	381	2	-59.74	-76.74	-57.91	0.619	0.786	0.001	-1.01	-0.62	4.09	0.79	2.40	-3.06	9.66
125904	381	3	-59.74	-76.74	-57.91	0.619	0.786	0.001	-12.51	-7.68	-2.66	9.80	2.41	-3.08	21.02
137541	381	1	-58.89	-77.39	-57.63	0.611	0.792	0.000	13.94	8.30	12.31	-10.76	1.33	-1.73	22.70
137541	381	2	-58.89	-77.39	-57.63	0.611	0.792	0.000	-0.96	-0.57	3.94	0.74	2.14	-2.77	8.89
137541	381	3	-58.89	-77.39	-57.63	0.611	0.792	0.000	-15.87	-9.45	-4.43	12.24	2.95	-3.82	26.37
140800	381	1	-58.89	-77.39	-54.91	0.611	0.791	-0.001	12.09	7.35	8.49	-9.42	2.39	-3.05	20.67
140800	381	2	-58.89	-77.39	-54.91	0.611	0.791	-0.001	-0.34	-0.19	1.72	0.26	2.76	-3.55	9.27
140800	381	3	-58.89	-77.39	-54.91	0.611	0.791	-0.001	-12.77	-7.73	-5.05	9.94	3.14	-4.05	22.04
142171	381	1	-58.89	-77.39	-52.20	0.617	0.787	-0.003	10.55	6.53	8.04	-8.30	3.61	-4.57	19.94
142171	381	2	-58.89	-77.39	-52.20	0.617	0.787	-0.003	0.12	0.08	0.57	-0.10	3.40	-4.32	11.00
142171	381	3	-58.89	-77.39	-52.20	0.617	0.787	-0.003	-10.32	-6.36	-6.90	8.10	3.19	-4.07	18.91

137542	388	1	-60.63	-79.84	-57.63	0.814	-0.581	0.000	-1.14	-2.24	-22.73	-1.60	-2.21	-1.57	23.10
137542	388	2	-60.63	-79.84	-57.63	0.814	-0.581	0.000	0.03	0.06	-21.56	0.05	-3.28	-2.34	23.11
137542	388	3	-60.63	-79.84	-57.63	0.814	-0.581	0.000	1.20	2.37	-20.39	1.69	-4.36	-3.11	26.24
140799	388	1	-60.63	-79.84	-54.91	0.814	-0.581	0.000	-1.39	-2.73	-18.83	-1.95	-2.37	-1.69	19.39
140799	388	2	-60.63	-79.84	-54.91	0.814	-0.581	0.000	0.00	0.00	-17.41	0.00	-3.44	-2.46	19.35
140799	388	3	-60.63	-79.84	-54.91	0.814	-0.581	0.000	1.39	2.72	-15.99	1.94	-4.52	-3.23	22.96
142172	388	1	-60.63	-79.84	-52.20	0.814	-0.581	0.000	-1.52	-2.99	-15.30	-2.13	-2.27	-1.62	15.98
142172	388	2	-60.63	-79.84	-52.20	0.814	-0.581	0.000	-0.02	-0.04	-13.74	-0.03	-3.38	-2.41	16.00
142172	388	3	-60.63	-79.84	-52.20	0.814	-0.581	0.000	1.48	2.91	-12.18	2.07	-4.49	-3.20	19.90
137541	388	1	-58.89	-77.39	-57.63	0.814	-0.581	0.000	7.94	15.61	11.33	11.13	-5.85	-4.17	26.87
137541	388	2	-58.89	-77.39	-57.63	0.814	-0.581	0.000	0.01	0.03	4.24	0.02	-4.78	-3.41	12.47
137541	388	3	-58.89	-77.39	-57.63	0.814	-0.581	0.000	-7.91	-15.56	-2.86	-11.10	-3.71	-2.65	24.43
140800	388	1	-58.89	-77.39	-54.91	0.814	-0.581	0.000	5.91	11.62	7.64	8.29	-5.56	-3.96	21.02
140800	388	2	-58.89	-77.39	-54.91	0.814	-0.581	0.000	-0.02	-0.03	2.04	-0.02	-4.86	-3.47	12.12
140800	388	3	-58.89	-77.39	-54.91	0.814	-0.581	0.000	-5.95	-11.69	-3.56	-8.34	-4.16	-2.97	19.30
142171	388	1	-58.89	-77.39	-52.20	0.814	-0.581	0.000	4.09	8.04	4.15	5.74	-5.34	-3.81	15.82
142171	388	2	-58.89	-77.39	-52.20	0.814	-0.581	0.000	-0.04	-0.07	0.23	-0.05	-5.05	-3.60	12.40
142171	388	3	-58.89	-77.39	-52.20	0.814	-0.581	0.000	-4.17	-8.19	-3.69	-5.84	-4.75	-3.39	15.30
126858	388	1	-58.45	-76.78	-57.63	0.814	-0.581	0.000	15.53	7.90	15.07	-11.08	4.10	-5.74	27.45
126858	388	2	-58.45	-76.78	-57.63	0.814	-0.581	0.000	0.28	0.14	10.65	-0.20	2.81	-3.94	14.08

126858	388	3	-58.45	-76.78	-57.63	0.814	-0.581	0.000	-7.86	-15.45	2.52	-11.02	-2.88	-2.05	26.78
126857	388	1	-58.45	-76.78	-54.91	0.814	-0.581	0.000	14.57	7.41	11.58	-10.39	4.41	-6.18	25.98
126857	388	2	-58.45	-76.78	-54.91	0.814	-0.581	0.000	0.76	0.39	6.88	-0.54	3.31	-4.64	12.77
126857	388	3	-58.45	-76.78	-54.91	0.814	-0.581	0.000	-7.38	-14.51	-0.49	-10.35	-3.64	-2.59	23.20
126859	388	1	-58.45	-76.78	-52.20	0.814	-0.581	0.000	13.90	7.07	8.44	-9.91	4.52	-6.33	24.69
126859	388	2	-58.45	-76.78	-52.20	0.814	-0.581	0.000	0.82	0.41	3.66	-0.58	3.81	-5.34	13.34
126859	388	3	-58.45	-76.78	-52.20	0.814	-0.581	0.000	-6.91	-13.59	-3.16	-9.69	-4.63	-3.30	22.21
126773	388	1	-56.56	-78.13	-54.30	-0.581	-0.814	0.000	15.12	7.69	14.56	-10.78	2.86	-4.01	25.10
126773	388	2	-56.56	-78.13	-54.30	-0.581	-0.814	0.000	0.57	0.29	8.86	-0.40	1.90	-2.66	10.33
126773	388	3	-56.56	-78.13	-54.30	-0.581	-0.814	0.000	-13.98	-7.11	3.15	9.97	0.94	-1.32	24.46
126776	388	1	-56.52	-78.17	-57.16	-0.581	-0.814	0.000	13.57	6.91	15.67	-9.68	3.01	-4.22	23.79
126776	388	2	-56.52	-78.17	-57.16	-0.581	-0.814	0.000	0.35	0.18	11.55	-0.25	1.34	-1.87	12.01
126776	388	3	-56.52	-78.17	-57.16	-0.581	-0.814	0.000	-12.88	-6.55	7.43	9.19	-0.34	0.47	26.89
138631	388	1	-55.75	-83.32	-57.63	-0.814	0.581	0.000	-25.95	-51.01	-8.22	-36.38	-2.04	-1.45	77.05
138631	388	2	-55.75	-83.32	-57.63	-0.814	0.581	0.000	0.41	0.80	14.66	0.57	-2.27	-1.62	15.21
138631	388	3	-55.75	-83.32	-57.63	-0.814	0.581	0.000	26.77	52.61	37.53	37.53	-2.50	-1.78	79.61
138635	388	1	-55.75	-83.32	-54.91	-0.814	0.581	0.000	-29.63	-58.23	-10.14	-41.54	-2.33	-1.66	87.97
138635	388	2	-55.75	-83.32	-54.91	-0.814	0.581	0.000	0.47	0.93	16.51	0.66	-2.01	-1.43	16.90
138635	388	3	-55.75	-83.32	-54.91	-0.814	0.581	0.000	30.57	60.09	43.16	42.86	-1.68	-1.20	90.75
138636	388	1	-55.75	-83.32	-52.20	-0.814	0.581	0.000	-33.08	-65.02	-13.47	-46.38	-2.25	-1.60	98.18

138636	388	2	-55.75	-83.32	-52.20	-0.814	0.581	0.000	0.53	1.04	16.89	0.74	-1.51	-1.08	17.12
138636	388	3	-55.75	-83.32	-52.20	-0.814	0.581	0.000	34.14	67.10	47.26	47.86	-0.78	-0.56	101.25
126791	388	1	-55.10	-79.18	-55.00	-0.581	-0.814	0.000	16.56	8.43	18.06	-11.81	1.07	-1.50	25.45
126791	388	2	-55.10	-79.18	-55.00	-0.581	-0.814	0.000	1.28	0.65	11.42	-0.91	0.60	-0.85	11.53
126791	388	3	-55.10	-79.18	-55.00	-0.581	-0.814	0.000	-14.00	-7.12	4.78	9.99	0.14	-0.20	25.91
126816	388	1	-54.66	-81.79	-57.65	-0.814	0.581	0.000	-10.56	-20.76	5.11	-14.81	-5.40	-3.85	38.78
126816	388	2	-54.66	-81.79	-57.65	-0.814	0.581	0.000	0.29	0.56	14.56	0.40	-3.07	-2.19	15.65
126816	388	3	-54.66	-81.79	-57.65	-0.814	0.581	0.000	11.14	21.89	24.00	15.62	-0.74	-0.53	33.12
126818	388	1	-54.66	-81.79	-52.20	-0.814	0.581	0.000	-11.97	-23.53	1.50	-16.79	-3.95	-2.82	38.25
126818	388	2	-54.66	-81.79	-52.20	-0.814	0.581	0.000	0.38	0.76	12.86	0.54	-1.06	-0.76	13.00
126818	388	3	-54.66	-81.79	-52.20	-0.814	0.581	0.000	12.74	25.05	24.22	17.86	1.82	1.30	38.15
126817	388	1	-54.66	-81.79	-54.92	-0.814	0.581	0.000	-11.27	-22.16	3.68	-15.80	-4.75	-3.38	38.89
126817	388	2	-54.66	-81.79	-54.92	-0.814	0.581	0.000	0.34	0.66	14.05	0.47	-2.07	-1.48	14.52
126817	388	3	-54.66	-81.79	-54.92	-0.814	0.581	0.000	11.95	23.48	24.41	16.75	0.60	0.43	35.48
		Max	-49.20	-76.43	-52.20										
		Min	-62.21	-87.34	-57.91										
		Range	13.01	10.92	5.72										

Table A-6. Access Hole Cover Plate FEM Stress Intensity Output – EPU FIV Load Case

node	icomp	BMU	x	y	z	nx	ny	nz	sxx	syy	szz	sxy	syz	szx	SI
146992	368	1	-11.00	-96.63	-37.25	0.129	0.992	0.000	-91.08	-0.66	-4.76	7.75	-1.68	15.09	94.39
146992	368	2	-11.00	-96.63	-37.25	0.129	0.992	0.000	-12.78	-0.16	15.75	1.42	-1.30	11.12	36.40
146992	368	3	-11.00	-96.63	-37.25	0.129	0.992	0.000	65.52	0.33	36.25	-4.91	-0.92	7.15	67.59
146992	378	1	-11.00	-96.63	-37.25	-0.994	0.113	0.000	-3.03	-233.86	-63.94	-26.62	-29.77	-3.39	241.94
146992	378	2	-11.00	-96.63	-37.25	-0.994	0.113	0.000	0.02	1.73	20.21	0.20	-8.25	-0.94	24.83
146992	378	3	-11.00	-96.63	-37.25	-0.994	0.113	0.000	3.08	237.32	104.37	27.02	13.27	1.51	241.69
146991	368	1	-11.00	-96.63	-33.78	0.129	0.992	0.000	-71.03	-0.49	-10.92	5.96	-2.62	23.83	80.00
146991	368	2	-11.00	-96.63	-33.78	0.129	0.992	0.000	-10.17	-0.13	16.45	1.13	-1.76	14.38	39.43
146991	368	3	-11.00	-96.63	-33.78	0.129	0.992	0.000	50.70	0.23	43.83	-3.71	-0.90	4.93	53.58
146991	378	1	-11.00	-96.63	-33.78	-0.994	0.113	0.000	-2.60	-201.00	-51.08	-22.88	-39.05	-4.45	213.13
146991	378	2	-11.00	-96.63	-33.78	-0.994	0.113	0.000	0.03	2.36	20.38	0.27	-15.46	-1.76	35.94
146991	378	3	-11.00	-96.63	-33.78	-0.994	0.113	0.000	2.67	205.72	91.84	23.42	8.13	0.93	208.96
146990	368	1	-11.00	-96.63	-30.31	0.129	0.992	0.000	-38.98	-0.26	9.01	3.28	-3.83	32.61	81.51
146990	368	2	-11.00	-96.63	-30.31	0.129	0.992	0.000	1.14	0.01	15.74	-0.11	-2.38	19.38	41.68
146990	368	3	-11.00	-96.63	-30.31	0.129	0.992	0.000	41.26	0.28	22.46	-3.50	-0.93	6.14	43.42
146990	378	1	-11.00	-96.63	-30.31	-0.994	0.113	0.000	-2.01	-155.24	-34.36	-17.67	-44.65	-5.08	171.93
146990	378	2	-11.00	-96.63	-30.31	-0.994	0.113	0.000	0.03	2.34	18.10	0.27	-21.28	-2.42	45.63
146990	378	3	-11.00	-96.63	-30.31	-0.994	0.113	0.000	2.07	159.91	70.57	18.20	2.09	0.24	162.04
146985	368	1	-11.00	-96.63	-26.56	0.129	0.992	0.000	45.81	0.70	24.42	-5.65	-4.06	33.27	70.75
146985	368	2	-11.00	-96.63	-26.56	0.129	0.992	0.000	18.53	0.24	15.45	-2.11	-1.86	14.48	31.81
146985	368	3	-11.00	-96.63	-26.56	0.129	0.992	0.000	-8.75	-0.22	6.47	1.43	0.33	-4.31	17.71
146985	378	1	-11.00	-96.63	-26.56	-0.994	0.113	0.000	-1.02	-78.86	-16.81	-8.98	-48.45	-5.52	116.15
146985	378	2	-11.00	-96.63	-26.56	-0.994	0.113	0.000	0.00	-0.01	12.38	0.00	-28.99	-3.30	59.66
146985	378	3	-11.00	-96.63	-26.56	-0.994	0.113	0.000	1.02	78.84	41.56	8.98	-9.53	-1.08	82.13
146985	378	1	-11.00	-96.63	-26.56	-0.994	0.113	0.000	-1.02	-78.86	-16.81	-8.98	-48.45	-5.52	116.15

146985	378	2	-11.00	-96.63	-26.56	-0.994	0.113	0.000	0.00	-0.01	12.38	0.00	-28.99	-3.30	59.66
146985	378	3	-11.00	-96.63	-26.56	-0.994	0.113	0.000	1.02	78.84	41.56	8.98	-9.53	-1.08	82.13
124858	368	1	-9.58	-96.78	-26.52	0.092	0.996	-0.001	36.94	0.35	-3.17	-3.59	-3.47	35.70	82.37
124858	368	2	-9.58	-96.78	-26.52	0.092	0.996	-0.001	25.83	0.25	14.81	-2.53	-0.16	1.67	26.32
124858	368	3	-9.58	-96.78	-26.52	0.092	0.996	-0.001	14.73	0.15	32.79	-1.47	3.15	-32.36	67.46
124859	368	1	-8.76	-96.85	-29.27	0.092	0.996	-0.001	13.24	0.11	-7.26	-1.19	-3.98	45.18	93.03
124859	368	2	-8.76	-96.85	-29.27	0.092	0.996	-0.001	7.89	0.06	14.27	-0.70	-0.79	8.83	20.52
124859	368	3	-8.76	-96.85	-29.27	0.092	0.996	-0.001	2.55	0.02	35.79	-0.21	2.41	-27.53	64.48
124815	368	1	-8.37	-96.89	-27.24	0.092	0.996	-0.001	28.52	0.22	17.01	-2.52	-4.61	53.74	108.50
124815	368	2	-8.37	-96.89	-27.24	0.092	0.996	-0.001	26.80	0.20	15.64	-2.30	-0.44	5.01	28.91
124815	368	3	-8.37	-96.89	-27.24	0.092	0.996	-0.001	25.07	0.17	14.27	-2.08	3.73	-43.71	88.41
124919	368	1	-7.59	-96.95	-27.21	0.080	0.997	0.002	36.33	0.23	-3.93	-2.90	-3.19	42.06	93.58
124919	368	2	-7.59	-96.95	-27.21	0.080	0.997	0.002	24.79	0.15	13.23	-1.93	-0.51	6.41	27.78
124919	368	3	-7.59	-96.95	-27.21	0.080	0.997	0.002	13.25	0.07	30.39	-0.95	2.18	-29.24	61.08
124857	368	1	-6.98	-97.00	-28.10	0.080	0.997	0.002	53.32	0.30	31.67	-3.99	-2.61	37.38	81.69
124857	368	2	-6.98	-97.00	-28.10	0.080	0.997	0.002	15.28	0.08	15.04	-1.10	-0.47	6.34	21.56
124857	368	3	-6.98	-97.00	-28.10	0.080	0.997	0.002	-22.76	-0.14	-1.59	1.80	1.67	-24.70	53.91
124915	368	1	-6.38	-97.04	-26.33	0.072	0.997	0.000	-1.49	0.00	-21.62	0.08	-2.63	39.41	81.52
124915	368	2	-6.38	-97.04	-26.33	0.072	0.997	0.000	44.17	0.20	8.10	-2.94	-0.10	1.53	44.43
124915	368	3	-6.38	-97.04	-26.33	0.072	0.997	0.000	89.83	0.40	37.83	-5.95	2.42	-36.35	108.89
124860	368	1	-5.93	-97.07	-33.27	0.035	0.999	0.007	102.02	0.44	70.45	-6.08	-1.33	14.74	108.11
124860	368	2	-5.93	-97.07	-33.27	0.035	0.999	0.007	-6.39	-0.03	1.82	0.44	-0.26	3.24	10.49
124860	368	3	-5.93	-97.07	-33.27	0.035	0.999	0.007	-114.80	-0.51	-66.81	6.97	0.80	-8.27	116.52
124861	368	1	-5.60	-97.09	-37.19	0.031	1.000	-0.001	135.27	0.48	56.66	-7.19	-0.72	12.42	137.46
124861	368	2	-5.60	-97.09	-37.19	0.031	1.000	-0.001	-10.67	-0.04	-3.84	0.62	-0.32	4.93	13.28
124861	368	3	-5.60	-97.09	-37.19	0.031	1.000	-0.001	-156.62	-0.57	-64.34	8.44	0.08	-2.57	157.02
124913	368	1	-5.27	-97.11	-26.64	0.042	0.999	0.002	11.59	0.03	-6.49	-0.59	-0.87	16.28	37.29

124913	368	2	-5.27	-97.11	-26.64	0.042	0.999	0.002	40.64	0.12	6.55	-2.20	-0.20	3.42	41.10
124913	368	3	-5.27	-97.11	-26.64	0.042	0.999	0.002	69.69	0.21	19.59	-3.81	0.48	-9.43	71.61
124832	368	1	-5.16	-97.11	-29.28	0.035	0.999	0.007	81.60	0.25	62.36	-4.32	-1.06	17.55	92.17
124832	368	2	-5.16	-97.11	-29.28	0.035	0.999	0.007	6.92	0.03	6.35	-0.41	-0.07	0.87	7.57
124832	368	3	-5.16	-97.11	-29.28	0.035	0.999	0.007	-67.75	-0.20	-49.66	3.51	0.92	-15.80	77.06
124911	368	1	-3.99	-97.17	-26.51	0.025	1.000	0.000	47.78	0.08	-13.94	-1.88	0.25	-7.10	63.41
124911	368	2	-3.99	-97.17	-26.51	0.025	1.000	0.000	39.10	0.06	4.58	-1.53	-0.22	5.28	39.95
124911	368	3	-3.99	-97.17	-26.51	0.025	1.000	0.000	30.42	0.05	23.10	-1.19	-0.68	17.66	44.83
124856	368	1	-2.54	-97.22	-29.27	0.035	0.999	0.007	101.84	0.08	67.59	-2.32	0.43	-18.72	110.11
124856	368	2	-2.54	-97.22	-29.27	0.035	0.999	0.007	6.42	0.01	1.73	-0.17	0.02	-0.87	6.58
124856	368	3	-2.54	-97.22	-29.27	0.035	0.999	0.007	-89.01	-0.07	-64.12	1.99	-0.39	16.99	97.64
124855	368	1	-0.90	-97.25	-28.19	0.002	1.000	0.000	106.58	0.02	56.19	-0.76	0.43	-41.21	129.68
124855	368	2	-0.90	-97.25	-28.19	0.002	1.000	0.000	13.18	0.00	6.30	-0.12	0.04	-4.20	15.17
124855	368	3	-0.90	-97.25	-28.19	0.002	1.000	0.000	-80.23	-0.02	-43.60	0.52	-0.35	32.81	99.48
		Max	-0.90	-96.63	-26.33										
		Min	-11.00	-97.25	-37.25										
		Range	10.10	0.62	10.92										

ATTACHMENT 1:

MONTICELLO NUCLEAR GENERATING STATION, RFO-22 IN-VESSEL VISUAL INSPECTION RELEVANT INDICATION NOTIFICATION FORM, INF# MNGP-2005-01

AREVA

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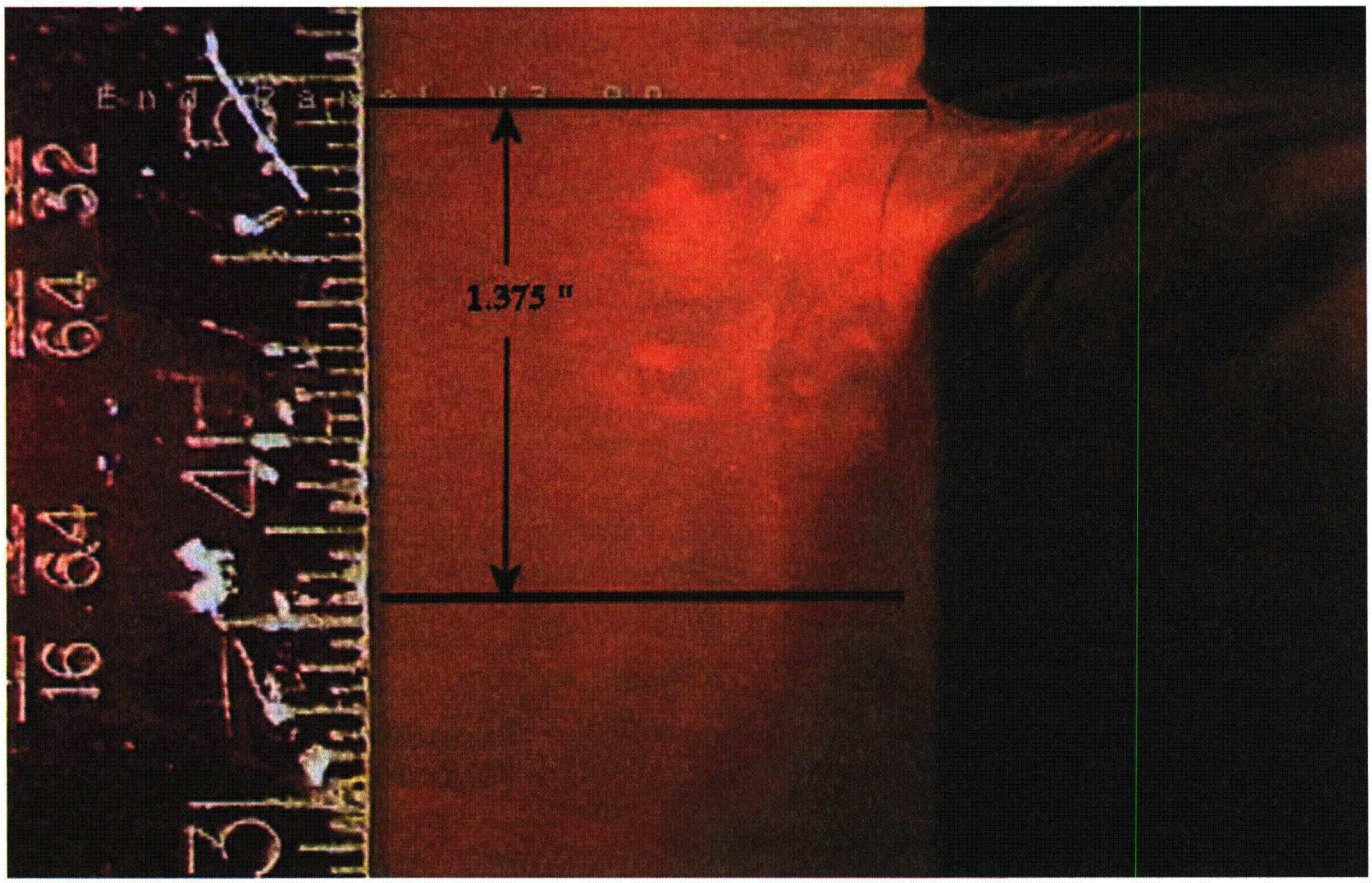
MONTICELLO NUCLEAR GENERATINGSTATION, RFO-22
IN-VESSEL VISUAL INSPECTION
RELEVANT INDICATION NOTIFICATION FORM

INF # MNGP-2005-01
Date: _3/09/05_
Time: : ____0530__
Disk Number: ____2_____
Title Number: _____3_____

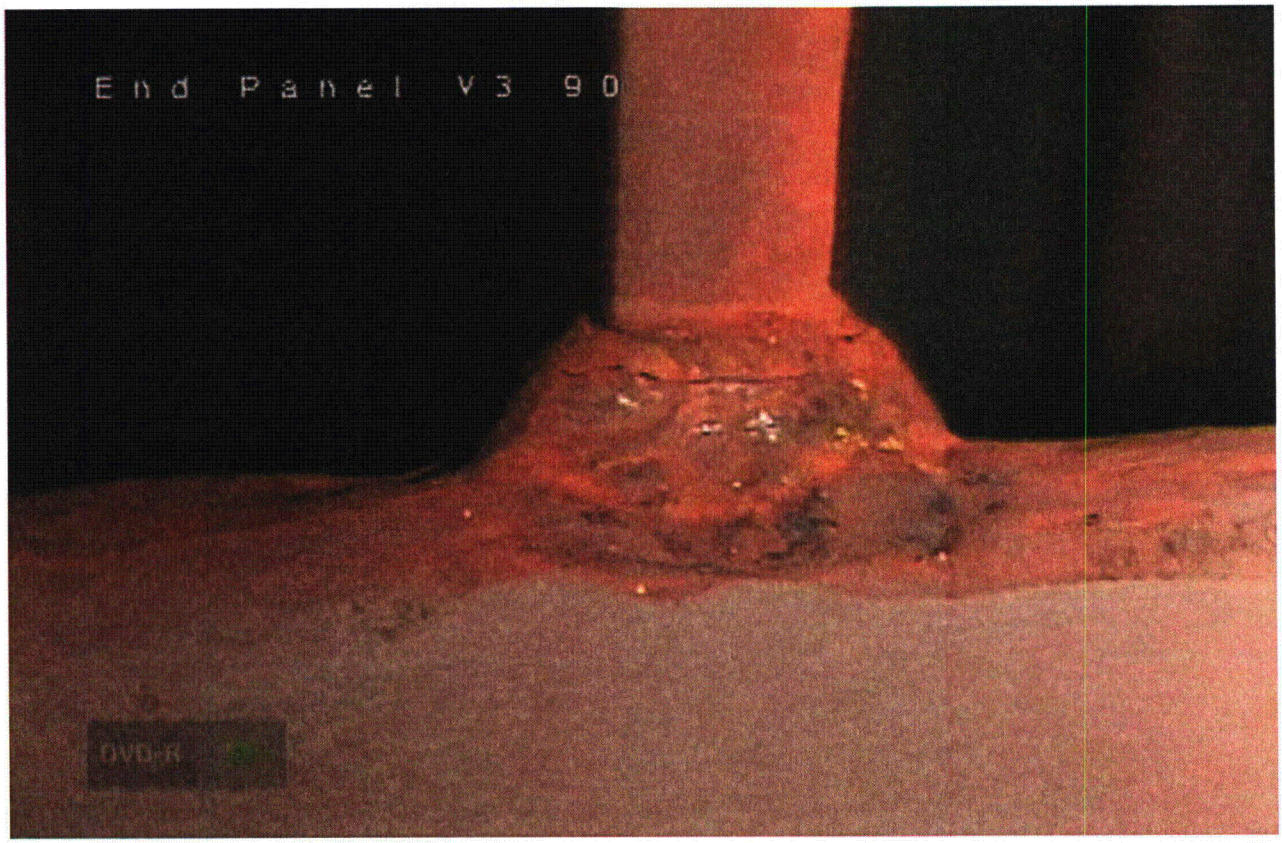
Component: ___Steam Dryer Weld V3 90_

Description of Relevant Indication:

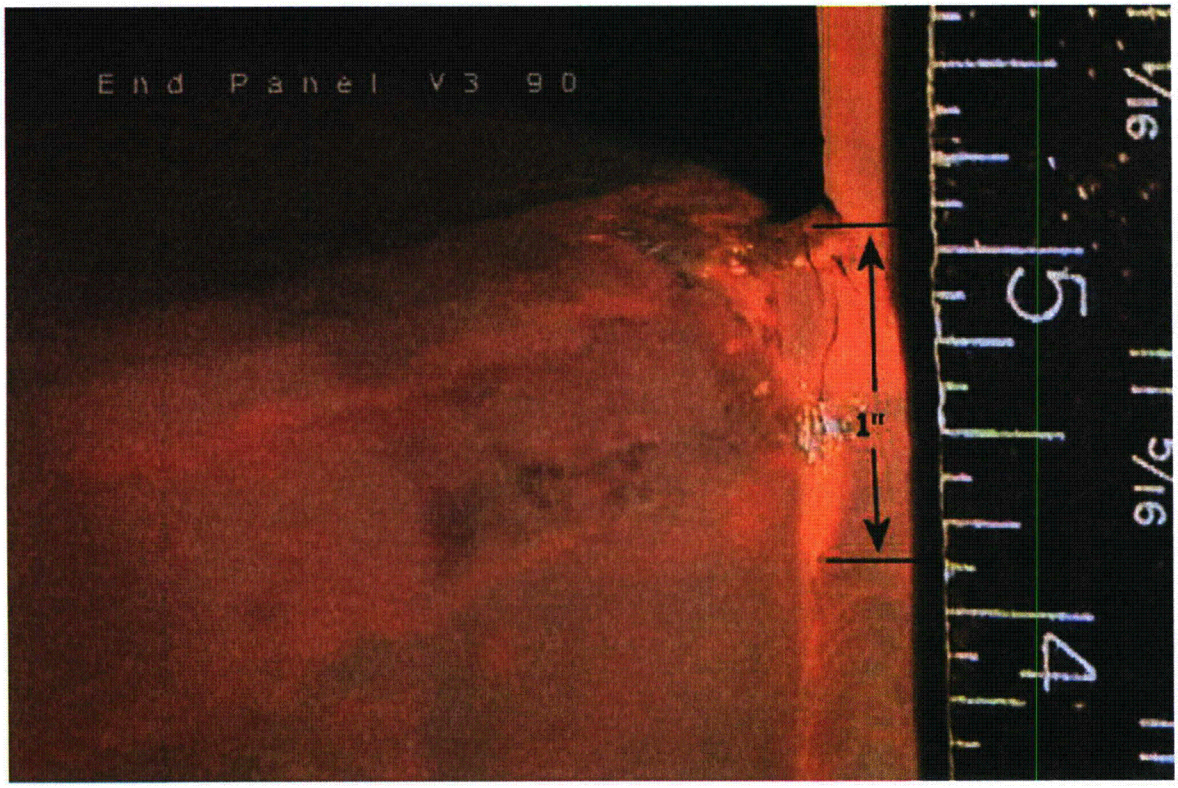
A crack was located at the top of Weld V3 90. This crack extends for approximately 1.375" on the outside of the End Panel weld across the top and down the inside of the weld to Dryer Bank "B" for approximately 1".



Outside View



Top View



Inside Measurement

FANP VT Level III: _____

MNGP Review: _____

(Return Copy to FANP for Records)

ATTACHMENT 2:

MONTICELLO NUCLEAR GENERATING STATION, RFO-22 IN-VESSEL VISUAL INSPECTION RELEVANT INDICATION NOTIFICATION FORM, INF# MNGP-2005-02

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IN-VESSEL VISUAL INSPECTION
RELEVANT INDICATION NOTIFICATION FORM

INF # MNGP-2005-02
Date: ____3/09/05____
Time: : _____0800_____
Disk Number: _____4_____
Title Number: _____8_____

Component: ____Dryer____

Description of Relevant Indication:
A crack was located at the top of Weld V10 90 of the Steam Dryer. The crack is from the Weld center moving to the right toe and wraps over the top. Approximately 1 3/8 inches in length. See attached pictures.

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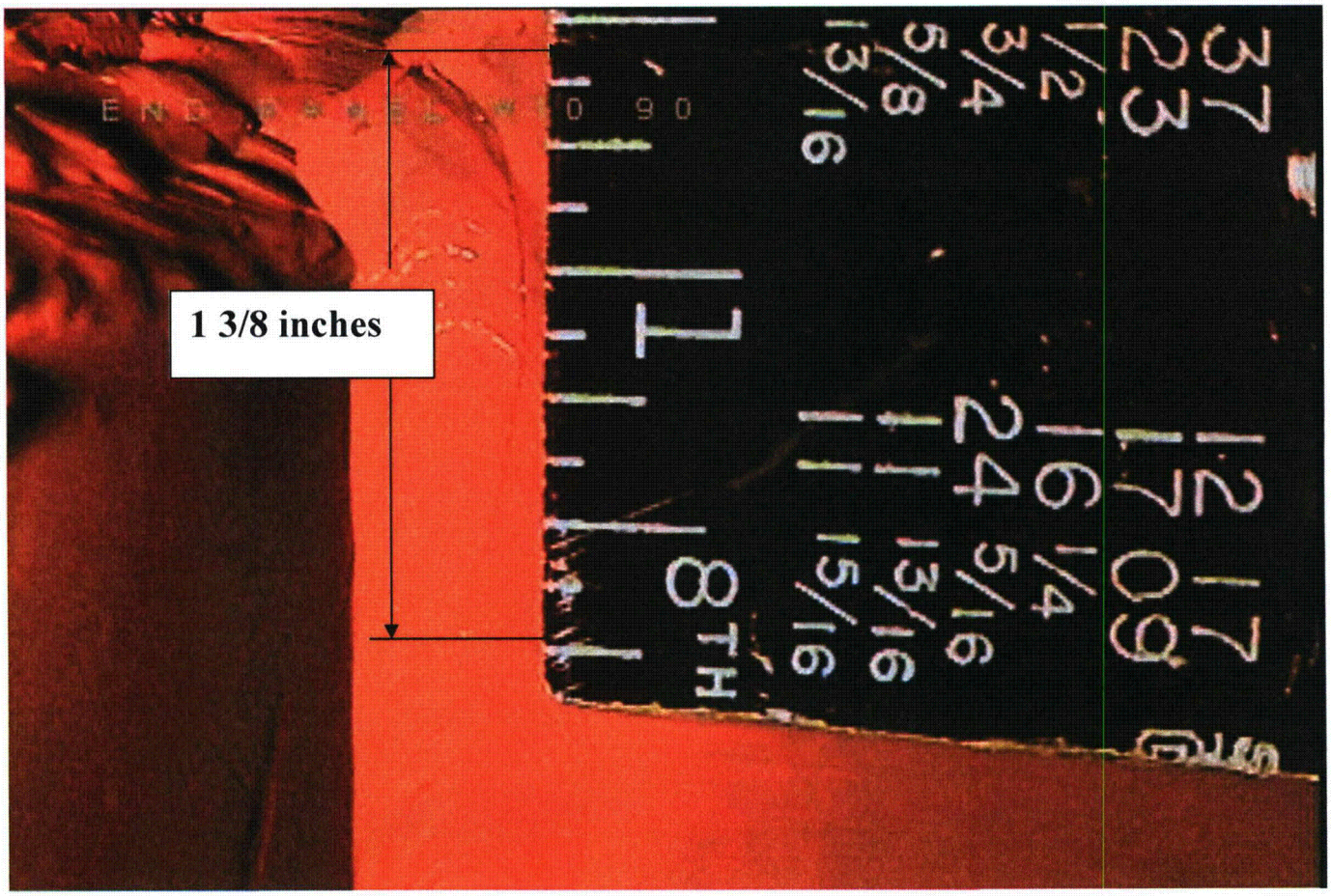
Top View

Measurement



FANP VT Level III: _____

MNGP Review: _____

(Return Copy to FANP for Records)

ATTACHMENT 3:

MONTICELLO NUCLEAR GENERATING STATION, RFO-23 IN-VESSEL VISUAL INSPECTION RELEVANT INDICATION NOTIFICATION FORM, INR-023-01

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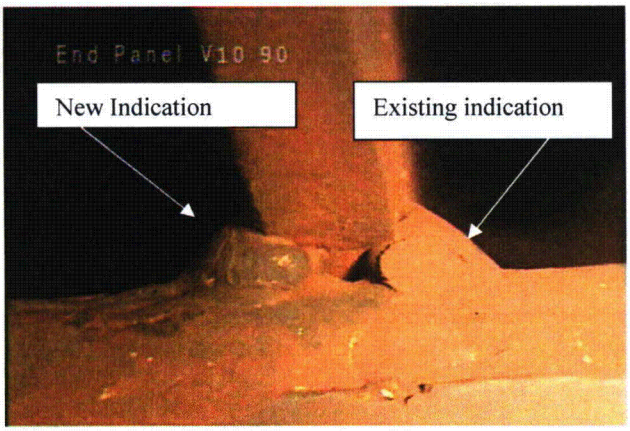
MONTICELLO NUCLEAR GENERATINGSTATION, RFO-23
IN-VESSEL VISUAL INSPECTION
RELEVANT INDICATION NOTIFICATION FORM

INR-R23-01

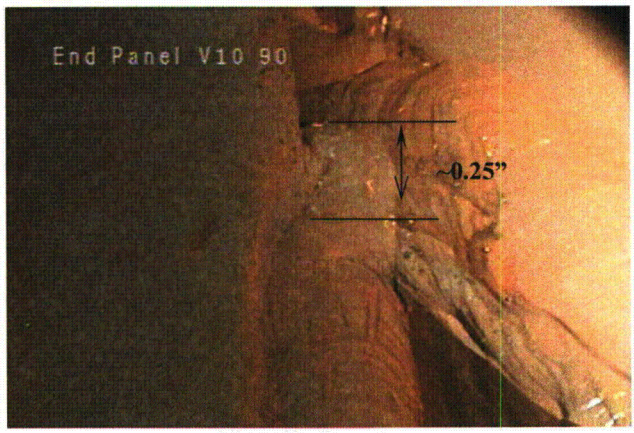
Date: 3/20/07
Time: : 06:00
Disk Number: 01
Title Number: 03

Component: **STEAM DRYER**
Description: **End Panel weld V10 90**

Description of Relevant Indication:
While searching for the existing indication on V10 90, an additional indication approximately 0.25 in. long was seen on the opposite (OD) side of the plate. It should be noted that a review of 2005 inspection data revealed no evidence of this area being examined.



Top View of weld



New Indication

AREVA VT Level III: _____

MNGP Review: _____

(Return Copy to AREVA for Records)

ATTACHMENT 4:

MONTICELLO NUCLEAR GENERATING STATION, RFO-22 IN-VESSEL VISUAL INSPECTION RELEVANT INDICATION NOTIFICATION FORM, INF# MNGP-2005-05

FRAMATOME ANP

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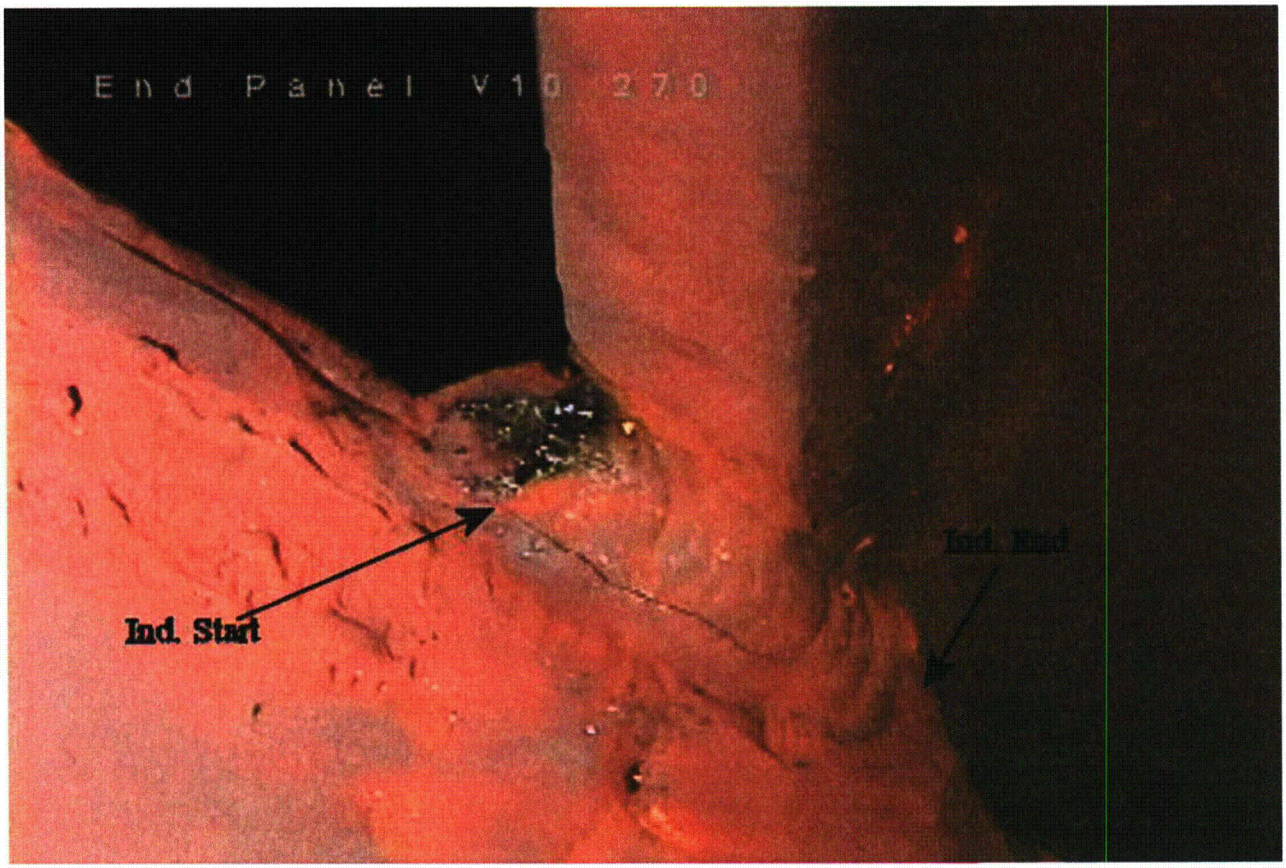
MONTICELLO NUCLEAR GENERATINGSTATION, RFO-22
IN-VESSEL VISUAL INSPECTION
RELEVANT INDICATION NOTIFICATION FORM

INF # MNGP-2005-05
Date: __3/9/05___
Time: : ____2131_____
Disk Number: __7_____
Title Number: ___5_____

Component: _Steam Dryer Weld V10 270

Description of Relevant Indication:
A crack was located at the top of Weld V10 270 of the Steam Dryer. The crack is located over the over the top at the junction of the End Panel and Dryer Bank. The indication is less than 1 inch in length. See attached pictures.



FANP VT Level III: _____

MNGP Review: _____

(Return Copy to FANP for Records)

ATTACHMENT 5:

MONTICELLO NUCLEAR GENERATING STATION, RFO-22 IN-VESSEL VISUAL INSPECTION RELEVANT INDICATION NOTIFICATION FORM, INF# MNGP-2005-03

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MONTICELLO NUCLEAR GENERATINGSTATION, RFO-22
IN-VESSEL VISUAL INSPECTION
RELEVANT INDICATION NOTIFICATION FORM

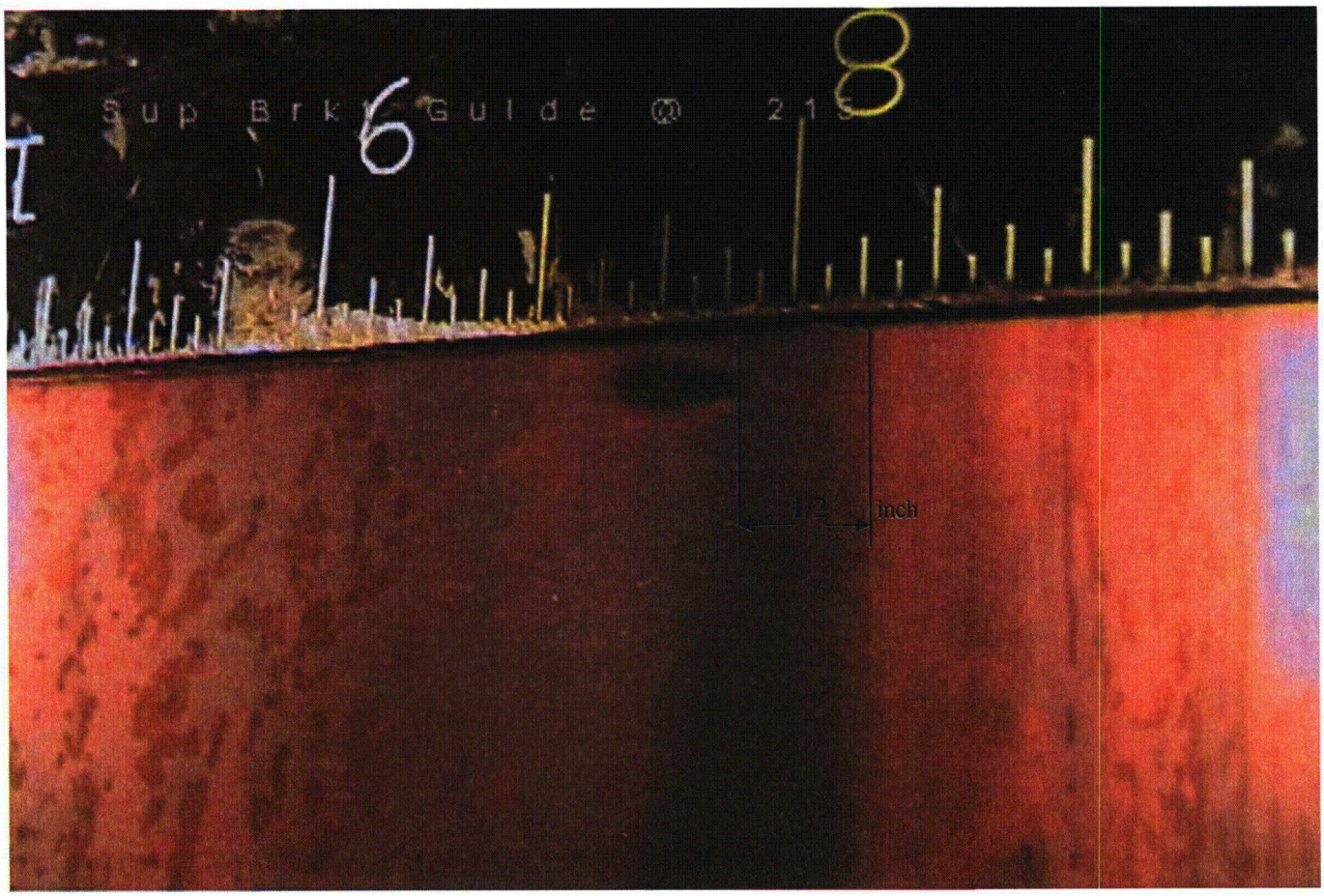
INF # MNGP-2005-03
Date: ____3/09/05____
Time: _____1200_____
Disk Number: _____4_____
Title Number: _____8_____

Component: ____Dryer____

Description of Relevant Indication:
A crack was located about 4 feet from the bottom of 215 degree Dryer support Bracket Guide Channel. The crack comes horizontally across from a possible arc strike around the corner of the channel and into the left toe of vertical weld on the face of the dryer. This is on the right side of the Channel Guide and the Length is approximately ¾ inch. See Attached Pictures.

Full Corner View

Measurement from Left side



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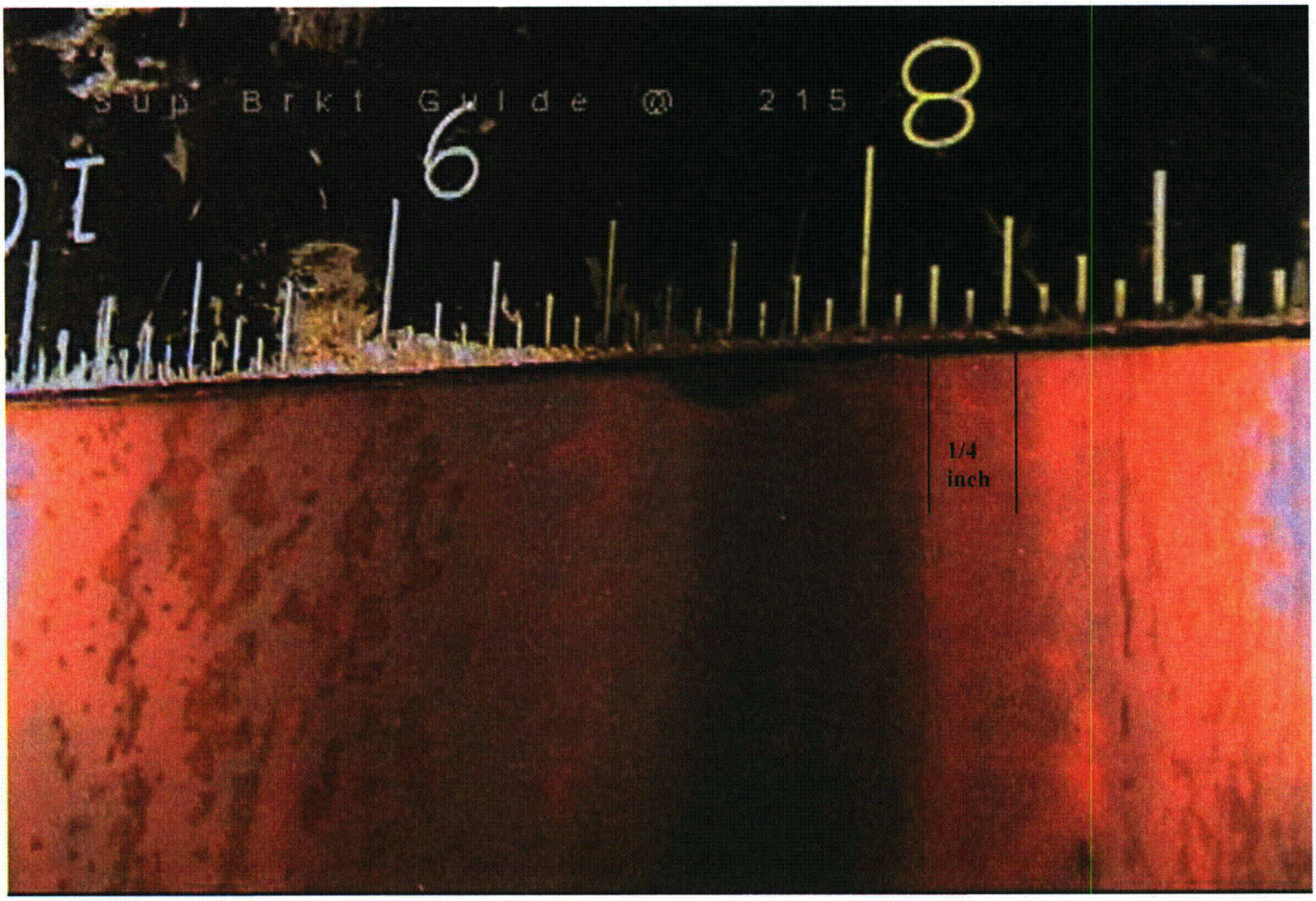
Measurement from the Right side



FANP VT Level III: _____

MNGP Review: _____

(Return Copy to FANP for Records)

Enclosure 17 to L-MT-08-052

NSPM Revised Response to NRC Electrical Engineering Branch (EEEB) Review Item Documented in L-MT-08-42

ENCLOSURE 17

Revised Response to NRC EEEB Review Question Documented in L-MT-08-042

NMC (now NSPM) provided a response in L-MT-08-042 (ML 081570467) dated June 5, 2008 to an NRC acceptance review question associated with the original EPU LAR submittal. The response below supersedes the response provided in L-MT-08-042.

In the following paragraphs, the original EEEB question as documented in L-MT-08-042 is followed by NSPM's revised response which is based on more accurate information.

NRC Item:

In Section 2.3 of the LAR under the section titled 'Outside Containment', the licensee stated the following:

"The total integrated doses (normal plus accident) for EPU conditions were evaluated and determined not to adversely affect qualification of most of the EQ equipment located outside of containment. Equipment not qualified to the new environmental conditions at EPU will be reanalyzed, re-qualified, or replaced prior to implementation of EPU."

In order for the Electrical Engineering Branch (EEEB) to start its review, the full EQ analysis must be completed. This includes any reanalysis, re-qualification, or replacement of equipment. The licensee must also describe how the equipment was evaluated (e.g., calculations, assessments, etc.) and show how the equipment remains bounded (i.e., provide the original design parameters and the updated values including the supporting calculations).

NSPM Response:

Since this question was originally posed, the scope of the response has been developed more fully. The NRC staff and NSPM have mutually determined that the standard PUSAR section that addresses the changes to the Environmental Qualification (EQ) program, of itself, is not sufficient for a complete review of the EPU effects without a significant amount of supplemental information. NSPM agrees that EPU changes to the EQ program are unique with respect to scope and interactions. The extent of the topic is quite large such that a summarized submittal of the EQ effects may be perceived as a fragmented response. In view of this perception combined with an effort to create a single, concise and comprehensive EQ document, NSPM prepared a revision to EQ task report T 1004, Revision 1 which is attached to the end of this response. Revision 1 has been significantly expanded to remove ambiguity and increase the level of detail. Further, NSPM is submitting the entire task report instead submitting it in summary form which has an effect of diluting the information.

ENCLOSURE 17

Revised Response to NRC EEEB Review Question Documented in L-MT-08-042

It should be noted that the report evaluates each and every environmental EPU effect against the equipment in the EQ program. In addition, the design values that support the conclusions are systematically incorporated into the evaluations such that the EPU effects on equipment margins are apparent.

Supporting calculations have been referenced throughout the task report. The report was written to present the calculation results without including the associated computations. Associated computations are available, as necessary, for review at the plant site; or can be made available, if requested.

Task Report T1004
Revision 1
October 2008

Project Task Report

Monticello Nuclear Generating Plant
Extended Power Uprate

Task T1004: Environmental Qualification

Rev.	Prepared By	Date	Reviewed By	Approved By	Date
0	C Nelson	3/24/08	See PassPort AR 01078173	Steve Hammer	3/24/08
1	Regis C Eibel	10-27-2008	See PassPort AR 01078173	S Hammer	10/27/08

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REVISION SUMMARY

No.	Change
0	Original
1	Incorporate 60 year normal dose from calculation CA-08-067 for EPU conditions. This calculation formally provides the normal doses for all plant areas using conservative survey data either from that previously used in CA-04-034 or recently obtained as part of CA-08-067. The latter calculation also corrects the Drywell normal dose for re-rated power and hydrogen chemistry conditions and includes all the EPU normal plant shut-down dose scaling factors (due to moisture carry-over) for the affected plant areas. In most cases for the Reactor Building, the 60 year normal total dose is lower than previously predicted due to the excess margin used in CA-04-034. However, for the Drywell, the 60 year normal dose from CA-08-067 is higher for EPU conditions than previously assessed under revision 0 of this task report. This revision also corrects the seven errors as listed in AR 01142134 made in Attachment 1 of the former revision to this task report related to accident Beta dose for EQ calculations files 98-025, 98-050, 98-055, 98-064, 98-065, and 98-071. Revised accident doses for equipment in the RHR rooms based on Source C, de-pressurized reactor water dose sources. Included corrected PLHU values from new internal plant calculations and assessed equipment completely without need for previous Key Assumption 3.2.1. Corrected HELB qualification data from DOR Limitorque actuators (CA-98-025) outside the Drywell. Strengthened the assumption made about Drywell submergence level with reference to an EPU specific ECCS pump flow calculation. Drywell submergence level clearly should remain bounded by CLTP level (see Key Assumption 3.2.2). Deleted Appendices 1 through 8 and incorporated data into new Section 3.4, Supporting Evaluation. This moved former Section 3.4 for Recommendations and Observations to Section 4.0.

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Attachments

Attachment A Series

(EQ file excerpts documenting the additional PLHU review for EPU power)

A1	CA-98-003, Allen Bradley Terminal Blocks	A1-1
A2	CA-98-006, ASCO Pressure Switches	A2-1
A3	CA-98-011, Barton Pressure Switches	A3-1
A4	CA-98-017, GE Type SI-58081 Control Cable	A4-1
A5	CA-98-021, GE Terminal Blocks (DOR)	A5-1

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(Continued)

Attachments (continued)

A6 CA-98-023, Hevi-Duty Transformer A6-1
A7 CA-98-024, GE Fan Motors A7-1
A8 CA-98-032, Namco EA180 Limit Switches without an EC210 ..... A8-1
A9 CA-98-040, Rosemount Model 1153 Series B Transmitters A9-1
A10 CA-98-041, Rosemount Conduit Seal A10-1
A11 CA-98-046, Yarway Level Switch A11-1
A12 CA-98-049, Valcor Solenoid Valves A12-1
A13 CA-98-052, Tavis Flow Transmitter A13-1
A14 CA-98-053, ITT Grinnel/Conoflow Transducer A14-1
A15 CA-98-062, Gould Contactor/Disconnect A15-1
A16 CA-98-077, Eaton/Cutler-Hammer Control Relay A16-1
A17 CA-98-079, ITT Royal PVC Cable A17-1
A18 CA-98-080, Okonite Control Cable A18-1
A19 CA-98-101, GE Terminal Blocks (50.49) A19-1
A20 CA-98-128, UCI Electrical Tape Terminations A20-1
A21 CA-03-105, Scotch Electrical Tape Splices A21-1
A22 CA-98-007, ASCO Temperature Switches A22-1
A23 CA-98-030, Microswitch Limit Switches A23-1

Attachment B Series

(Reactor Building HELB temperature profile comparisons where EPU liquid HELB exceeds the CLTP analysis of HELB conditions and where EQ components are located)

B1 Reactor Building Volume 13 ..... B1-1
B2 Reactor Building Volume 14 ..... B2-1
B3 Reactor Building Volume 18 ..... B3-1
B4 Reactor Building Volume 19 ..... B4-1
B5 Reactor Building Volume 20 ..... B5-1
B6 Reactor Building Volume 22 ..... B6-1
B7 Reactor Building Volume 27 ..... B7-1
B8 Reactor Building Volume 31 ..... B8-1
B9 Reactor Building Volume 32 ..... B9-1
B10 Reactor Building Volume 33 ..... B10-1
B11 Reactor Building Volume 34 ..... B11-1

ACRONYMS AND ABBREVIATIONS

Item	Short Form	Description
1	CLTP	Current Licensed Thermal Power
2	CLTR	Constant Pressure Power Uprate Licensing Topical Report
3	CS	Core Spray
4	DBA	Design Basis Accident
5	DOR	NRC Division of Operating Reactors
6	ECCS	Emergency Core Cooling System
7	EPU	Extended Power Uprate
8	EQ	Environmental Qualification
9	EQ	Environmentally Qualified
10	HVAC	Heating, Ventilation and Air Conditioning System
11	IEEE	Institute of Electrical and Electronics Engineers
12	LOCA	Loss of Coolant Accident
13	MNGP	Monticello Nuclear Generating Plant
14	NRC	Nuclear Regulatory Commission
15	OOS	Equipment Out of Service
16	OOS	Out of Service
17	PLHU	Post-LOCA Heatup
18	RHR	Residual Heat Removal System
19	RWCU	Reactor Water Cleanup System
20	TID	Total Integrated Dose

1.0 SCOPE AND SUMMARY

1.1 Project Summary

Item	Parameter	Scope
1	Plant	Monticello Nuclear Generating Plant
2	Project	Extended Power Uprate (EPU)
3	Project Scope	Task T1500
4	Reactor Thermal Power Levels and Pressure	• Original Licensed Thermal Power (OLTP) of 1670 MWt • Current Licensed Thermal Power (CLTP) of 1775 MWt • Target Power Uprate (TPU) level of 2044 MWt • Licensed Power Uprate (LPU) level of 2004 MWt • No change in maximum normal operating reactor dome pressure of 1025 psia.

1.2 Task Scope

Item	Parameter	Scope
1	Task Number	1004
2	Task Title	Environmental Qualification (EQ)
3	Task Evaluations	• Normal Temperature • Radiation • Drywell Accident Response • Reactor Building Accident Response • The safety-related electrical equipment was reviewed consistent with the requirements discussed in section 1.4 to determine if the existing qualifications for the normal and accident conditions expected in the area where the devices are located remain adequate. The 10 CFR 50.49 acceptance criteria including pressure, temperature, and radiation were used in making this determination.

1.3 Results Summary

Item	Result	Summary
1	Key Evaluation Results	Key results within safety and design limits: • All EQ equipment is qualified to the environments postulated to exist under EPU operation, with the exception of two level transmitters located in the Torus compartment (LT-7338A/B). This equipment will be replaced prior to the implementation of EPU. All other EQ files documenting the environmental qualification of EQ equipment at MNGP will be revised prior to implementation of EPU.
		Key results outside design limits: • None
		Other key evaluation results: For the EQ equipment at MNGP: • The post-accident temperatures inside the Drywell increase slightly in the short and long terms but are generally bounded by the CLTP Drywell profile • The post-LOCA temperatures under EPU increase (<22°F) in the Reactor Building in the short-term, but a generally bounded by CLTP conditions in the long-term. • Normal temperatures outside containment are not significantly changed due to EPU • The radiation levels under normal plant conditions were evaluated to increase at EPU conditions • Post Accident radiation levels increase. • Shutdown radiation levels increase at EPU conditions • Post-Accident Pressures are bounded by CLTP values for the Drywell and all but one volumes of the Reactor Building (RB Volume 32). All equipment remains qualified for the postulated accident pressure conditions under EPU.
2	Impact on Other Tasks	• None
3	Direct Impact on Plant Configuration	• None
4	Impact on Design Operating Margins	• None
5	Implementation Recommendations	Design Limits • None Other Recommendations: See Section 4.1.

Item	Result	Summary
6	Limitation of Performance Improvement and OOS Options	• None

1.3.1 Plant Specific Applicability to CLTR Generic Disposition

Item	CLTR Generic Scope	Applicability Assessment Parameter	Justification	Met
1	None	N/A	N/A	N/A

1.4 Design and Licensing Bases

This task evaluates the qualification of equipment in the MNGP Environmental Qualification Program at EPU conditions as required by 10 CFR 50.49 using the guidance of RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants."

The MNGP EQ Program was developed to the guidance and requirements contained in the Division of Operating Reactors (DOR) Guidelines [115][1], and Category II of NUREG 0588 [119] for equipment that predates the issuance of 10CFR50.49 [128] as delineated in 10CFR50.49 paragraph (k). For other equipment in the EQ program and for general guidance, NRC Regulatory Guide 1.89 [129] contains methods for complying with regulatory requirements of 10CFR50.49.

Mechanical Equipment with Non-Metallic components and mechanical component design qualification is not applicable. The Monticello design and licensing bases do not require a formal mechanical EQ program like the EQ program applied to electrical equipment. The design control program ensures that mechanical components are specified and procured for the environment in which they are intended to function. Periodic maintenance and testing are performed in accordance with industry operating experience and vendor recommendations to ensure continued functionality. Causes of failures are investigated as part of MNGP Maintenance Rule programs and incorporated into equipment reliability improvement efforts. Aging Management, Long Term Planning and Life Cycle Management strategies deal with equipment aging and obsolescence. As such, mechanical equipment qualification is not assessed in this task report.

A more detailed description of the design basis for the MNGP EQ Program is located in DBD T.04 [130].

[1] Numbers in brackets "[ ]" indicate reference index number as listed in Section 2 of this report.

2.0 REFERENCES

EQ Files (CLTP versions)

1. CA-98-003, Revision 4, Allen Bradley Terminal Boards
2. CA-98-004, Revision 10, Environmental Qualification (50.49) of ASCO Solenoid Valves (Normally Energized)
3. CA-98-005, Revision 10, Environmental Qualification (50.49) of ASCO Solenoid Valves (Normally De-energized)
4. CA-98-006, Revision 9, Environmental Qualification (50.49) of ASCO Pressure Switches
5. CA-98-007, Revision 9, Environmental Qualification (50.49) of ASCO Temperature Switches
6. CA-98-008, Revision 5, Environmental Qualification (50.49) of Automatic Valve Company (AVCO) Air Control Assembly
7. CA-98-010, Revision 6, Environmental Qualification (DOR) of Barksdale Pressure Switch
8. CA-98-011, Revision 8, Environmental Qualification (DOR) of Barton Pressure Switches Model 278, 288, 288A, 289, 289A
9. CA-98-012, Revision 5, Environmental Qualification (50.49) of Barton Pressure Switch Model 580A-0
10. CA-98-014, Revision 7/7A, Environmental Qualification (DOR) of E.F. Johnson Banana Plug
11. CA-98-017, Revision 6, Environmental Qualification (DOR) of General Electric Cables
12. CA-98-018, Revision 6, Environmental Qualification of General Electric Pump Motors
13. CA-98-020, Revision 6, Environmental Qualification (DOR) of General Electric Containment Penetrations
14. CA-98-021, Revision 3, Environmental Qualification (DOR) of General Electric Terminal Blocks
15. CA-98-022, Revision 2/2A, Environmental Qualification (DOR) of General Electric Motor Control Center
16. CA-98-023, Revision 7, Environmental Qualification (DOR) of Hevi-Duty Electric Transformer
17. CA-98-024, Revision 1, Environmental Qualification (DOR) of General Electric Fan Motors
18. CA-98-025, Revision 7, Environmental Qualification (DOR) of Limitorque Motor Operators
19. CA-98-026, Revision 9/9A, Environmental Qualification (50.49) of Limitorque Motor Operators

20. CA-98-027, Revision 3, Environmental Qualification (50.49) of Magnetrol Level Switches
21. CA-98-028, Revision 3, Environmental Qualification (DOR) of McDonnell & Miller Flow Switch
22. CA-98-030, Revision 4, MicroSwitch Limit Switches
23. CA-98-032, Revision 11, Environmental Qualification (50.49) of Namco EA740/EA180 Limit Switches
24. CA-98-033, Revision 3, Environmental Qualification (50.49) of Namco EC210 Quick Disconnects
25. CA-98-035, Revision 4, Environmental Qualification (50.49) of Raychem NEIS Environmental Seals
26. CA-98-036, Revision 8, Environmental Qualification (50.49) of Raychem Low Voltage Splices
27. CA-98-037, Revision 6, Environmental Qualification (DOR) of Robertshaw Level Switch
28. CA-98-038, Revision 7, Environmental Qualification (50.49) of Rockbestos Coax Cable
29. CA-98-039, Revision 7, Environmental Qualification (DOR) of Rosemount 1153A Transmitters
30. CA-98-040, Revision 11, Environmental Qualification (50.49) of Rosemount 1153B Transmitters
31. CA-98-041, Revision 7, Environmental Qualification (50.49) of Rosemount Conduit Seals
32. CA-98-042, Revision 8/8B, Environmental Qualification (DOR) of Rotork "A" Range Actuators
33. CA-98-043, Revision 3/3A, Environmental Qualification (50.49) of Rotork Valve Operators
34. CA-98-044, Revision 8, Environmental Qualification (DOR) of Static O-ring Pressure Switches
35. CA-98-046, Revision 7, Environmental Qualification (DOR) of Yarway Level Indicator/Transmitter
36. CA-98-047, Revision 6, Environmental Qualification (50.49) of Samuel Moore Instrument Cable
37. CA-98-049, Revision 10, Environmental Qualification (50.49) of Valcor Solenoid Valves
38. CA-98-050, Revision 6, Environmental Qualification (50.49) of DG O'Brien Electrical Penetrations
39. CA-98-051, Revision 6, Environmental Qualification (50.49) of Reliance Motors
40. CA-98-052, Revision 4, Environmental Qualification (50.49) of Tavis Flow Transmitter
41. CA-98-053, Revision 3, Environmental Qualification (DOR) of ITT Grinnel/Conoflow I/P Transducer

42. CA-98-054, Revision 3, Environmental Qualification (50.49) of Consolidated Control Relays
43. CA-98-055, Revision 8, Environmental Qualification (DOR) of General Atomic Radiation Detector
44. CA-98-059, Revision 5, Environmental Qualification (50.49) of Kerite Cable/Termination
45. CA-98-060, Revision 6, Environmental Qualification (50.49) of Westinghouse Motor Control Box
46. CA-98-062, Revision 2, Environmental Qualification (50.49) of Gould Contactor/Disconnect
47. CA-98-064, Revision 7, Environmental Qualification (50.49) of Eaton Thermocouple Extension Cable
48. CA-98-065, Revision 8, Environmental Qualification (50.49) of Brand Rex 600V Cable
49. CA-98-066, Revision 4, Environmental Qualification (50.49) of Boston Insulated Wire Control Cable
50. CA-98-067, Revision 5, Environmental Qualification (50.49) of CONAX Electrical Connector Seal Assembly (ESCA)
51. CA-98-068, Revision 4, Environmental Qualification (50.49) of CONAX RTDs
52. CA-98-069, Revision 7, Environmental Qualification (50.49) of Patel Conduit Seals
53. CA-98-070, Revision 5, Environmental Qualification (50.49) of Patel Conformal Coating
54. CA-98-071, Revision 3/3A, Environmental Qualification (50.49) of EGS Grayboot Connector
55. CA-98-072, Revision 1, Environmental Qualification (50.49) of EGS Quick Disconnect
56. CA-98-073, Revision 6, Environmental Qualification (50.49) of Raychem/Swagelok Conduit Seals
57. CA-98-075, Revision 4, Environmental Qualification (50.49) of Weed Thermocouples
58. CA-98-076, Revision 5, Environmental Qualification (DOR) of Rome Cable Type SIS Switchboard Wire
59. CA-98-077, Revision 1, Environmental Qualification (50.49) of Eaton Cutler-Hammer Relays
60. CA-98-078, Revision 3/3A, Environmental Qualification (50.49) of Fenwal/Patel Temperature Switch
61. CA-98-079, Revision 2, ITT-Royal PVC Cable Environmental Qualification Calculation
62. CA-98-080, Revision 3, Okonite Control Cable EQ Calculation
63. CA-98-081, Revision 3, Triangle Triolene-Trioseal Control Cable, Polyethylene Insulated Environmental Qualification Calculation
64. CA-98-082, Revision 3, Environmental Qualification (DOR) of MNGP-A Control Cable

65. CA-98-083, Revision 3, Environmental Qualification (DOR) of MNGP-B PVC Insulated Cable

66. CA-98-084, Revision 1, Amphenol Connectors Environmental Qualification Calculation

67. CA-98-085, Revision 1, Pyco Temperature Elements EQ Calculation

68. CA-98-086, Revision 1, Environmental Qualification (50.49) of Static O-Ring Pressure Switches

69. CA-98-101, Revision 1, Environmental Qualification (50.49) of General Electric Terminal Blocks

70. CA-98-103, Revision 2, Environmental Qualification (50.49) of Patel P-1 Thread Sealant

71. CA-98-104, Revision 2, Environmental Qualification (50.49) of Rockbestos Firewall SR Control Cable

72. CA-98-107, Revision 2, Environmental Qualification (50.49) of Rockbestos Firewall III Cable and SIS Wire

73. CA-98-108, Revision 1, Environmental Qualification (50.49) of Rockbestos Firewall EP Power Cable

74. CA-98-109, Revision 1, Environmental Qualification (50.49) of Valcor MSIV Solenoid Valves

75. CA-98-128, Revision 1/1A, Environmental Qualification (50.49) of UCI Electrical Tape Terminations

76. CA-02-197, Revision 1, Environmental Qualification (50.49) of Dow Corning 3-6548 Silicone RTV Foam

77. CA-03-096, Revision 2, Environmental Qualification (50.49) of Loctite PST 580 Thread Sealant

78. CA-03-105, Revision 1, Environmental Qualification (50.49) of Scotch 130C and 69 Electrical Tape

79. CA-05-137, Revision 1, Environmental Qualification (50.49) of Fisher Controls Model 546 E/P Converter

80. CA-05-138, Revision 0, Environmental Qualification (50.49) of Cutler-Hammer Motor Starter and Control Transformer

81. CA-05-140, Revision 0, Environmental Qualification (50.49) of ASCO Scram Solenoid Pilot Valves

Other Calculations

82. CA-05-133, Revision 0A (EC 13044), EQ Validation of Normal Temperatures in the Reactor Building and Drywell

83. EC 11869/12147 (CA-99-110, Revision 1), EPU - CALC 99-110 - CRD HELB 935 Elev 2059 MWT Analysis Case RFD-RB-CRD-B-R1.

84. EC 11869 (CA-08-006, Revision 0), CRD Line Break (HELB) for EPU 2059 MWt, 935' elevation Reactor Building (with operator action)
85. EC 11869 (CA-97-039, Revision 3), FEEDWATER HELB IN THE STEAM CHASE, For EPU 2059 MWt, Analysis Case RFD-RB-FW-B-16-R3
86. EC 11869 (CA-97-149, Revision 2), FEEDWATER CRITICAL CRACK IN THE STEAM CHASE, For EPU 2059 MWT, Analysis Case: RFD-RB-FW-C-16-R2
87. EC 11869 (CA-07-058, Revision 0), RWCU Break in the Steam Chase for EPU, 2059 MWt, Analysis Case RFD-RB-RWCU-B-16-Rev.0
88. EC 11869 (CA-07-062, Revision 0), RWCU Break in the RWCU Room for EPU, 2059 MWt, Analysis Case RFD-RB-RWCU-B-28-Rev.0
89. EC 11869 (CA-08-008, Revision 0), RWCU HELB in the RWCU Room at the Regen Hx Outlet, for EPU 2059 MWt, Analysis Case RFD-RB-RWCU-B-30-R0
90. EC 11869 (CA-96-175, Revision 1), RWCU HELB IN THE RWCU Room, for EPU 2059 MWt, Analysis Case RFD-RB-RWCU-B-30-R1
91. EC 11869 (CA-07-057, Revision 0), RWCU HELB in the RWCU Room for Extended Power Uprate, 2059 MWt, Analysis Case RFD-RB-RWCU-B-32-Rev. 0.
92. EC 11869 (CA-96-082, Revision 1), RWCU Crack in the RWCU Hx. Room, For EPU 2059 MWt, Analysis Case RFD-RB-RWCU-C-30-R1
93. EC 11869 (CA-07-061, Revision 0), RWCU Crack in the RWCU room for EPU, 2059 MWt, Analysis Case RFD-RB-RWCU-C-31-Rev.0
94. EC 11869 (CA-07-060, Revision 0), RWCU Crack in the RWCU room for EPU, 2059 MWt, Analysis Case RFD-RB-RWCU-C-32-Rev.0
95. EC 13182 (CA-08-162, Revision 0), HPCI Room Heatup For DBA Under EPU Conditions
96. CA-04-098, Revision 1, Instrument Setpoint Calculation, Recirculation Riser Differential Pressure High (LPCI Loop Select)
97. EC 11869 (CA-08-067, Revision 0), Sixty-Year Normal Radiation Dose for EQ
98. EC 12880 (CA-08-085, Revision 0), Post-LOCA Reactor Building Heatup Analysis for EPU
99. EC 12958 (CA-08-145, Revision 1), MNGP EQ - Scaling Factors for EPU TIDs (ALION-CAL-MNGP-4370-01 Rev. 1)
100. CA-94-086, Revision 3, Maximum Allowable Leakage Rates and Test Acceptance Criteria for SRV Accumulator Systems 'D' and 'G'
101. CA-03-099, Revision 1, Drywell Temperature and Pressure EQ Profiles
102. CA-08-125 (EC 12942), Revision 0, "MNGP ECCS Pump System Resistance Evaluation (acceptance of vendor calculation: Shaw Document 12767201-M-001 Revision 0 associated with Task T0407)
103. CA-96-166, Revision 1, Drywell Flooding for Post DBA-LOCA

Drawings

104. NF-73880, Revision G, RWCU from Recirc System to RWCU Pumps (Line REW-3-4")
105. NX-7905-6-1, Revision B, Function Control Diagram Residual Heat Removal System
106. NX-9301-73-1, Revision H, Reactor Building Elevation 935'-0" Communications
107. NX-13142-18, Revision M, Torus Water
108. NX-13142-26, Revision 77, Torus Water Reactor Building

Miscellaneous Documents

109. EQ-Part-B, Revision 9, EQ Central File Part B Environmental Specifications
110. Task Report T0400, Revision 1, Containment System Response
111. Task Report T0610, Revision 0, Power Dependent HVAC
112. Task Report T0803, Revision 0, Radiation Levels
113. Task Report T1009, Revision 0, HELB Subcompartment Evaluation
114. NRC Safety Evaluation Report, September 16, 1998, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: Power Uprate Program (TAC No. M96238).
115. IE Bulletin 79-01b, January 14, 1980, Environmental Qualification of Class IE Equipment (Enclosure 4 includes "DOR Guidelines")
116. EC 12421, EPU - Effect of Reactor Building HELB Liquid Breaks on EQ Specifications, Part B. EPU Task Report T1009, Feb 2008
117. EWRA 1131374-19, T1004 - review EQ-PART B area volumes
118. CAP 1125675, Non-conservative HELB Gothic Model on HELBs in the Condenser Room
119. NUREG-0588, Revision 1, Interim Staff Position on Environmental Qualification of Safety-Related Electric Equipment
120. MPS-0167-AB, Revision 2, (GE Specification 21A1060AB) Relief Valve in Steam Piping
121. MPS-0277, Revision 3, (GE Specification 22A1132) Containment Isolation System
122. CAP 01106163, Ambient temperature from vendor calc are overly conservative
123. CAP 01115107, USAR App I HELB TB Volume Designator mismatch with EQ Volume Designators in EQ-Part-B
124. Modification 90Z052, EQ Instrument Cable Replacement
125. Modification 97Q055, SCTMT Trip of the Radwaste Exhaust Fans
126. DBD-T.13, Revision 1, Regulatory Guide 1.97
127. GE Letter 0000-0084-5876-R-0 (DRF 0000-0060-9169), dated April 25, 2008, MNGP EPU T0400 FTR CSV File Turnover.

128. 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

129. Regulatory Guide 1.89, Revision 1, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants

130. DBD-T.04, Revision 76, Environmental Qualification

131. EC 11988, EPU - Inboard MSIV AVCO 4-Way Control Valve Replacement with New Solenoids that do not Require as Much DP to Operate

132. EC 12044, CRD SCRAM Solenoid Pilot Valve Replacement

133. EC 13086, EPU - EQ Transmitter Upgrades for EPU Conditions

134. EC 11561, Outboard MSIV Limit Switch Replacement

135. Qual Doc 08-013 (reserved), McDonnell & Miller Flow Switches (50.49)

136. GAR 01150954, Measure filter distances in SBGT Room

3.0 EVALUATION

3.1 Methodology

Item	Evaluation Method	Task Application
1	NRC approved or accepted method (including level 2 computer codes)	None
2	Level 2 computer code not approved by NRC	None
3	Non Level 2 numerical analysis	Arrhenius Methodology, Excel spreadsheets, hand calculations
4	Qualitative method	Reviews of existing analyses are used as indicated in Section 3.4.

3.2 Definitions

Item	Analysis Category	Description
1		None

3.3 Key Input, Assumptions, and Results

3.3.1 Key Inputs

Item	Parameter	EPU Inputs/Impacts	CLTP Inputs
1	Normal Temperatures	Location dependent. EPU impacts on normal design temperature limits for EQ areas per Task Report T0610 [111].	Location dependent. CLTP values per calculation [82] (Drywell/Steam Chase) and monitored data CAP 01106163[122]
2	Drywell EQ Accident Reponse	Per Task Report T0400 [110], steam break accident results in the bounding temperature response while the design basis LOCA case results in the bounding pressure response. Task Report T0400 result files provide the time dependent Drywell temperature profile. Peak Drywell temperature resulting from the steam line break event was shown to be 338°F in Figure 5-15 of Task Report T0400. This is considered for the first 300 seconds of the Drywell response, the remainder of the 180 day post-accident profiles follows from the tabular data presented in Section 5.1.3.5 of Task Report T0400. For conservatism, the peak Drywell pressure of 44.1 psig is taken from Section 3.3.1 of Task Report T0400 [110] which used methods[2] to maximize Drywell pressure response under a LOCA in the short-term.	The CLTP basis for the Drywell temperature and pressure response is provided by calculation CA-03-099 [101].

[2] Results of Section 3 of Task Report T0400 based on Moody slip critical flow model whereas the results of Section 5 are based on Moody HEM critical flow model. Using the peak Drywell pressure of 44.1 psig (58.8 psia) from Section 3 of Task Report T0400 is conservative. Section 5.3.2, Item 4 of Task Report T0400 indicates a Drywell pressure of 50 psia (35.3 psig) at maximum EQ boundary peak temperature conditions.

Item	Parameter	EPU Inputs/Impacts	CLTP Inputs
3	Normal Radiation	Normal radiation doses under EPU conditions plant-wide based on calculation CA-08-067 [97]. This calculation includes the EPU moisture carry-over affects for Reactor Building Volumes 5 and 8 and Turbine Building Volumes 2, 3, 13, 14, 25, 41, 42, and 44 as prescribed in Task Report T0803 [112].	Normal radiation for CLTP conditions have been reassessed under the same calculation (CA-08-067 [97]) to remove excess conservatisms, update conditions with newer dose rate surveys, correct Drywell dose rate for CLTP power, and adjust doses consistently for bounding hydrogen water chemistry affects. In general, the revised CLTP normal 60-year doses were reduced from previously determined values due to increased accuracy and removal of arbitrary conservatisms present in the former calculation.

Item	Parameter	EPU Inputs/Impacts	CLTP Inputs
4	Reactor and Turbine Building HELB	For EPU HELB affects, Task Report T1009 [113] provides peak condition inputs for temperature, pressure, and liquid level effects due to EPU and compares to CLTP values. Source calculations for Reactor Building EQ evaluations made within this report include: • CA-99-110 [83], case CRD-B-18-R1 • CA-08-006 [84], case CRD-B-18-R0 • CA-97-039 [85], case FW-B-16-R3 • CA-97-149 [86], case FW-C-16-R2 • CA-07-058 [87], case RWCU-B-16-R0 • CA-07-062 [88], case RWCU-B-28-R0 • CA-08-008 [89], case RWCU-B-30-R0 • CA-96-175 [90], case RWCU-B-30-R1 • CA-07-057 [91], case RWCU-B-32-R0 • CA-96-082 [92], case RWCU-C-30-R1 • CA-07-061 [93], case RWCU-C-31-R0 • CA-07-060 [94], case RWCU-C-32-R0	EQ-Part-B [109][3]
5	Reactor Building Post-LOCA Heatup Temperatures	For EPU, the Reactor Building Post-LOCA temperature conditions are provided by calculation CA-08-085 [98].	EQ-Part-B [109]
6	Accident Radiation	For EPU, accident radiation doses are provided by calculation CA-08-145 [99].	EQ-Part-B [109]

[3] EQ-Part-B is the MNGP document that provides the environmental parameters for the EQ program. The current revision reflects CLTP operating conditions.

3.3.2 Key Assumptions

Item	Assumption	Ref./Basis
1	Submergence level in the Drywell is unchanged by EPU. (Vent flows continue to exceed maximum ECCS flows)	For CLTP conditions, calculation CA-96-166 [103] determines a maximum submergence level of 922'-0" for the Drywell. The calculated flow through the containment vents is 27,233 gpm which exceeds the maximum ECCS (RHR and CS) flow of 25,560 gpm under CLTP conditions. For EPU, Attachment 125 of CA-08-125 [102] provides a maximum total flow into reactor/Drywell (at reactor pressure of 0 psig) from all six (6) RHR and CS pumps of 25,319 gpm. As such, the Drywell submergence level is not expected to increase under EPU, particularly considering that all six pumps are not assumed to run at the same time.

3.3.3 Key Results

Item	Parameter	CLTP Value	EPU Value	Comments
1	Normal Ambient Temperature	As detailed in EQ-PART-B [109]	As detailed in Task Report T0610 [111]	Task Report T0610 [111] shows that the design temperature of most areas in the plant is not impacted by EPU. One plant area warrants further discussion: Main Steam Pipe Chase – a marginal temperature increase in this area of 0.4°F with indication that temperature in this area will remain below the design value of 130°F. Since the CLTP EQ value for this area is 135°F per EQ-Part-B [109], the EPU normal temperature conditions for this area are bounded. All other area normal temperatures remain bounded by the normal temperatures used in the EQ analyses.

Item	Parameter	CLTP Value	EPU Value	Comments
2	Normal Radiation Dose	As detailed in EQ-PART-B [109] and CA-08-067 [97]	As detailed in CA-08-067 [97]	Per Task Report T0803 [112], normal radiation doses change due to EPU, by 13% in all Reactor Building Areas except Reactor Building Volumes 5, 8 and 16. Normal plant shut-down doses in RB Volumes 5 & 8 increase by a factor of 11.3, while normal operating doses in RB Volumes 5, 8, and 16 do not change due to EPU. CA-08-067 [97] computes the 60 years dose values for EPU and CLTP conditions replacing previous CLTP normal dose basis calculation (CA-04-034). The new calculation corrects survey data for consistent hydrogen water chemistry flow rates where appropriate and incorporates an updated set of plant radiation survey data. The EPU shutdown factor of 11.3 for RB Volumes 5 & 8 was also conservatively included in the dose computation for EPU operation. Previous CLTP 60 year dose basis calculation included several layers of conservatism that resulted in higher normal dose for most areas of the Reactor Building. The revised CLTP 60 year normal dose is utilized in this task report as well as the EPU 60 year normal dose, both taken from CA-08-067 [97].
3	Accident Radiation Dose	As detailed in EQ-PART-B [109])	As detailed in CA-08-145 [99]	Accident Gamma doses increase by a maximum factor of 1.083 (Standby Gas Treatment Room is bounding). Beta doses for EQ, as provided by DOR Guidelines [115], remain valid for EPU. All equipment is qualified to the new Total Integrated Dose (TID), the sum of the normal and accident doses, with details noted in the supporting evaluation, Section 3.4..
4	Reactor Building HELB Profiles	As detailed in EQ-PART-B [109]	As highlighted in Task Report T1009 [113]	The CLTP inputs include steam HELB events while the EPU inputs are only liquid HELB events. Steam HELB effects do not change at EPU conditions. As such, the peak CLTP

Item	Parameter	CLTP Value	EPU Value	Comments
				conditions for some Reactor Building areas bound the EPU conditions. In those cases, no further review for EPU impact was considered necessary for the EQ equipment.
5	RB Post-LOCA Heat-Up	As detailed in EQ-Part-B [109]	As detailed in CA-08-085 [98]	The post-LOCA temperature conditions in the Reactor Building under EPU are generally higher than determined under CLTP, except for the RHR pump rooms which decreased marginally due to modeling refinements. The CLTP EQ analysis of PLHU operation for common equipment items included Drywell locations, utilizing bounding conditions in a conservative manner to bound worst-case PLHU conditions plant-wide. In other cases, some equipment only serves short-term HELB functions. With these considerations, approximately 25% of the CLTP EQ files required further review for PLHU conditions under EPU, these are included in Attachment A of this evaluation. In all other cases, the CLTP EQ files continue to provide a bounding analyses for EPU conditions.
6	Turbine Building HELB	As detailed in EQ-Part-B [109]	As highlighted in Task Report T1009 [113]	Several Turbine Building volumes become harsh as a result of HELB conditions under EPU. These Volumes include 7, 8, 10, 11, 13, 16, 20, and 27 (designations per USAR, App. I). As detailed in AR 1131374-19 [117], five (5) safety related cables runs were discovered as the only safety-related equipment/cables either in or traversing these Turbine Building areas. Based on the identified cables, three EQ files are being revised to include the specific Turbine Building volumes. These file revisions are tracked in the Corrective Action Program.
7	Drywell Accident Profile	As detailed in EQ-PART-B [109]	As detailed in Task Report T0400 [110]	All EQ equipment is qualified for the new Drywell peak temperature of 338°F; except for the D.G. O'Brien electrical penetration triaxial plugs (EQ File CA-98-050 [38]).

Item	Parameter	CLTP Value	EPU Value	Comments
				However, these penetrations serve the High Range Radiation Monitors (HRRM) and were accepted to only be required for DBA LOCA events [114]. The DBA LOCA case has a lower peak temperature of 324.7°F (RSLB20 case per GE letter 0000-0084-5876-R-0, DRF 0000-0060-9169 [127]), but exceeds the qualification testing at 320°F for less than 30 seconds. The penetrations are mounted on the Reactor Building side of Drywell penetration pipe (NX 9301-73-1 [106]). Configured in this manner, the only component running through into the Drywell penetration tube and into the Drywell is the cables serving the HRRM detectors. Task Report T0400 [110] provides a peak Drywell wall temperature of 278°F under the worst-case steam line break event. Given the Reactor Building side mounting of the D.G. O'Brien penetrations, there is sufficient evidence that the penetration will not experience the Peak Drywell air temperature of 324.7°F postulated under the DBA LOCA case with Drywell spray at EPU power conditions. Therefore, the equipment remains qualified at EPU conditions. Some EQ equipment will have less than the recommended margin of 15°F in IEEE Standard 323 at EPU conditions. Margin issues already exist under CLTP conditions for some EQ equipment, and were previously determined to be acceptable if the test profile included two transients. Since all EQ equipment with less than a 15°F margin included a test with two transients, all EQ equipment in the Drywell is considered qualified to peak EPU temperature conditions (excludes DOR equipment, which does not require margin). Per Task Report T0400 [110], the peak Drywell pressure under

Item	Parameter	CLTP Value	EPU Value	Comments
				Moody slip critical flow methods may increase under EPU within the first 30-seconds to 58.8 psia. Since there are no known time-dependent aging effects for pressure, all EQ equipment in the Drywell were shown to have qualified pressure levels in excess of the new peak Drywell pressure in Section 3.4.2 of this task report. The Drywell EQ equipment remains qualified for the EPU accident pressure conditions.
8	Submergence Level	As detailed in EQ-PART-B, (Reference 2.3)	As detailed in Task Report T1009 [113]	Submergence level for many Reactor Building and Turbine Building areas increase under EPU as detailed in T1009 [113]. However, there are no additional EQ components subject to submergence at EPU conditions. The details of the EPU higher submergence levels are provided in the supporting evaluation section of this report.

3.4 Supporting Evaluations

3.4.1 Normal Temperature Evaluation

Task Report T0610 [111] provides an evaluation of the impact EPU will have on normal design ambient temperatures of the Reactor Building. For general areas of the Reactor Building, there was no increase of significance noted. The EQ program utilizes monitored temperature data, the most recent of which has been assessed for impact in AR 01106163 [122]. The results of the AR evaluation were to increase the normal ambient temperature conditions outside the Drywell. For the Drywell, calculation CA-05-133 [82] provides a basis for the normal ambient temperatures. Task Report T0610 indicates no change in normal Drywell temperatures for EPU. Normal plant area ambient temperature will continue to be monitored by the EQ program in lieu of using the maximum design temperature for assessing qualified lifetimes. As such, there is no impact of EPU conditions on normal plant temperature inputs to qualified life assessments. The impact of the recently increased ambient temperatures due to measured data was addressed in the corrective action program.

3.4.2 Drywell Temperature and Pressure Evaluation

The time-dependent Drywell EQ temperature profile is graphically compared for CLTP and EPU plant conditions in Figure 3.4.2-1.

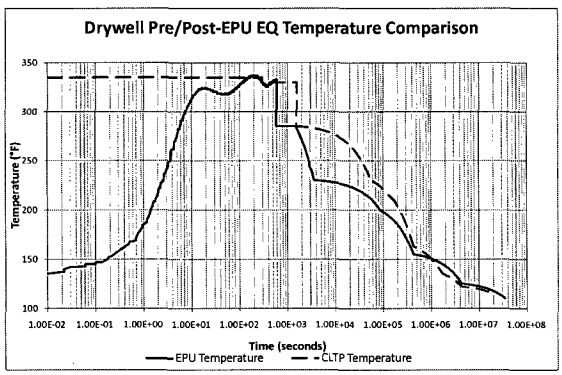

Figure 3.4.2-1

The CLTP EQ Drywell temperature does not bound the peak EPU temperature in the short term (by 3°F within the first 600 seconds) and also at greater than a million seconds in the long term. The peak temperature difference (3°F) can be justified by the application of existing EQ margins,

thermal lag considerations, or testing that includes dual transient exposures. In the short term, the new EPU peak temperature of 338°F (335°F for CLTP) is not a significant concern for the EQ components. In the long term (greater than a million seconds), the temperatures of concern are less than 150°F. Existing EQ analyses for equipment functioning over the long term have demonstrated qualification for post-accident periods with significant margin with respect to temperatures < 150F.

Although the peak Drywell accident pressure under EPU may increase to 58.8 psia (44.1 psig), this is not a concern as the minimum qualified test level was 50 psig as shown in Table 3.4.2-1 that follows.

The EQ components located within the Drywell and their peak qualification parameters are shown in Table 3.4.2-1.

Table 3.4.2-1

EPU Maximum Drywell Conditions:		**Temp 338°F**	**Press 44.1 psig**	
MNGP Drywell EQ Equipment				
Calculation	**Title**	**Qual Temp (°F)**	**Qual Press (psig)**	**Qualified at EPU**
CA-98-004	ASCO Solenoid Valves (Normally Energized)	346	68	Y
CA-98-005	ASCO Solenoid Valves (Normally De-energized)	346	68	Y
CA-98-008	Automatic Valve Solenoid Valves	360	63	Y
CA-98-017	GE Cable (XLPE type, SI-57275/58109)	340	62	Y
CA-98-020	General Electric Containment Penetrations	340	63	Y
CA-98-025	Limitorque MOVs (DOR, class RH motor)	329	91	*
CA-98-026	Limitorque MOVs (50.49, class RH, AC motor)	340	55	Y
CA-98-026	Limitorque Fibrite Switches	420	72	Y
CA-98-032	Namco Limit Switches (EA740 w/o EC210)	352	75	Y
CA-98-032	Namco Limit Switches (EA180 & EC210)	365	72	Y
CA-98-033	Namco Quick Disconnects EC210	353	72.5	Y
CA-98-036	Raychem WCSF-N Splices	350	120	Y
CA-98-038	Rockbestos Coax Cable	346	122	Y
CA-98-050	DG O'Brien Electrical Penetrations	320	66	**
CA-98-050	DG O'Brien Electrical Penetrations (Plugs)	345	72	Y
CA-98-055	General Atomic Radiation Detector	355	78	Y
CA-98-064	Eaton Thermocouple Extension Cable	375	72	Y
CA-98-065	Brand Rex 600V Instrument Cable	385	98.3	Y
CA-98-067	CONAX Electrical Connector Seal	375	75	Y
CA-98-069	Patel Conduit Seals	354	77	Y

Table 3.4.2-1

EPU Maximum Drywell Conditions: **Temp 338°F** **Press 44.1 psig**

MNGP Drywell EQ Equipment				
Calculation	**Title**	**Qual Temp (°F)**	**Qual Press (psig)**	**Qualified at EPU**
CA-98-070	Patel Conformal Coating (Inside Drywell)	503	75.3	Y
CA-98-071	EGS Grayboot Electrical Connectors	450	81	Y
CA-98-072	EGS Quick Disconnect	435	77	Y
CA-98-075	Weed Thermocouples	503	75.3	Y
CA-98-103	Patel P-1 Thread Sealant	400	57.3	Y
CA-98-104	Rockbestos Firewall SR Cable	351	132.1	Y
CA-98-107	Rockbestos Firewall III/SIS Cable	342	117.8	Y
CA-03-096	Loctite PST 580 Thread Sealant	364	50	Y

* The Limitorque actuator installed in the Drywell is MO-2397 (RWCU inboard isolation) and is qualified in accordance with DOR Guideline [115] criteria (Limitorque, Class H motor) as documented in EQ Calculation CA-98-025 [18]. The actuator serves a Group 3 isolation function for the RWCU system as indicated in USAR Table 5.2-3b. The current EQ evaluation presents a thermal lag evaluation for the equipment's short operating time. A review of the evaluation indicates that given the small magnitude of the peak temperature change of 3°F, EPU will have a negligible impact on the accident/post-accident temperature qualification of this actuator. Therefore, the equipment remains qualified at EPU conditions.

** The D.G. O'Brien electrical penetrations only serve the Containment High Range Radiation monitors. The peak test temperature of 320°F does not currently bound the peak EQ temperature conditions which includes steam line break events. The equipment is currently qualified for LOCA only Drywell temperature profile and has been accepted as qualified for LOCA only in an NRC Safety Evaluation Report [114].

The Drywell temperature response under the design basis LOCA case RSLB20 (with Drywell spray at 600 seconds) under EPU is shown in Figures 5-4 of Task Report T0400 [110]. GE Letter 0000-0084-5876-R-0 [127] indicates that the input data file for Figure 5-4 of T0400 is the file: MNGP_EPUMP_SHEX06A_RSLB20.CSV. The peak Drywell temperature under this event as shown in this file is 324.7°F. Using the time and Drywell temperature data from this file and the test data for the D.G. O'Brien penetrations and electrical plugs as given in EQ File CA-98-050 [38], the graphical comparison is shown in Figures 3.4.2-3 and 3.4.2-4.

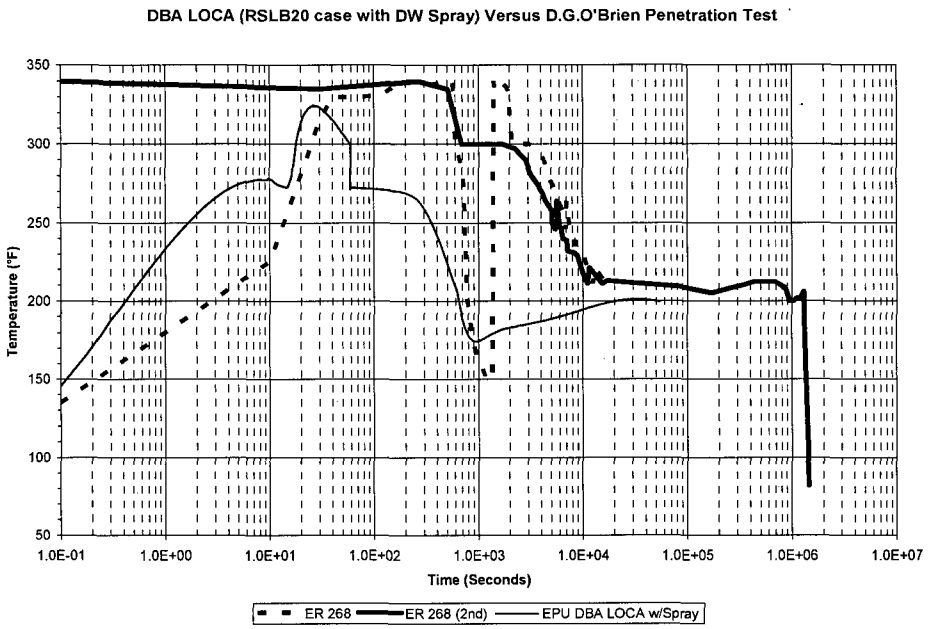

Figure 3.4.2-3

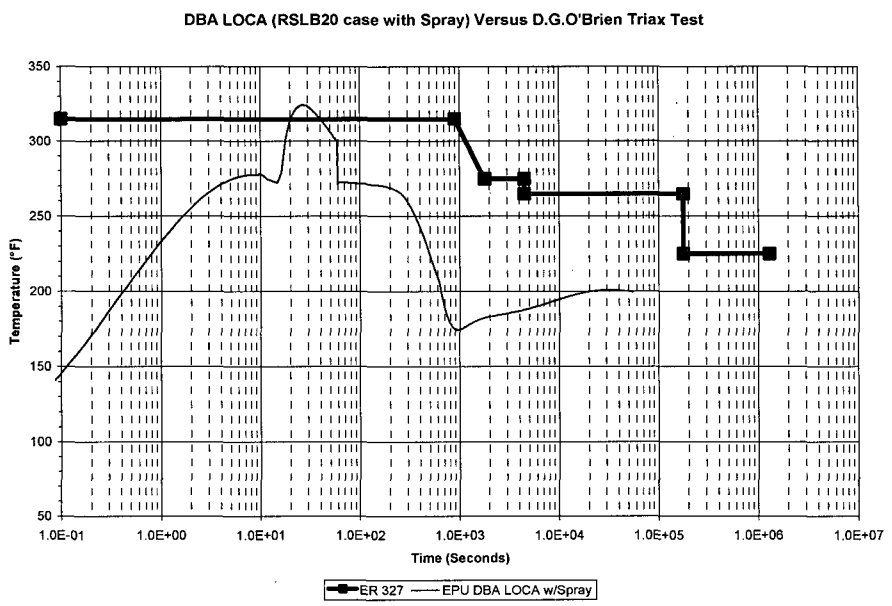

Figure 3.4.2-4

As shown the electrical plug testing does not bound the peak Drywell temperature of 324.7°F for the DBA LOCA case with Drywell spray. Testing was sustained at 320°F as shown in the figure above. Although the difference is <5°F, there is reasonable basis for establishing qualification for EPU DBA LOCA conditions based on two points: the installed

configuration and the calculated Drywell wall temperature under an MSLB event having the higher peak Drywell temperature of 338°F.

The installation details demonstrate that the penetrations consist of a header plate and feedthrough modules mounted on the Reactor Building side of Drywell penetration pipe (NX 9301-73-1 [106]). Configured in this manner, the only component running through into the Drywell penetration tube and into the Drywell is the cables serving the HRRM detectors. Task Report T400 [110], Containment System Response, provides a peak Drywell wall temperature of 278°F under the worst-case steam line break event. Given the Reactor Building side mounting of the D.G. O'Brien penetrations, there is sufficient evidence that the penetration will not experience the Peak Drywell air temperature of 324.7°F postulated under the DBA LOCA case with Drywell spray at EPU power conditions. Therefore, it is reasonable to conclude that the equipment remains qualified at EPU conditions.

3.4.3 Reactor Building HELB Overall Evaluation

According to T1009 [113], a constant pressure EPU has no effect on the steam pressure or enthalpy at the postulated break locations. Therefore, EPU has no effect on the mass and energy releases from an HELB in a steam line. Therefore, no plant-specific evaluation is required for steam line breaks to support EPU.

Under EPU Task T1009, twelve high energy liquid break or critical crack scenarios were considered. The following data form the basis of this evaluation:

HELB Case#	Calculation
CRD-B-18-R1	CA-99-110 [83]
CRD-B-18-R0	CA-08-006 [84]
FW-B-16-R3	CA-97-039 [85]
FW-C-16-R2	CA-97-149 [86]
RWCU-B-16-R0	CA-07-058 [87]
RWCU-B-28-R0	CA-07-062 [88]
RWCU-B-30-R0	CA-08-008 [89]
RWCU-B-30-R1	CA-96-175 [90]
RWCU-B-32-R0	CA-07-057 [91]
RWCU-C-30-R1	CA-96-082 [92]
RWCU-C-31-R0	CA-07-061 [93]
RWCU-C-32-R0	CA-07-060 [94]

Using the output data from these calculations, the peak parametric EPU HELB conditions are shown below along with the corresponding bounding qualification conditions for the respective Reactor Building Volume. Notes following the table explain or justify the differences between CLTP and EPU pressure and submergence levels. HELB temperature differences are evaluated in Section 3.4.4 of this task report. Note that the CLTP temperatures below include the effects of steam HELBs while the EPU results only reflect the twelve liquid break cases. The EPU steam HELB temperatures are inherently included in the CLTP steam HELB temperatures as the

constant pressure power uprate methodology of the EPU implementation does not affect steam HELB events.

Table 3.4.3-1
Peak Reactor Building HELB Condition Comparison

RB Volume	CLTP Temp (°F)	CLTP Press (psia)	CLTP Submergence (ft)	EPU Temp (°F)	EPU Press (psia)	EPU Submergence (ft)
1	142.97	14.86	0.05	114.2	14.81	0.4 (1)
2	142.97	14.85	0.01	115.2	14.8	0
3	143.8	14.98	0.05	115.7	14.83	0.4 (1)
4	144.4	14.97	0	135.6	14.82	0
5	256	15.15	0.05	121.3	14.81	0.5 (1)
6	263.6	15.63	3.95	157.2	14.8	0.6
7	240.4	15.19	4.38	184.5	14.84	0.8
8	272.6 (2)	16.05	1.11 (2)	140.7 (2)	14.8	0.6 (2)
9	187.6	15.59	0.01	127.4	14.79	0.5 (1)
10	159.9	15.59	0	127.2	14.79	0.5 (1)
11	158.7	15.59	0	127.3	14.79	0.5 (1)
12	193	15.59	0.01	127.6	14.79	0.5 (1)
13	127.3	14.88	0.24	166.2	14.8	0
14	153.3	14.83	0.27	174.7	14.8	0
15	209.6	15.3	0.05	180.2	14.8	0
16	311.3	21.16	6.68	213.1	15.22	8.8 (1)
17	222.4	15.14	0	130.6	14.79	0
18	208.1	15.11	0.68	209.7	14.83	0
19	175.9	15.13	0.53	209.3	14.83	0.1
20	112.5 (PLHU) 104.8 (HELB)	15.14	0	179.4	14.8	0.1 (3)
21	112.2	14.87	0	108	14.79	0
22	170.4	14.84	0	184	14.79	0
23	143.7	14.87	0	164	14.79	0
24	104	14.7	0	104.4	14.7	0
25	145.8	14.85	0	183.5	14.79	0
26	109.1	14.89	0	104.4	14.7	0
27	203.6	14.99	0.01	211.7	14.83	0
28	216.7	15.85	0.38	213.3	15.03	1.6 (4)
29	214.6	15.85	0.3	213.5	15.4	1.7 (4)
30	220.5	17.2	0.27	218.2	16.52	1.4 (4)
31	188.3	17.19	0.27	213	16.52	1.4 (4)
32	164.1	15.85	0.19	219.8	17.09 (5)	1.7 (4)
33	128.3	14.99	0	211.7	14.83	0
34	168.3	14.83	0.005	191.4	14.79	0

Table 3.4.3-1
Peak Reactor Building HELB Condition Comparison

RB Volume	CLTP Temp (°F)	CLTP Press (psia)	CLTP Submergence (ft)	EPU Temp (°F)	EPU Press (psia)	EPU Submergence (ft)
35	207.8	14.84	0.02	199.2	14.79	0
36	142.4	14.88	0	141.6	14.79	0
37	112.8	14.88	0	143.9	14.8	0
38	100	14.7	0	100.3	14.7	0
39	215.3	14.87	0.01	112.2	14.8	0
40	126.2	14.84	0	127.7	14.8	0
41	106.9	14.88	0	104.4	14.7	0
42	204.7	14.85	0.01	198.8	14.79	0
43	128.3	14.83	0.01	122.2	14.78	0
44	160.8	14.82	0.01	138.1	14.78	0
45	169.7	14.85	0.01	198.6	14.78	0
46	219.2	14.86	0.02	106.2	14.77	0
47	101.5	14.84	0	187.8	14.78	0
48	131.1	14.8	0.01	137.3	14.76	0

Notes:

1) Higher submergence levels result under EPU due to a RWCU system break in RB Volume 16 (RWCU-B-16-R0). The Reactor Building Volumes affected by the event (those have submergence levels in excess of CLTP values) are: 1, 3, 5, 9, 10, 11, 12, and 16. The up to half foot (6-inch) submergence level postulated for these areas (except the Steam Chase at 8.8-foot) are mounted above this level. The calculated 8.8-foot submergence level in the Steam Chase will engulf MO-2107 (RCIC pump discharge isolation), MO-2068 (HPCI pump discharge inboard isolation), CV-1478 (and SV-1478, Instrument Air to DW isolation valve), and MO-2374 (Main Steam line Drain Isolation outboard). EC 12421 [116] evaluates the effect of submergence on all four devices in the steam chase under the postulated EPU liquid break events and concludes with no impact on plant safety or shutdown capabilities.

2) For CLTP, peak HELB temperature in Volume 8 (HPCI room) is due to the HPCI steam line break in the room, which EPU has no effect. CLTP submergence level of 1.11-foot in the HPCI pump room was previous derived from a Feed water HELB in the Steam Chase (RB Volume 16) as determined in CA-97-039 [85]. For EPU, the HPCI pump room predicted peak EPU HELB temperature and submergence levels shown for this room are due to a liquid RWCU system break in the Steam Chase (RB Volume 16), the revised feedwater break analysis (CA-97-039) under EPU results in only 0.3-foot submergence level in the HPCI pump room (RB Volume 8). Although the EPU peak temperature condition of 140.7°F exceeds the harsh threshold for event consideration, the HPCI system would be lost during this event

as the HPCI injection valve (MO-2068) in the steam chase will be submerged from the same event (8.8 feet postulated from RWCU-B-16-R0 in RB Volume 16). All other EPU liquid break HELB events have peak accident temperatures less than 135°F (or below the 140°F harsh designation) in the HPCI room. As noted in USAR Table I.5-2, the HPCI system is considered unavailable or failed during either a RWCU or Feedwater system break in the Steam Chase. Therefore, there is no change to the environmental qualification of this equipment.

3) Submergence level in RB Volume 20 under EPU is postulated at 0.1-foot due to breaks in the CRD or RWCU systems in RB Volumes 18 and 28, respectively (events CRD-B-18-R1 and RWCU-B-28-R0). The only EQ components in RB Volume 20 are two Limitorque Valve actuators: MO-2013 (RHR Division 2, LPCI injection outboard isolation) and MO-2015 (RHR Division 2, LPCI injection inboard isolation). Isometric drawing NX-13142-18 [107] indicates that both valves/actuator are more than 3-foot above the floor. Therefore, there is no change to the environmental qualification of this equipment.

4) Under EPU, the postulated submergence level in the RWCU rooms is greater under an RWCU system break in RB Volume 30 (EPU event RWCU-B-30-R0). The worst-case submergence is due to the RWCU break in RB Volume 30. All other RWCU break and crack events in the RWCU rooms result in less than 0.4-feet of submergence (RWCU-B-28-R0, RWCU-B-30-R1, RWCU-B-32-R0, RWCU-C-30-R1, RWCU-C-31-R0, and RWCU-C-32-R0). There are no EQ components in RB Volume 29. There are EQ components located in RB Volumes 28, 31, and 32. This equipment and their functions are:

a) Containment Isolation, Valve Position Indication (R.G. 1.97), and Drywell Spray

Equipment	Description	Component needed for function	Volume
AO-2386	DW Purge Exhaust Inboard	Namco Limit switches & SOV (also PS-4666 for open indication)	31
AO-2387	DW Outboard Vent	Namco limit switches & SOV (also PS-4671 for open indication)	31
CV-2385	DW Vent to SGTS	Namco Limit switches & SOV	31
CV-2791	Recirc Suction Line Outboard Isolation	Namco Limit switches & SOV	31
MO-2023	DW Spray Loop 12, Inboard	Drywell spray header valve (actuator)	31

The above equipment is not credited for mitigating the postulated RWCU system breaks which are commensurate with the submergence.

b) Core Spray system components

Equipment	Description	Equipment Function	Volume
MO-1752	12 CS Injection Outboard	CS Injection valve	31
MO-1754	12 CS Injection Inboard	CS Injection valve	31

Isometric drawing NX-13142-26 [108] indicates these valves are located at elevation 978-foot or higher whereas the submergence level for RB Volume 31 is < 964 -foot. Therefore, the actuators are located above the postulated submergence level in this area.

c) RWCU HELB Detection/Mitigation Equipment

Equipment	Description	Equipment Function	Volume
MO-2398	RWCU Inlet, Outboard Isolation	Isolate HELB/Crack	32
TE-6017 (A-D)	RWCU High Area Temp.	Detect HELB	28

Actuator MO-2398 is one of the isolation points for RWCU system breaks. Isometric drawing NF-73880 [104] shows the centerline elevation for valve MO-2398 to be at 974-foot whereas the floor elevation for RB Volume 32 is at 962-foot elevation. Therefore, the actuator is located above the postulated submergence level. The temperature elements are located above the piping to detect the break event, they are not located near or within 2-foot of the floor. The cables originally serving MO-2398 actuator were GE butyl rubber type (SI-58007 and SI-58136). However, modification 90Z052 [124] replaced the entire cable run with Rockbestos Firewall III type. Qualification testing of the Rockbestos cables included a post-accident submergence of at least 18-hours (EQ file CA-98-107 [72]). Thus, if the cables were to be routed below the postulated submergence level, they are considered qualified for that effect.

5) For all EPU liquid HELB events, the CLTP accident pressure is bounding except for RB Volume 32. The only EQ component in RB Volume 32 is MO-2398. This actuator has its environmental qualification documented in EQ calculation CA-98-026 [19] and has demonstrated qualification levels to 129 psig.

3.4.4 Reactor Building HELB Temperature Review

From the table above, the peak EPU liquid break HELB temperatures for Reactor Building Volumes 13, 14, 18, 19, 20, 22, 23, 24, 25, 27, 31, 32, 33, 34, 37, 38, 40, 45, 47, and 48 exceed the CLTP peak HELB temperatures. The peak EPU (or pre-EPU) HELB temperature in Reactor

Building Volumes 24, 38, 40, and 48 are less than the harsh/mild threshold temperature of 140°F, so these areas remain mild. Also, there are no EQ end devices currently installed in Reactor Building Volumes 23, 25[4], 45, and 47.

Although the peak EPU liquid HELB temperature condition in Reactor Building Volume 37 exceeds the CLTP HELB temperature, this area only contains EQ equipment supporting the Standby Gas Treatment system, which is only required for design basis LOCA conditions. The accident for which Standby Gas Treatment is designed to mitigate does not create a harsh temperature environment during the time the equipment needs to function.

The CLTP peak HELB temperature for all other Reactor Building Volumes bounds the peak EPU liquid HELB temperature. As such, the impact of EPU HELB conditions will focus on EQ equipment located in Reactor Building Volumes 13, 14, 18, 19, 20, 22, 27, 31, 32, 33, and 34. Profile comparisons between the CTLP analyzed HELB conditions and those postulated under the EPU revised liquid break events are shown in Attachment B of this task report.

The EQ equipment within these Reactor Building volumes was identified. The affected equipment is listed below along with its qualified temperature level. Where the peak EPU HELB temperature exceeds the qualified level, an evaluation note is provided.

Table 3.4.4-1
Peak Reactor Building HELB Temperature Review

Calculation	Title	RB Volume	Qualified Temp. (°F)	EPU Peak Temp (°F)	EPU Evaluation Note
CA-98-004	ASCO Solenoid Valves (Normally Energized)	18, 19	346	209.7	
CA-98-005	ASCO Solenoid Valves (Normally De-energized)	31	346	213	
CA-98-006	ASCO Pressure Switches	14, 19, 31	210	213	Note 1
CA-98-010	Barksdale Pressure Switch	14, 18, 22, 33	212	211.7	
CA-98-011	Barton Pressure Switches	14, 19	212	209.3	
CA-98-012	Barton Pressure Switches 580A-0, 580A-1	14, 18	200	209.7	Note 2
CA-98-017	G.E. Cable (butyl rubber SI-58007/58136)	13, 14, 18, 19, 20, 22, 27, 33, 34	340	219.8	

[4] Relay 94-5C is borderline location between RB Volumes 22 and 25, RB Volume 22 temperature is bounding.

Table 3.4.4-1
Peak Reactor Building HELB Temperature Review

Calculation	Title	RB Volume	Qualified Temp. (°F)	EPU Peak Temp (°F)	EPU Evaluation Note
CA-98-017	GE Cable (PE type SI-58081)	13, 14, 18, 19, 20, 22, 27, 33, 34	236	219.8	
CA-98-017	GE Cable (XLPE type, SI-57275/58109)	13, 14, 18, 19, 20, 22, 27, 31, 32, 33, 34	340	219.8	
CA-98-020	General Electric Containment Penetrations	14, 18	340	209.7	
CA-98-021	General Electric Terminal Blocks	13, 14, 18, 19, 20, 22, 27, 31, 32, 33, 34	340	219.8	
CA-98-026	Limitorque MOVs (50.49, class RH, AC motor)	13, 19, 20, 22, 31	340	213	
CA-98-026	Limitorque MOVs (50.49, class RH, DC motor)	13, 32	340	219.8	
CA-98-026	Limitorque Fibrite Switches	13, 19, 20, 22, 31, 32	420	219.8	
CA-98-032	Namco Limit Switches (EA740 w/o EC210)	31	352	213	
CA-98-032	Namco Limit Switches (EA180 w/o EC210)	18, 19, 31	361	213	
CA-98-035	Raychem NEIS Seals	18, 19, 31	366	213	
CA-98-036	Raychem WCSF-N Splices	13, 14, 18, 19, 20, 22, 27, 31, 32, 33, 34	350	219.8	

Table 3.4.4-1
Peak Reactor Building HELB Temperature Review

Calculation	Title	RB Volume	Qualified Temp. (°F)	EPU Peak Temp (°F)	EPU Evaluation Note
CA-98-037	Robertshaw Level Switch (DOR)	18	220	209.7	
CA-98-038	Rockbestos Coax Cable	14, 19	346	209.3	
CA-98-039	Rosemount 1153 Series A Transmitter (DOR)	33, 34	340	211.7	
CA-98-040	Rosemount 1153 Series B	14, 18, 19, 22, 27, 33, 34	318	211.7	
CA-98-041	Rosemount Conduit Seals	14, 18, 22, 27, 33	420	211.7	
CA-98-043	Rotork Valve Operators (50.49)	14, 31, 33	385	213	
CA-98-044	Static O-ring (DOR)	22, 33	212	211.7	
CA-98-046	Yarway Level (DOR)	14, 18	250	209.7	
CA-98-047	Samuel Moore Instrument Cable	13, 14, 18, 19, 20, 22, 27, 31, 32, 33, 34	340	219.8	
CA-98-049	Valcor Solenoid Valves	18, 19	365	209.7	
CA-98-050	DG O'Brien Electrical Penetrations	14, 19	320	209.3	
CA-98-050	DG O'Brien Electrical Penetrations (Plugs)	14, 19	345	209.3	
CA-98-051	Reliance Motors	14, 19	464	209.3	
CA-98-054	Consolidated Control Relays	18	223.1	209.7	
CA-98-060	Westinghouse Starter and Transformer	14, 19	260	209.3	
CA-98-064	Eaton Thermocouple Extension Cable	14	375	174.7	
CA-98-065	Brand Rex 600V Instrument Cable	13, 14, 18, 19, 20, 22, 27, 31, 32, 33, 34	385	219.8	

Table 3.4.4-1
Peak Reactor Building HELB Temperature Review

Calculation	Title	RB Volume	Qualified Temp. (°F)	EPU Peak Temp (°F)	EPU Evaluation Note
CA-98-066	Boston Control Cable	13, 14, 18, 19, 20, 22, 27, 31, 32, 33, 34	340	219.8	
CA-98-069	Patel Conduit Seals	18	354	209.7	
CA-98-071	EGS Grayboot Electrical Connectors	31	450	213	
CA-98-072	EGS Quick Disconnect	19, 31	435	213	
CA-98-073	Raychem/Swagelok Conduit Seals	14, 18, 19, 22, 33, 34	340	211.7	
CA-98-077	Eaton Cutler-Hammer Relays	22, 33, 34	232	211.7	
CA-98-079	ITT-Royal PVC Cable (DOR)	14, 18, 19, 22, 33	211	211.7	Note 3
CA-98-080	Okonite Control Cable	14, 18, 19, 22	211	209.7	
CA-98-081	Triangle Triolene Control Cable (DOR)	18, 19	211	209.7	
CA-98-082	MNGP-A Cable (DOR)	Various	230	211.7	
CA-98-083	MNGP-B Cable (DOR)	Various	211	209.7	
CA-98-084	Amphenol Connectors (DOR)	14, 18, 19, 22, 33	266	211.7	
CA-98-086	SOR Pressure Switches (50.49)	14, 22, 33	350	211.7	
CA-98-101	General Electric Terminal Blocks (50.49)	22, 33, 34	340	211.7	
CA-98-103	Patel P-1 Thread Sealant	Various	400	219.8	

Table 3.4.4-1
Peak Reactor Building HELB Temperature Review

Calculation	Title	RB Volume	Qualified Temp. (°F)	EPU Peak Temp (°F)	EPU Evaluation Note
CA-98-104	Rockbestos Firewall SR Cable	Various	351	219.8	
CA-98-107	Rockbestos Firewall III/SIS Cable	Various	342	219.8	
CA-98-108	Rockbestos Firewall EP Cable	Various	320	219.8	
CA-03-096	Loctite PST 580 Thread Sealant	Various	364	219.8	
CA-03-105	Scotch 130C and 69 Electrical Tape	Various	372	219.8	
CA-05-138	Cutler-Hammer Motor Starter/Control Transformer	14, 19	253	209.3	
CA-05-140	ASCO Scram Solenoid Pilot Valves	14, 18	277	209.7	

Notes

1) The ASCO pressure switches are located in Reactor Building Volumes 14, 19, and 31. The qualified peak temperature of 210°F bounds the postulated EPU HELB conditions for RB Volumes 14 and 19 having peak temperatures of 174.7°F and 209.3°F, respectively. Although qualified for peak temperature, the qualification of the pressure switches in RB Volume 19 do not bound the new EPU conditions and the recommended +15°F margin specified in IEEE Standard 323-1974. However, the ASCO pressure switch testing included the 210°F accident temperature for at least 10 hours total exposure (the combined duration of the dual transient test). The postulated peak HELB condition of 209.3°F exists briefly, and the event condition drops to less than 180°F within 200 seconds. Therefore, there is sufficient basis to justify adequate temperature margin for this equipment under EPU conditions. The ASCO pressure switches located in RB Volume 31 (PS-4666 and PS-4671) serve as the open indication devices for Containment isolation valves AO-2386 and AO-2387, respectively[5]. The switches serve only the Regulatory Guide 1.97 function of containment isolation valve position indication and are not required under a Reactor Building HELB (reference EC 12421 [116]). As such, the peak EPU HELB temperature of 213°F in this area has no relevance to function or qualification of this particular pressure switch.

[5] Loss of valve T-ring seal pressure indicates valve "open".

2) The Barton Model 580 series pressure switches are located in Reactor Building Volumes 14 and 18. The peak EPU HELB temperature of 174.7°F in Reactor Building Volume 14 is bounded by the qualified test level of 200°F, only those pressure switches in Reactor Building Volume 18 and exposed to the 209.8°F are not bounded.

   The pressure switches in RB Volume 18 are DPIS-2-129A/C serve, in part, for LPCI loop select logic by detecting reverse flow in the broken recirculation loop per General Electric Specification 22A1132 (Monticello specification MPS-0277 [121]) and Functional Control diagram NX-7905-6-1 [105]. Calculation CA-04-098 [96], Instrument Setpoint Calculation, Recirculation Riser Differential Pressure – LPCI Loop Select, further corroborates the LOCA only function for these switches and substantiates a 10-minute operating time for their function.

   Therefore this equipment has no required safety function during a HELB in RB volume 18. In addition, the current EQ file (CA-98-012 [9]) conservatively only uses the mild area qualification basis report from Barton for assessing the attained qualification levels for the switches. Additional testing from Barton on this switch model would easily demonstrate qualification for the 209.7°F postulated under EPU. Therefore, there is sufficient basis to justify adequate temperature margin for this equipment under EPU conditions.

3) ITT Royal cable is only installed at local instrument panels in Reactor Building Volumes 14, 18, 19, 22, and 33. The qualified test level of 211°F per EQ File CA-98-079 [61] bounds the peak EPU HELB temperatures of 174.7°F, 209.7°F, 209.3°F, and 184°F in RB Volumes 14, 18, 19, and 22, respectively. However, the ITT cable installed in RB Volume 33 exposed to the postulated EPU peak HELB temperature of 211.7°F. The cable is qualified in accordance with DOR Guidelines [115]. The ITT cable is routed through flex-conduit from the actual instrument to the local junction box. Using engineering judgment, it is reasonable to conclude that thermal lag considerations would assure that the cable temperature resulting from 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.

   To support the application of thermal lag effects, a review of the EPU HELB events which have peak temperature exceeding 211°F were reviewed. There are two EPU liquid break event cases having peak temperature conditions exceeding 211°F, both are related to RWCU breaks: RWCU-B-30-R1 (CA-96-175 [90]) at 211.5°F and the RWCU-B-28-R0 (CA-07-062 [88]) at 211.7°F. For each of these events, the peak HELB temperature occurs around 70 seconds and the profile drops below 210°F within 100 seconds. Therefore, there is sufficient basis to justify adequate temperature margin for this equipment under EPU conditions.

3.4.5 Turbine Building HELB Review

The EPU HELB evaluation identified areas within the Turbine Building that changed from a mild to harsh environments, as identified in Task T1009 HELB Subcompartment Evaluation. The areas listed below become harsh under EPU.

EQ Part B Volume	USAR App I Volume	Location
7	7	TB Sump and MCC B-31 Area
8	8	4 KV and Load Center Division A East
9	8	4 KV and Load Center Division A West
10	10	Hydrogen Seal Oil Unit and Condensate Pump Area North
10	11	Hydrogen Seal Oil Unit and Condensate Pump Area South
12	13	Condensate Backwash-Receiving Tank Area
15	16	Pipe Tunnel to Intake
19	20	Turbine Building Southeast Stairway from 911' to 931' El
21	27	TB Corridor Northwest 931' El
22	7	TB Northwest Stairway 931' to 951' El (Vestibule 931-911)
43	27	Hallway to No.11 Diesel Generator Entry Area

Walk downs of USAR Appendix I Volumes 7, 10, 11, 16, 20 and 27 were performed to identify safety related equipment in these new harsh areas that may fall under the scope of the EQ program. See AR 1131374-19 [117]. No discrete equipment that would fall under the scope of the EQ program was identified during the walk down in these volumes. However, some safety-related cabling was discovered that warranted further review for EQ Program inclusion.

Electrical cabling located within USAR Appendix I Volumes 7, 10, 11, 13, 16, 20 and 27 were evaluated utilizing plant drawings and the Cabling and Raceway database (CARIS). The evaluation found five (5) cable runs/routes that would be scoped as EQ equipment. They are all power cables, 3/C AWG# 6. CARIS only listed the mfr/model as Unspecified so field verification revealed these cables to be Rockbestos Firewall III, Rockbestos EP, and Brand Rex. The five (5) cables are as follows:

1B3431-A 6 AWG 3C Rockbestos T600V Firewall III XHHW NEC Type TC (UL)
1B3433-A 6 AWG 3C Rockbestos T600V Firewall III XHHW NEC Type TC (UL)
1B3434 6 AWG 3C Rockbestos T600V Firewall III XHHW NEC Type TC (UL)
B3347-P73A/1 Rockbestos Firewall EP 6AWG 600V Type RHH or RHW or USE
1B3472A Brand Rex 6AWG

Further investigation and contact with the vendor of these cables as documented in AR 1131374-19, indicated that the installed cables were of similar type to those already qualified for bounding

conditions present in either the Drywell or Reactor Building. Thus it can be reasonably concluded that the cables are also qualified. As indicated in the AR response, the following EQ follow-up actions to address the harsh Turbine Building areas were identified:

1. Revise EQ Qualification files 98-107 [72] (Rockbestos FW III) and 98-108 [73] (Rockbestos FW EP) to address Turbine Building configurations (EPU related). Action 01131374-43 was created to track this item.
2. Revise EQ Qualification files 98-065 [48] (Brand Rex) to address Turbine Building configurations (EPU related). Action 01131374-44 was created to track this item.

The EPU harsh environmental conditions and its effect on equipment located in Volumes 7 and 8 have been evaluated by Monticello in CAP 1125675 [118] Non-Conservative HELB Gothic Model on HELBs in the Condenser Room. The CAP was generated on 01/31/2008 when it was identified that temperatures in the lower 4KV area (Volume 8) would become harsh during postulated HELB cracks. An Operability Recommendation, OPR 1125675-01 was prepared and subsequently approved on 02/08/2008 which stated that the plant remained within its original licensing bases and that a safe shutdown path exists with the loss of the lower 4KV area as a result of a HELB crack of the Main Steam line or a HELB crack of the FW and Condensate line in the main condenser room.

In addition to the cable identified above, there is a single Valcor solenoid valve in the EQ program located in USAR Appendix I Volume 21[6]. The solenoid valve remains qualified for the postulated conditions in the TB under EPU. The qualification level of this solenoid valve and that of the solenoid valve in the redundant train of the same Valcor design located in Reactor Building satisfies both locations as the conditions of the Reactor Building location are bounding.

3.4.6 Reactor Building Post-LOCA Heat-up Evaluation

Under EPU, the post-LOCA heat-up (PLHU) ambient temperature conditions of certain areas in the Reactor Building are calculated to increase (CA-08-085 [98]). For a visual comparison between pre and post EPU on Reactor Building post-LOCA heat-up conditions, a composite curve of the entire PLHU effects of each Reactor Building volume was created. This is shown in Figure 3.4.6-1.

[6] USAR Turbine Building Volume 21 matches EQ Program TB Volume 25 until volume designator discrepancy is resolved under CAP 01115107 [123].

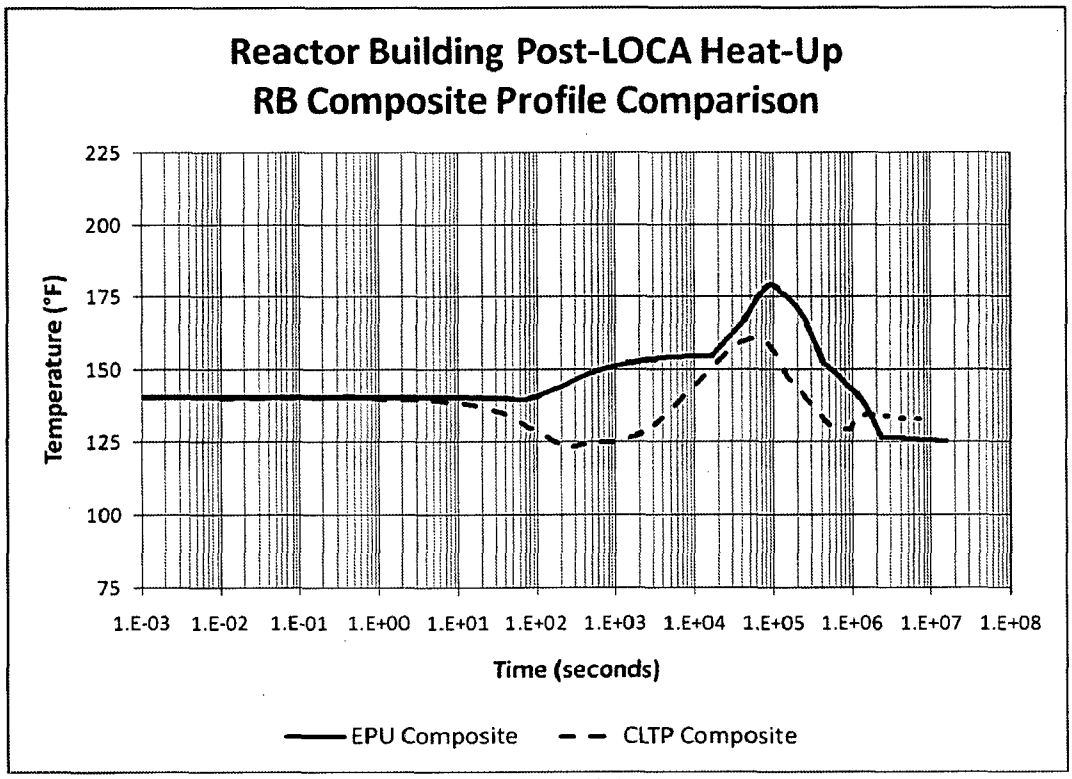

Figure 3.4.6-1

A comparison between the CLTP and EPU temperature conditions in the Reactor Building following a LOCA is shown in Table 3.4.6-1.

Table 3.4.6-1
Reactor Building Peak Post-LOCA Temperature Comparison

RB Volume	CLTP PLHU Temp (°F)	EPU PLHU Temp (°F)	RB Volume	CLTP PLHU Temp (°F)	EPU PLHU Temp (°F)
1	142.97	109.7	25	113.9	119
2	142.97	111.4	26	109.1	118.1
3	143.8	128.6	27	109.1	121.2
4	143.8	127.6	28	120	124.6
5	109.9	112.6	29	120.1	124
6	140.4	140.3	30	120.2	129.8
7	116.3	123.9	31	120.2	131.9
8	125	110.1	32	120.2	125.4
9	160.4	179	33	115.6	121.2
10	159.9	179.1	34	107.9	119.2
11	158.7	179	35	107.8	119.2
12	157.3	179	36	134.4	124.2
13	112.6	118.3	37	112.8	124.2

Table 3.4.6-1
Reactor Building Peak Post-LOCA Temperature Comparison

RB Volume	CLTP PLHU Temp (°F)	EPU PLHU Temp (°F)		RB Volume	CLTP PLHU Temp (°F)	EPU PLHU Temp (°F)
14	107.5	115.6		38	100	100
15	106.3	122.5		39	126.3	132.6
16	135.2	154.4		40	105.6	116.5
17	108.1	115.7		41	106.9	117.2
18	109.3	121.3		42	104.6	119.2
19	108.3	121.3		43	109.7	114.3
20	112.5	119.9		44	108.1	117.8
21	112.2	118.4		45	104	115.1
22	106.7	119		46	107.2	116.1
23	105.4	118.2		47	100	118.7
24	104	104		48	103.9	115.3

The impact of the EPU increase in PLHU temperature conditions shown in Table 3.4.6-1 on the MNGP EQ equipment is evaluated in Table 3.4.6-2. Common equipment or commodity items qualified for the Drywell were generally assessed at 163°F after 5-days post-accident. These common equipment items, for which qualification is demonstrated to these levels, will also be qualified for Reactor Building locations under PLHU conditions under EPU as well. In some cases, additional analysis is provided in Attachment A to ensure the equipment remains qualified for PLHU conditions calculated under EPU.

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-98-003	Allen Bradley Terminal Boards	These terminal blocks are only located in Reactor Building Volume 37. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A1).
CA-98-004	ASCO Solenoid Valves (Normally Energized)	The CLTP analysis of this equipment addressed four (4) groups by model and test relevance. Three (3) of the groups were analyzed for PLHU operation by using the first five (5) days of bounding test conditions and extrapolating the last stage of testing to an equivalent duration at 160.4°F. These solenoids all de-energize post-accident so there is no need to include temperature rise effects. Under EPU, all of the RB area PLHU temperature conditions

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		are below 106°F at 5-day post-LOCA. The CLTP analysis for these solenoids remains bounding. The remaining solenoid group is only located in RB Volume 9. The CLTP analysis for this group equated later portions of the relevant testing to a plant temperature of 190°F. Under EPU, the peak RB PLHU temperature is less than 180°F at all times. As such, the CLTP analysis remains bounding.
CA-98-005	ASCO Solenoid Valves (Normally De-Energized)	The CLTP analysis of this equipment addressed two (2) groups by model and test relevance. The first group was further subdivided into three (3) sub-set analyses for PLHU evaluations, two energized with equipment temperature rise considerations and the third de-energized sub-group. <u>First model group (three sub-groups):</u> The PLHU analysis of the first energized sub-group within this model/test group are for solenoid valves in the RHR rooms (RB Volumes 1 and 3) using a continuous PLHU ambient temperature of 145°F. The PLHU analysis of the second sub-group are for energized solenoid valves located in the Drywell using an assumed continuous ambient of 163°F after five days. The remaining third sub-group PLHU analysis was for de-energized valves post-accident located RB Volumes 9-12, and 31. The CLTP evaluation states that the energized PLHU evaluation for the Drywell solenoid valves bound the conditions for PLHU after the five day point. For EPU, the RHR room PLHU temperatures are less than 145°F and the Drywell as well as all of the RB PLHU temperatures at 5-days and beyond remains below 163°F. As such, the CLTP analysis for the first solenoid group remains bounding for EPU conditions. For the second model group:

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		The solenoid valves in this group remain de-energized post-accident. The CLTP analysis for this group assumed a continuous PLHU temperature of 160.4°F after five days and extrapolated test conditions beyond that point to the assumed ambient temperature. For EPU, the PLHU temperatures for all RB areas after 5-days are below 160°F. As such, the CLTP analysis remains bounding for EPU conditions.
CA-98-006	ASCO Pressure Switches	The equipment is located in RB Volumes 1, 3, 9, 11, 12, 14, 19, and 31. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A2).
CA-98-007	ASCO Temperature Switches	The equipment is only located in RB Volumes 36 and 39 serving SGTS operation. The current EQ analysis evaluated to the CLTP PLHU temperature of 134.4°F in RB Volume 36. Due to normal ambient temperature changes under CLTP operation, the qualified life is reduced as documented in the Corrective Action Program (CAP) 01106163 [122]. Although the PLHU temperature is reduced from CLTP values in RB Volumes 36/39, an evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified within the reduced qualified life established in the CAP (see Attachment A22).
CA-98-008	Automatic Valve Solenoid Valves	Equipment located in the Drywell only (MSIV), no RB PLHU evaluation needed.
CA-98-010	Barksdale Pressure Switch	Equipment has a 10-hour operating time. No long-term PLHU evaluation needed.
CA-98-011	Barton Pressure Switches	The equipment is located in RB Volumes 1, 3, 14, and 19. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A3).

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-98-012	Barton Pressure Switches 580A-0, 580A-1	Equipment has short operating time (<10 hours) or function to mitigate a HELB only. No long term RB PLHU evaluation needed.
CA-98-014	E.F. Johnson Banana Plug	The equipment is part of the electrical circuit associated with the Patel Temperature switches (CA-98-078) for HELB mitigation only. No RB PLHU evaluation required.
CA-98-017	G.E. Cable (butyl rubber SI-58007/58136)	Cables qualified for long term use at 189.2°F per CLTP analysis. This was based on assuming a worst-case cable temperature rise of 16°C onto the worst-case PLHU temperature of 160.4°F. The peak PLHU temperature occurs in Torus compartment. The EQ equipment located in this area are only control, instrument, or valve actuator devices that would not cause cable temperature rise effects. Therefore, the CLTP analysis was conservative in addressing temperature rise effects onto peak Torus PLHU temperature conditions. However, an evaluation of this cable type for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A4).
CA-98-017	GE Cable (XLPE type, SI-57275/58109)	GE cable type SI-57275 qualified for inside and outside the Drywell. The CLTP analysis assesses worst-case PLHU of the Reactor Building with rise as bounding condition at 189.2°F. This was based on assuming a worst-case cable temperature rise of 16°C onto the worst-case PLHU temperature of 160.4°F. The peak PLHU temperature occurs in Torus compartment. The EQ equipment located in this area are only control, instrument, or valve actuator devices that would not cause cable temperature rise effects. Therefore, the CLTP analysis was conservative in addressing temperature rise effects onto peak Torus PLHU temperature conditions. However, an evaluation of this cable type

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A4). GE cable type SI-58109 only qualified for limited Drywell applications having short operating times. As such, there is no further need for an RB PLHU evaluation for EPU conditions.
CA-98-017	GE Cable (PE type SI-58081)	The cables are used throughout the Reactor Building, exclusive of the Steam Chase and RWCU rooms (RB Volumes 16 and 28 to 32). An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A4).
CA-98-018	General Electric Motors	The motors are only located in the RHR room, RB Volumes 1 and 3. The current analysis assessed post-accident operation at an ambient temperature of 143.8°F (plus motor rise). The worst-case specified RHR PLHU temperature is 128.6°F (RB Volume 3). As such, the current analysis remains bounding.
CA-98-020	General Electric Containment Penetrations	Qualified for Drywell conditions, bounds RB PLHU conditions.
CA-98-021	General Electric Terminal Blocks	These terminal blocks may be located throughout the Reactor Building. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A5).
CA-98-022	General Electric MCCs	The equipment principally serves to mitigate HELB; the LOCA duration for HPCI system MCC is less than 10 hours. PLHU under EPU for HPCI room (110.1°F) is not calculated to increase beyond the current room temperature of 125°F. As such, no further PLHU evaluation is needed.
CA-98-023	Hevi-Duty Electric Transformer (DOR)	This equipment is located in Reactor Building Volume 37 only. An evaluation of the equipment for PLHU conditions

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		under EPU concludes the equipment remains qualified (see Attachment A6).
CA-98-024	General Electric Fan Motors (DOR)	This equipment is located in Reactor Building Volume 37 only. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A7).
CA-98-025	Limitorque MOV (DOR, class H motor, MO-2397 only)	Equipment located in the Drywell, no RB PLHU evaluation needed.
CA-98-025	Limitorque MOV (DOR, class B motor, MO-2107 only)	RCIC system injection valve, no LOCA function and no RB PLHU evaluation needed.
CA-98-026	Limitorque MOVs (class RH, AC motor)	Qualified for long-term bounding Drywell conditions. The CLTP analysis for the class RH AC motor also includes miscellaneous subcomponents (melamine and Fibrite limit and torque switches and applicable terminal blocks) associated with Limitorque Reports 600376A and B0212.
CA-98-026	Limitorque MOVs (class RH, DC motor)	The CLTP analysis focuses on the short operating time of the class RH, DC motors qualified for under 10-hours using Limitorque Report B0009. The remaining components within the actuator are qualified via other Limitorque reports for long-term bounding Drywell conditions.
CA-98-026	Limitorque MOVs (class B motor)	The CLTP analysis only applies to two (2) actuators with containment isolation functions in RB Volume 12 using Limitorque Report B0003. These actuators have less than a 10-hour operating time. The testing remains bounding for EPU conditions throughout this duration.
CA-98-027	Magnetrol Level Switches	The CLTP analysis bounds the short operating time (8-hours or less) using bounding testing of >280°F. The components are only located in RB Volume 12. The peak PLHU temperature in this area under EPU is less than 180°F. The CLTP analysis remains bounding.
CA-98-028	McDonnell & Miller Flow Switches (DOR)	The equipment is only located in RB Volume 37. The CLTP analysis for this

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		DOR component utilizes the CLTP analysis provided by CA-98-030 for the internal microswitch component which was analyzed at 134.4°F for PLHU conditions of RB Volume 36. For the remaining critical age-sensitive material (Viton gasket) of the flow switch, the CLTP analysis provides for a significant life (>1,000 years) at CLTP PLHU temperatures of 112.8°F for RB Volume 37. For EPU, the PLHU temperature in RB Volume 37 is 124.2°F. Given the life margin of the gasket and the CLTP bounding temperature analysis for the microswitch, the CLTP analysis remains bounding for EPU.
CA-98-030	MicroSwitch Limit Switches (DOR)	The equipment is only located in RB Volume 36, 37, and 39. The CLTP analysis addressed PLHU at a bounding temperature of 134.4°F. Under EPU, the PLHU temperature of the affected volumes is 132.6°F. However, the normal ambient temperature increased in this area as reported under the Corrective Action Program (CAP) 01106163 [122], an evaluation of the equipment for PLHU conditions under EPU confirms that the equipment remains qualified (see Attachment A23).
CA-98-032	Namco Limit Switches (EA740 w/o EC210)	Qualified for long-term bounding Drywell conditions.
CA-98-032	Namco Limit Switches (EA740 & EC210)	This equipment is only located in the Steam Chase (RB Volume 16). The CLTP analysis extrapolated testing beyond 20 days to an equivalent duration at 135°F to bound PLHU operation. Accident testing of the limit switches within the first 20 days was 205°F or greater. Under EPU, the PLHU temperature profile for RB Volume 16 peaks at 154.4°F, dropping below 135°F within 8-days post-LOCA. As such, the CLTP analysis remains bounding.

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-98-032	Namco Limit Switches (EA180 & EC210)	Qualified for long-term bounding Drywell conditions.
CA-98-032	Namco Limit Switches (EA180 w/o EC210)	The equipment may be used throughout the Reactor Building. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A8).
CA-98-033	Namco Quick Disconnects EC210	Qualified for long-term bounding Drywell conditions.
CA-98-035	Raychem NEIS Seals	Serves as HELB environmental seal. Under LOCA operation, the Reactor Building is not steam harsh and device sealing is not needed, thus no RB PLHU review required for this equipment.
CA-98-036	Raychem WCSF-N Splices	Qualified for long-term bounding Drywell conditions.
CA-98-037	Robertshaw Level Switch (DOR)	Equipment located in RB Volume 18 only, supporting short term operation of the HPCI system (<10-hours). Testing of the equipment was at least 10-hours and greater than 150°F (220°F peak). The EPU temperature for PLHU in RB Volume 18 is 121.3°F. As such, the CLTP testing and analysis remains bounding for EPU.
CA-98-038	Rockbestos Coax Cable	Qualified for long-term bounding Drywell conditions.
CA-98-039	Rosemount 1153 Series A (DOR)	These transmitters are located in RB Volumes 9 and 12 (LT-7338A/B) or 33 and 34 (PT-77251A/B). The transmitters in the Torus compartment (RB Volumes 9 and 12) have lower performance issues under EPU radiation doses and will also be exposed to the peak PLHU temperatures of 179°F, ~20°F higher than CLTP conditions predict. The CLTP analysis for PLHU qualification utilizes thermal aging to address the long-term operability of the transmitters for 6 days at 160.4°F followed by 192 days at 130°F. The postulated peak PLHU temperature conditions for RB Volumes 33 and 34 is 121.2°F. As such, the CLTP analysis remains bounding for

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		the transmitters located in RB Volumes 33 and 34. Transmitters LT-7338A/B are being replaced with qualified transmitters under EC 13086 [133].
CA-98-040	Rosemount 1153 Series B	The equipment may be used throughout the Reactor Building. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A9).
CA-98-041	Rosemount Conduit Seals	The equipment may be used throughout the Reactor Building. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A10).
CA-98-042	Rotork "A" Range Actuators (DOR)	This equipment has a 1-hour operating time. The testing and qualification temperature of 163°F for the limiting component (Helix) is not reached under EPU PLHU conditions until well after 1-hour. As such, the CLTP analysis remains bounding.
CA-98-043	Rotork Valve Operators (50.49)	The CLTP analysis uses the first 10-days of testing to bound plant conditions. Beyond 10-days, the remaining 20 days and 17 hours of testing (30 day + 17 hour test) was extrapolated to an equivalent duration at 145°F for PLHU operability. For EPU, all RB PLHU temperature conditions are less than 145°F beyond the 10-day point. As such, the CLTP analysis remains bounding.
CA-98-044	Static O-ring (DOR)	The equipment is located in RB Volumes 1, 3, 22, and 33. The CLTP analysis evaluates PLHU conditions at 145°F. Under EPU, the peak PLHU temperature for the affected volumes is 128.6°F. As such, the CLTP analysis remains bounding for EPU.
CA-98-046	Yarway Level (DOR)	The equipment is located in RB Volumes 14 and 18. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A11).

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-98-047	Samuel Moore Instrument Cable	The current EQ analysis equates the last 26 days of testing at 200°F to an equivalent duration at 160°F to bound all RB locations after 4-days post-LOCA. The EPU PLHU data for all areas of the Reactor Building at 4-days post-LOCA is at 159.4°F or less. As such, the current EQ analysis remains bounding for EPU.
CA-98-049	Valcor Solenoid Valves	The equipment is located in RB Volumes 11, 18, and 19. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A12).
CA-98-050	DG O'Brien Electrical Penetrations	Qualified for long-term bounding Drywell conditions.
CA-98-050	DG O'Brien Electrical Penetrations (Plugs)	Qualified for long-term bounding Drywell conditions.
CA-98-051	Reliance Motors	Equipment located in RB Volumes 1, 3, 14, and 19 only. The CLTP analysis evaluates PLHU conditions at 145°F. The peak PLHU temperature under EPU is only 128.6°F. As such, the CLTP analysis bounds EPU conditions.
CA-98-052	Tavis Flow Transmitter	The equipment is located in RB Volume 37. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A13).
CA-98-053	ITT Grinnel/Conoflow Transducer	The equipment is located in RB Volume 37. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A14).
CA-98-054	Consolidated Control Relays	Equipment has less than 1-hour operating time and only located in RB Volume 18. The CLTP analysis based on minimal thermal aging of 185°F remains bounding for service life and post-accident operation.
CA-98-055	General Atomic Radiation Detector	Drywell location only, no RB PLHU evaluation needed.
CA-98-059	Kerite Cable/Termination	The cables and splices are only exposed to the PLHU conditions of RB Volumes 1, 3,

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		5, and 6. The CLTP analysis evaluated PLHU operation at 144°F. Under EPU, the PLHU temperatures are less than 144°F. As such, the CLTP analysis remains bounding.
CA-98-060	Westinghouse Starter and Transformer	Equipment only located in RB Volumes 14 and 19. The only component remaining from the originally supplied motor starter is the fuse and fuse block. Other portions qualified by CA-05-138. The CLTP analysis of the fuse/block is based on Westinghouse proprietary testing at 260°F. The CLTP analysis remains bounding for EPU.
CA-98-062	Gould Contactor/Disconnect	The equipment is located in RB Volume 37. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A15).
CA-98-064	Eaton Thermocouple Extension Cable	The current analysis assesses post-accident operation at an assumed steady state temperature of 163°F at and after 5 days. The EPU PLHU data for all areas of the Reactor Building at 5 days and after is less than 155°F. As such, the current analysis remains bounding for EPU.
CA-98-065	Brand Rex 600V Instrument Cable	The current analysis assesses post-accident operation at an assumed steady state temperature of 163°F at and after 5 days. The EPU PLHU data for all areas of the Reactor Building at 5 days and after is less than 155°F. As such, the current analysis remains bounding for EPU.
CA-98-066	Boston Control Cable	The current analysis includes testing that ran for >360 days. The first 100 days of testing was at 200°F or more, while the remaining test temperature was at least 160°F. This testing remains bounding for the PLHU conditions postulated under EPU.
CA-98-067	CONAX Electrical Connector Seal	Equipment located in the Drywell only, no RB PLHU evaluation needed.

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-98-068	CONAX RTDs	Equipment only located in the Torus compartment (RB Volumes 9 to 12). The CLTP analysis assesses post-accident operation at 193°F. Under EPU, the PLHU temperature is less than 180°F. As such, the CLTP remains bounding for EPU.
CA-98-069	Patel Conduit Seals (and flex conduit)	Serves as HELB environmental seal, no RB PLHU required.
CA-98-070	Patel Conformal Coating	Serves as HELB environmental seal, no RB PLHU evaluation required.
CA-98-071	EGS Grayboot Electrical Connectors	Qualified for long-term bounding Drywell conditions.
CA-98-072	EGS Quick Disconnect	Qualified for long-term bounding Drywell conditions.
CA-98-073	Raychem/Swagelok Conduit Seals	Serves as HELB environmental seal, no RB PLHU evaluation required.
CA-98-075	Weed Thermocouples	Equipment located in the Drywell only, no RB PLHU evaluation needed.
CA-98-076	Rome Cable Type SIS (DOR)	This cable is associated with GE MCC with LOCA operating time (HPCI system MCC) of less than 10 hours. The HPCI room PLHU temperature under EPU is not calculated to exceed the currently specified 125°F ambient temperature.
CA-98-077	Eaton Cutler-Hammer Relays	This equipment is located in RB Volumes 22, 33, 34. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A16).
CA-98-078	PEI/FENWAL Temperature Switch	HELB detection only, no RB PLHU evaluation needed.
CA-98-079	ITT Royal PVC Cable (DOR)	The CLTP analysis evaluates PLHU for RB Volumes 5, 14, 18, 19, and 22. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A17).
CA-98-080	Okonite Control Cable	The CLTP analysis evaluates PLHU for RB Volumes 5, 14, 18, 19, and 22. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
		Attachment A18).
CA-98-081	Triangle Triolene Control Cable (DOR)	Cable is only used in RB Volumes 18 and 19 serving one solenoid valve (SV-3269A). The operating time of the solenoid is to de-energize within 1-hour for containment isolation. As such, no PLHU evaluation needed for this cable.
CA-98-083	MNGP-B Cable (DOR)	Cables serve limited applications in the Reactor Building. The current analysis only shows cables in RB Volumes 1, 3, 14, 19, and 37 have long-term functions. The current analysis assumes at least 145°F over the PLHU period. As such, the current analysis bounds the PLHU conditions in these areas postulated under EPU, no further review required.
CA-98-084	Amphenol Connectors (DOR)	Equipment only located in RB Volumes 1, 3, 14, 18, 19, 22, and 33. The CLTP analysis evaluates PLHU operation at 145°F. Under EPU, the PLHU temperature of the affected areas is less than 145°F. As such, the CLTP analysis remains bounding.
CA-98-085	Pyco Temperature Elements	HELB detection only, no RB PLHU evaluation needed.
CA-98-086	SOR Pressure Switches (50.49)	Equipment located in RB Volumes 14, 22, and 33 only. Switches located in RB Volume 14 serve HELB functions only. The CLTP analysis assesses PLHU operation at 120°F plus a very conservative +10°F rise. Under EPU, the PLHU temperature of the affected areas is 121.2°F. The CLTP analysis is judged to have sufficient margin (with +10°F rise) to account for the 1.2°F difference and remain bounding for EPU.
CA-98-101	General Electric Terminal Blocks (50.49)	The terminal blocks are only located in RB Volumes 22, 33, and 34. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A19).
CA-98-103	Patel P-1 Thread Sealant	Serves as HELB environmental seal, no RB PLHU evaluation required.

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-98-104	Rockbestos Firewall SR Cable	Qualified for long-term bounding Drywell conditions.
CA-98-107	Rockbestos Firewall III/SIS Cable	Qualified for long-term bounding Drywell conditions.
CA-98-108	Rockbestos Firewall EP Cable	The current EQ analysis evaluates post-accident operation at an assumed steady state temperature of 189°F. This was based on assuming a worst-case cable temperature rise of 16°C onto the PLHU temperature of 160°F (after 1.00E+05 seconds of CLTP conditions). The bounding PLHU temperature condition occurs in Torus compartment. The EQ equipment located in this area are only control, instrument, or valve actuator devices that would not cause cable temperature rise effects. Therefore, the CLTP analysis was conservative in addressing temperature rise effects onto Torus PLHU temperature conditions. The CLTP analysis at 189°F is bounding for the EPU postulated PLHU temperature conditions in the Reactor Building.
CA-98-109	Valcor MSIV Solenoid Valves	LOCA/MSLB isolation, short operating time (1 hr), no RB PLHU evaluation needed
CA-98-128	UCI Electrical Tape Terminations	For medium voltage splice, the equipment is only located in RHR pump rooms RB (Volumes 1 and 3). The CLTP analysis for this application remains bounding for EPU. For general lower voltage use (600 Volt or less), the splices may be installed throughout the Reactor Building. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A20).
CA-02-197	Dow Corning 3-6548 Silicone RTV Foam	Associated with the GE MCCs only in RB Volumes 5 (RCIC Room) and 8 (HPCI Room) for precluding gross moisture ingress into MCC during a HELB only, no RB PLHU evaluation required.

Table 3.4.6-2
Reactor Building Post-LOCA Temperature Review

EQ File	Equipment Type/Description	Comment
CA-03-096	Loctite PST 580 Thread Sealant	Serves as HELB environmental seal, no RB PLHU evaluation required.
CA-03-105	Scotch 130C and 69 Electrical Tape Splices	The equipment may be installed throughout the Reactor Building. An evaluation of the equipment for PLHU conditions under EPU concludes the equipment remains qualified (see Attachment A21).
CA-05-137	Fisher E/P Transducer	Equipment located in the RHR pump rooms only (RHR room stairways, RB Volumes 2 and 4), current analysis bounds PLHU under EPU.
CA-05-138	Cutler-Hammer Motor Starter/Control Transformer	Equipment located in RB Volumes 14 and 19 only. The CLTP analysis is based on post-accident testing equivalent of 1 year at 122°F. Under EPU, the PLHU temperature of the affected volumes is 121.3°F. As such, the CLTP test basis remains bounding for EPU conditions.
CA-05-140	ASCO Scram Solenoid Pilot Valves	Under LOCA, the equipment de-energizes to support scram function. As such, no PLHU evaluation needed.

3.4.7 Normal and Accident Radiation Evaluation

The environmental qualification radiation analyses are based on the total combined normal and accident doses. In a few cases, the normal dose has been reduced for specific equipment items to correctly match the anticipated dose over its shorter qualified lifetime. While in other cases, equipment specific accident doses were determined which included reduction factors for shorter operating times and/or distance shielding. Calculation CA-08-067 [97] provides 60-year normal dose for both CLTP and EPU conditions while CA-08-145 [99] provides the EPU accident doses. Under EPU, the normal plant doses are generally increased by 13% over CLTP doses while some steam line containing areas also experience increased doses during shut-down due to moisture carry-over issue related to EPU. The accident dose calculation determined increases ranging from 2.5% to 8.3% for EPU over CLTP conditions.

For EPU, the accident radiation doses throughout the Reactor Building are now based on de-pressurized water which is in accordance with source term guidance given in NUREG-0737 Item II.B.2. This resulted in a small reduction in the accident dose for the RHR rooms. The safety related functions of the equipment in the RHR rooms are for supporting low pressure core injection (LPCI), containment cooling (Drywell/Torus spray and/or suppression pool cooling) and reactor core spray (CS). For all these functions and for EOP sequences, de-pressurized

Torus water is injected/re-circulated post-accident. Thus, it is deemed appropriate to use de-pressurized reactor water source terms for assessing EPU accident doses for the RHR rooms. Accordingly, there is a corresponding decrease between CLTP and EPU doses for the equipment located in the RHR rooms.

To facilitate the impact of increased normal and accident doses under EPU, and to still maintain consistency with the CLTP methodology of total dose computations, a detailed evaluation for EPU impact is made for all EQ equipment. For conservatism, the evaluation for radiation conditions generally considered the EPU accident dose to be 1.083 times the CLTP value, except for the RHR rooms. This evaluation is provided in Table 3.4.7-1. The margin column indicates the level of compliance with the recommended margin of IEEE Standard 323-1974, that is +10% on accident dose. The margin in the following analysis is computed as follows (margins greater than 100%, are simply shown as ">100%"):

$$\text{margin} = \frac{\text{Qualified dose - EPU TID}}{\text{EPU accident dose}} \times 100\%$$

The Beta dose specified for the Drywell in the Monticello EQ Program is taken from the DOR Guidelines [115] as an unshielded 200 Mrad dose which was developed for a 4,100 Mwth reactor[7]. Therefore, there is significant Beta dose margin included in the Beta dose as it applies to Monticello. No increase in the Drywell Beta dose for EPU or additional margin is addressed in this evaluation for the Drywell accident Beta dose. The Beta dose only is applied to the EQ components in the Drywell. In a few case, the device sealing precluded the need to address Beta dose, and the equipment is considered shielded. In most cases, though, a conservatively reduced Beta dose was considered as computed in the CLTP EQ file analyses. However, for this evaluation when appropriate (cable and devices at the Drywell wall), a 50% reduction in Beta dose was considered in this analysis. This reduction is based on the reduction permitted in Item 9 of Section 1.4 of NUREG-0588 [119].

[7] Per Appendix D of NUREG-0588 [119] as referenced in the DOR Guidelines.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-003	Allen Bradley Terminal Boards	6.31E+02	6.60E+02	4.00E+06	4.33E+06		4.33E+06	2.51E+08	>100%	RB Vol 37 only, 10 ft dose from SGTS filter
CA-98-004	ASCO Solenoid Valves (Normally Energized) Group 1	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	2.00E+08	>100%	RB Volume 1 worst-case among grouping. No valves change state at >20 Mrads exposure
CA-98-004	ASCO Solenoid Valves (Normally Energized) Group 2	1.01E+05	1.06E+05	1.15E+07	1.25E+07		1.26E+07	2.00E+08	>100%	RB Volume 9 only. TID remains below 20 Mrad.
CA-98-004	ASCO Solenoid Valves (Normally Energized) Group 3	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+07	9.63E+07	2.00E+08	>100%	Drywell only, Beta shielded dose. Valves have EPDM elastomers, no issue with shifting after 20 Mrad.
CA-98-004	ASCO Solenoid Valves (Normally Energized) Group 4	3.79E+05	3.96E+05	2.34E+06	2.53E+06		2.93E+06	2.00E+08	>100%	RB Vol. 18 only, TID less than 20 Mrad for Viton elastomers shift issue.
CA-98-005	ASCO Solenoid Valves (Normally De-energized) Group 1	1.79E+07	1.88E+07	5.31E+07	5.75E+07	(shielded)	7.63E+07	2.00E+08	>100%	Drywell worst-case among grouping. Internals shielded from Beta dose.
CA-98-005	ASCO Solenoid Valves (Normally De-energized) Group 2	1.01E+05	1.06E+05	1.15E+07	1.25E+07		1.26E+07	2.00E+08	>100%	RB Vol. 9 worst-case TID among grouping. TID still less than 20 Mrads shift concern for Viton elastomers.
CA-98-006	ASCO Pressure Switches	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	1.93E+07	54.0%	RB Vol 12 worst-case TID for equipment considering 10 hour operating time for those in RB Volumes 1 and 3.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-007	ASCO Temperature Switches	6.31E+02	6.60E+02	1.00E+07	1.08E+07		1.08E+07	1.50E+07	38.5%	RB Vols. 36 and 39 only, accident dose at 5-foot distance from SGTS filter.
CA-98-008	Automatic Valve Solenoid Valves		3.67E+06		3.74E+06	(Shielded)	7.41E+06	7.79E+06	10.0%	Drywell only (MSIV SOVs). 30 minute EPU accident dose. EPU normal dose is difference of qualified dose and EPU accident with 10% margin. At 43.32 R/hour for EPU Drywell conditions (CA-08-067 [97]), this yields a radiation life of 9.6 years exceeding the thermal life of 6.64 years. MSIV SOVs changed every 6 years per PMID 00009221, item 03. CLTP values not shown due to different methods of analysis.
CA-98-010	Barksdale Pressure Switch	1.26E+05	1.32E+05	6.00E+06	4.00E+06		4.13E+06	1.00E+07	>100%	RB Vol. 1 has worst-case TID (10-hour accident dose) of installed locations.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-011	Barton Pressure Switches (RHR)	1.26E+05	1.32E+05		4.47E+05		5.79E+05	1.00E+06	94%	RB Vol. 1/3 switches have 1.0E+6 rad limitation due to fill fluid but only a LPCI critical function for 10-minutes. 18-minute EPU accident dose shown, CLTP accident dose not shown due to change in dose source basis for RHR room.
CA-98-011	Barton Pressure Switches (Non RHR)	1.01E+05	1.06E+05	2.34E+06	2.53E+06		2.64E+06	3.00E+06	14.2%	RB Vols. 14 & 19 only, RB Vol. 14 worst-case TID.
CA-98-012	Barton Pressure Switches 580A-0, 580A-1	3.79E+05	3.96E+05	2.34E+06	2.53E+06		2.93E+06	1.00E+07	>100%	RB Vols. 14 and 18 only. RB Vol. 18 worst-case TID.
CA-98-014	E.F. Johnson Banana Plug	2.97E+06	2.97E+06				2.97E+06	4.70E+06		No accident dose, only needed for HELB. RB Vol. 16 worst-case normal dose.
CA-98-017	G.E. Butyl Cables	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	4.00E+07	90.9%	Cables located throughout RB excluding RB Vols 16 and 28 to 32. RB Vol. 1 worst-case TID (with cables >4 foot from SGTS filter).

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-017	G.E. Cable SIS & SI-58109 Control	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+07	9.63E+07	2.00E+08	>100%	Cables located throughout plant, Drywell worst-case. Current EQ file Beta analysis utilizes sacrificial shielding layer that should have a confirmation of potential IR effects on instruments. However, crediting a 50% reduction in Beta (i.e. 1.0E+08 Rad) due to localized shielding and without sacrificing insulation would negate need for analysis as test dose would bound a revise TID of 1.96E+08 rad. Margin N/A for this DOR cable.
CA-98-017	G.E. Cable SI-58081 Control	6.31E+02	6.60E+02	2.00E+07	2.17E+07		2.17E+07	2.44E+07	12.6%	Cables located outside Drywell and RB Vols. 16, and 28 to 32. EQ file justifies greater than 3-foot distance from SGST filter. RB Vols. 36-39 thus worst-case (at >3 foot from SGTS filter).

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-018	General Electric Motors	1.26E+05	1.32E+05	2.62E+07	2.09E+07		2.10E+07	3.00E+07	43.1%	Located in RHR rooms only (RB Vols. 1 and 3).
CA-98-020	General Electric Containment Penetrations	1.79E+07	1.88E+07	5.31E+07	5.75E+07	(shielded)	7.63E+07	1.00E+08	41.2%	Drywell end of penetration worst-case. Equipment addressed by CA-98-020 is all Beta shielded.
CA-98-021	General Electric Terminal Blocks	6.31E+02	6.60E+02	1.87E+08	2.03E+08		2.03E+08	2.20E+08	8.6%	All RB areas except RB Vols. 16. RB Vols. 36 to 39 worst-case TID. Margin NA, DOR.
CA-98-022	General Electric MCCs	4.45E+04	7.57E+04				7.57E+04	1.00E+06		RB Vols 5 & 8 only, no accident dose. RB Vol 5 bounding under EPU. CLTP EQ file normal dose from RB Vol. 8.
CA-98-023	Hevi-Duty Electric Transformer	6.31E+02	6.60E+02	<2.00E6	<2.00E+06		2.00E+06	2.00E+06	NA	RB Volume 37 only. CLTP accident dose based on 9-foot distance from SGTS filter and reduction for 6" concrete intervening wall. Margin NA, for this DOR equipment.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-024	General Electric Fan Motors	6.31E+02	6.60E+02	1.00E+06	1.08E+06		1.08E+06	1.00E+06		RB Vol. 37 only. Accident doses based on 20-foot distance from SGTS filter. Fan motors are actually >20-foot distance. Further, qualified dose based on perceived threshold value. Thus, EPU dose are justified through re-analysis. Margin N/A for this DOR equipment.
CA-98-025	Limitorque Motor Operators (DOR) (MO-2397)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+06	7.83E+07	2.00E+08	>100%	Drywell only, Beta dose shielding considerations.
CA-98-025	Limitorque Motor Operators (DOR) (MO-2107)	2.97E+06	2.97E+06	2.54E+06	2.75E+06		5.72E+06	2.00E+07	>100%	RB Vol 16 only.
CA-98-026	Limitorque Motor Operators (Class RH AC motor)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+07	9.63E+07	2.00E+08	>100%	Drywell worst-case among grouping with Beta dose shielding considerations.
CA-98-026	Limitorque Motor Operators (Class RH DC motor)	2.97E+06	2.97E+06	2.54E+06	2.75E+06		5.72E+06	1.00E+07	>100%	RB Volumes 8, 9, 10, 13, and 16 only. RB Vol. 16 worst-case TID (MO-2407 in RB Vol. 10 has short operating time).
CA-98-026	Limitorque Motor Operators (class B motor)	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	2.00E+07	59.6%	RB Vol 12 only.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-026	Limitorque Motor Operators (Fibrite switches)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+07	9.63E+07	2.27E+08	>100%	Drywell worst-case, Beta shielding considered.
CA-98-027	Magnetrol Level Switches	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	2.20E+08	>100%	RB Vol. 12 only.
CA-98-028	McDonnell & Miller Flow Switches	6.31E+02	6.60E+02	<5.00E+06	3.58E+06		3.58E+06	5.00E+06	39.6%	RB Vol. 37 only and 10-foot from the SGTS filter. CLTP accident dose was rounded up for conservatism. Qualified dose based on material threshold dose level for Viton gasket.
CA-98-030	MicroSwitch Limit Switches	6.31E+02	6.60E+02		1.06E+07		1.06E+07	1.00E+07		RB Vols. 36, 37, and 39 only. Margin NA, DOR. The 1.0E+07 rad qualified dose based on threshold value. Tested to 1.1E8 rads with some embrittlement. EPU accident dose taken at 5-foot distance from SGTS filters. Reasonable to use more detailed analysis of distance from filters or higher change dose tolerance to qualify EPU conditions.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-032	Namco Limit Switches (50.49)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	(shielded)	7.63E+07	2.04E+08	>100%	Drywell worst-case TID among equipment locations (one limit switch in SGTS room located more than 10-foot filter, thus TID is bounded by Drywell doses). Beta dose shielded (sealed limit switches in Drywell).
CA-98-033	Namco Quick Disconnects EC210	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.04E+08	17.6%	Drywell worst-case. Beta dose reduced 50% due to localized shielding.
CA-98-035	Raychem NEIS Seals	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	5.00E+07	>100%	RB Vols. 9, 11, 12, 18, 19, and 31 only. RB Vol. 12 worst-case TID.
CA-98-036	Raychem Splices	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.25E+08	30.9%	Drywell worst-case (with splices >1-foot from SGTS filters). Beta dose reduced 50% due to localized shielding.
CA-98-037	Robertshaw Level Switch	3.79E+05	3.96E+05	3.30E+05	3.57E+05		7.53E+05	2.00E+06	>100%	RB Vol. 18 only. Accident dose based on 9 hour operating time (level switches only serve HPCI system).

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-038	Rockbestos Coax Cable	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.00E+08	15.0%	Drywell worst-case TID among equipment locations (none in SGTS rooms). Beta dose reduced 50% due to localized shielding.
CA-98-039	Rosemount Transmitter	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	4.40E+07	>100%	RB Vols 9 & 12 for Torus wide range level (LT-7338A/B) and RB Vols. 33 & 34 for containment wide range pressure (PT-7251A/B). Although test dose bounds postulated dose levels, performance issues related to radiation exposure in ranged-down specimens cause concern over the ranged-down condition of LT-7338A/B. At EPU, the increased accident dose may worsen the indicated level error in LT-7338A/B. These transmitters are obsolete models. Replacement of these transmitters is being made under EC 13086 [133.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-040	Rosemount 1153 Series B	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	2.62E+07	24.8%	RB Vols. 1 and 3 worst-case TID among installed locations.
CA-98-041	Rosemount Conduit Seals	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	1.11E+08	>100%	RB Vol 12 worst-case TID for equipment locations.
CA-98-042	Rotork "A" Range Actuators (DOR)	1.26E+05	1.32E+05		1.08E+06	.	1.21E+06	4.00E+06	>100%	RB Vols. 1, 3, and 10 only all with a 1-hour operating time. Thus RB Vol. 1 has worst-case TID (normal plus 1-hour dose). CLTP accident dose not shown due to dose source basis change for the RHR rooms.
CA-98-043	Rotork Valve Operators (50.49)	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	1.84E+08	>100%	RB Vol. 1 has worst-case TID of equipment locations.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-044	Static O-ring	4.29E+05	4.49E+05	5.30E+06	<4.00E+06		4.45E+06	8.00E+06	88.7%	RB Vols. 1, 3, 22, and 33 only. 10-hour accident dose for RB Vol. 1, normal dose for RB Vol.33 used for conservatism.
CA-98-046	Yarway Level	3.79E+05	3.96E+05	3.14E+05	3.40E+05 (3.35E5 actual per Ref. 99)		7.36E+05	1.00E+06	77.6%	RB Vols. 14 and 18 only. Level switches serve containment spray permissive. CLTP and EPU 180-day accident dose based on 3-foot distance from Source C piping (RWCU room).
CA-98-047	Samuel Moore Instrument Cable	6.31E+02	6.60E+02	1.00E+08	1.08E+08		1.08E+08	2.00E+08	84.7%	Outside Drywell, throughout RB, SGTS (>1 foot from filter) worst-case TID.
CA-98-049	Valcor Solenoid Valves	8.84E+04	9.24E+04	1.15E+07	1.25E+07		1.25E+07	5.90E+07	>100%	RB Vols. 11, 18, and 19 only. RB Vol. 11 worst-case TID.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-050	DG O'Brien Electrical Penetrations (M06 & M62/R31E5067G01)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	(shielded)	7.63E+07	2.20E+08	>100%	Although these penetrations are mounted on outboard side of Drywell nozzle, Drywell dose considered for conservatism. Beta shielded.
CA-98-050	DG O'Brien Electrical Penetrations (R19P1010G06 Series Plugs)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.25E+07	8.88E+07	1.25E+08	51.7%	Beta dose reduction credited.
CA-98-051	Reliance Motors	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	2.00E+08	>100%	RB Vols. 1, 3, 14, and 19 only. RB Vol. 1 worst-case TID.
CA-98-052	Tavis Flow Transmitter	6.31E+02	6.60E+02	1.00E+06	1.08E+06		1.08E+06	1.40E+06	29.2%	RB Vol. 37 only at 20 ft from SGTS filter.
CA-98-053	ITT Grinnel/Conoflow Transducer	6.31E+02	6.60E+02	5.00E+06	5.42E+06		5.42E+06	1.00E+07	84.7%	RB Vol. 37 only at 10 ft from SGTS filter.
CA-98-054	Consolidated Control Relays	3.79E+05	3.96E+05	7.00E+04	7.58E+04		4.72E+05	5.00E+05	37.3%	RB Vol. 18 only, 1 hour accident dose (relays support LLS function which has a 10-minute operating time per Criterion #3 of CA-94-086 [100]).

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-055	General Atomic Radiation Detector	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+08	2.76E+08			No radiation sensitive components. Raychem sealing sleeve at detector connector is addressed in EQ file CA-98-036 [26].
CA-98-059	Kerite Cable/Termination	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	2.20E+08	>100%	RB Vols. 1, 3, 5, 6, and 9 to 12 (cable runs midplane of Torus basemat). RB Vol. 1 worst-case TID.
CA-98-060	Westinghouse Starter and Transformer	1.01E+05	1.06E+05	2.34E+06	2.53E+06		2.64E+06	5.00E+06	93.1%	RB Vols. 14 and 19 only. RB Vol. 14 worst-case TID.
CA-98-062	Gould Contactor/ Disconnect	6.31E+02	6.60E+02	4.00E+06	4.33E+06		4.33E+06	1.00E+07	>100%	RB Vol. 37 location only at 10 ft (or more) from SGTS filter.
CA-98-064	Eaton Thermocouple Extension Cable	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.00E+08	15.0%	Drywell and RB locations, Drywell worst-case TID. Beta dose reduced 50% by localized shielding.
CA-98-065	Brand Rex 600V Instrument Cable (Drywell)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.00E+08	15.0%	Beta dose reduced 50% by localized shielding.
CA-98-065	Brand Rex 600V Instrument Cable (RB)	6.31E+02	6.60E+02	1.00E+08	1.08E+08		1.08E+08	2.00E+08	84.7%	All RB locations >1 ft from SGTS filter. RB Vols. 36 to 39 worst-case TID (>1 ft from filter).

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-066	Boston Control Cable	6.31E+02	6.60E+02	1.00E+08	1.08E+08		1.08E+08	2.00E+08	84.7%	All RB locations >1 ft from SGTS filter. RB Vols 36 to 39 worst-case TID (>1 ft from filter).
CA-98-067	CONAX Electrical Connector Seal	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.25E+08	30.9%	Drywell only. Beta reduced 50% due to localized shielding. Method of addressing Beta dose in is this assessment is more conservative than current EQ file basis.
CA-98-068	CONAX RTDs	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	2.27E+08	>100%	RB Vols. 9 to 12 only, RB Vol. 12 worst-case TID.
CA-98-069	Patel Conduit Seals (Drywell)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+07	9.63E+07	2.00E+08	>100%	Beta dose reduced by seal shell.
CA-98-069	Patel Conduit Seals (outside Drywell)	6.31E+02	6.60E+02	1.00E+08	1.08E+08		1.08E+08	2.00E+08	84.7%	All RB locations >1 ft from SGTS filter. RB Vols 36 to 39 worst-case TID (>1 ft from filter).
CA-98-070	Patel Conformal Coating (outside Drywell)	1.14E+05	1.19E+05	1.15E+07	1.25E+07		1.26E+07	2.00E+07	59.6%	All RB locations outside of RHR and SGTS rooms (RB Vols. 1 to 4 and 36 to 39). RB Vol. 12 worst-case TID. (RHR and SGTS rooms not steam harsh environments where coating needed)
CA-98-071	EGS Grayboot Electrical Connectors	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+07	9.63E+07	2.08E+08	>100%	Drywell worst-case TID, Beta dose reduction credited.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-072	EGS Quick Disconnect	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.00E+08	15.0%	Drywell worst-case TID (when considering >1' from SGTS filter dose). Beta dose reduced 50% due to localized shielding,
CA-98-073	Raychem/Swagelok Conduit Seals	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	5.00E+07	>100%	RB Vols. 1, 3, 7, 9, 12, 14, 18, 19, 22, 33, and 34 only. RB Vol. 1 worst-case TID.
CA-98-075	Weed Thermocouples (and Patel Conformal Coating)	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+08	2.76E+08	3.06E+08	11.5%	Drywell only. Although Beta dose is shielded for this equipment, addressed for conservatism.
CA-98-076	Rome Cable Type SIS	4.45E+04	7.57E+04	2.54E+06	2.75E+06		2.83E+06	5.00E+08	>100%	RB Vols. 5 and 8 only. RB Vol. 5 worst-case TID.
CA-98-077	Eaton Cutler-Hammer Relays	4.29E+05	4.49E+05	4.88E+05	5.29E+05 (9.27E5 per new Alion data)		9.77E+05	1.43E+06	85.7%	RB Vols. 22, 33, and 34 only, RB Vol. 33 worst-case normal dose. CLTP and EPU accident dose based on 1-ft dose distance from Source "C" and for 30 day functional time per modification 97Q055 [125].

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-078	PEI/FENWAL Temperature Switch	2.97E+06	2.97E+06				2.97E+06	5.00E+07		RB Vols. 5, 8, 9, 12, and 16 only. No accident dose (and thus no margin required), temperature switches only needed for HELB. RB Vol. 16 worst-case.
CA-98-079	ITT-Royal PVC Cable	4.29E+05	4.49E+05	2.34E+06	2.53E+06		2.98E+06	1.00E+07	>100%	RB Vols. 14, 18, 19, 22, and 33 locations only. RB Vol. 33 worst-case TID.
CA-98-080	Okonite Control Cable	3.79E+05	3.96E+05	2.34E+06	2.53E+06		2.93E+06	1.00E+07	>100%	Cable evaluated for RB Vols. 5, 14, 18, 19, and 22. RB Vol. 18 has worst-case TID.
CA-98-081	Triangle Triolene Control Cable	3.79E+05	3.96E+05	7.00E+04	7.58E+04		4.72E+05	1.00E+06	>100%	Cable only to SV-3269A routed through RB Vols. 18 and 19. Accident dose based on 1-hour operating time.
CA-98-082	MNGP-A Cable	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	5.00E+07	>100%	All RB except Vols. 16, and 28 to 32. RB Vol. 1 has worst-case TID.
CA-98-083	MNGP-B Cable	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	4.10E+07	95.7%	All RB except Vols. 16, and 28 to 32. RB Vol. 1 has worst-case TID.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-084	Amphenol Connectors	4.29E+05	4.49E+05	3.50E+06	4.60E+06		5.05E+06	6.00E+06	20.6%	RB Vols. 1, 3, 14, 18, 19, 22, and 33 at instrument racks. Normal dose from RB Vol. 33 for conservatism. Associated equipment in RHR rooms, PS-14-44(A-D) and PS-10-105(A-H) all support ADS. Per CA-94-086 [100] criterion #2 and MPS-0167AB, ADS functions within 10-hours. 10-hour accident dose of RHR rooms bounds 180-day dose of other volumes.
CA-98-085	Pyco Temperature Elements	8.84E+04	9.24E+04	2.34E+06	2.53E+06		2.63E+06	2.20E+08	>100%	RB Vol. 28 only. Further, equipment mitigates an RWCU HELB only, no accident dose would be postulated but is considered for conservatism.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-98-086	SOR Pressure Switches	4.29E+05	4.49E+05	2.34E+06	2.53E+06		2.98E+06	3.30E+07	>100%	RB Vols. 14, 22, and 33 only. RB Vol. 33 has worst-case TID.
CA-98-101	General Electric Terminal Blocks	4.29E+05	4.49E+05	2.34E+06	2.53E+06		2.98E+06	2.20E+08	>100%	RB Vols. 22, 33, and 34 only. RB Vol. 33 has worst-case TID.
CA-98-103	Patel P-1 Thread Sealant	1.79E+07	1.88E+07	5.31E+07	5.75E+07	2.00E+08	2.76E+08	1.50E+09	>100%	Drywell worst-case TID.
CA-98-104	Rockbestos Firewall SR Cable	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.00E+08	15.0%	Drywell worst-case when considering cables >1' from SGST filter. Beta dose reduced 50% due to localized shielding.
CA-98-107	Rockbestos Firewall III/SIS Cable	1.79E+07	1.88E+07	5.31E+07	5.75E+07	1.00E+08	1.76E+08	2.00E+08	15.0%	Drywell worst-case when considering cables >1' from SGST filter. Beta dose reduced 50% due to localized shielding.
CA-98-108	Rockbestos Firewall EP Cable	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	1.30E+08	>100%	All of RB and >10 ft from SGTS filter. RB Vol. 1 has worst-case TID.
CA-98-109	Valcor MSIV Solenoid Valves	2.97E+06	2.97E+06	8.00E+04	8.66E+04		3.05E+06	5.00E+07	>100%	RB Vol. 16 only, 1-hour accident dose.
CA-98-128	UCI Electrical Tape Terminations	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	5.51E+07	>100%	All of RB, but >20 foot from SGST filter. RB Vol. 1 has worst-case TID.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-02-197	Dow Corning 3-6548 Silicone RTV Foam	4.45E+04	7.57E+04				7.57E+04	1.36E+06		RB Vols. 5 and 8 for HPCI/RCIC steam break HELBs only, no accident dose. RB Vol. 5 has bounding normal dose.
CA-03-096	Loctite PST 580 Thread Sealant (outside Drywell)	1.26E+05	1.32E+05	2.61E+07	2.09E+07		2.10E+07	7.37E+07	>100%	All of RB except SGTS rooms (RB Vols. 36 to 39 as no device sealing required there). RB Vol. 1 thus is worst-case TID.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-03-096	Loctite PST 580 Thread Sealant (Drywell)	1.79E+07	1.88E+07 (60 year)	5.31E+07	5.44E+07 (actual)	(shielded)	7.32E+07	7.37E+07	0.9%	Earliest use in Drywell was on MSIV cluster in February 2000. If installed for remaining plant life of 31 years, the EPU normal dose would be 1.18E+07 rad, determined using the 43.32 R/hour EPU dose rate per CA-08-067 [97]. The total dose would then be 6.62E7 rad and the margin would then be 13.7%. Beta dose is considered shielded by defined use of thread sealant on metallic connection interface.
CA-03-105	Scotch 130C and 69 Electrical Tape	6.31E+02	6.60E+02	1.00E+08	1.08E+08		1.08E+08	1.83E+08	69.0%	All of RB and >1' from SGTS filter. RB Vols. 36 to 39 thus have worst-case TID.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-05-137	Fisher E/P Transducer	1.26E+05	1.32E+05		<4.00E+06		4.13E+06	6.00E+06	46.8%	RB Vols. 2 and 4 only. E/P converters serve CV-1728/1729 and located part way down RHR room stairways. Thus, they are located distant from the RHR heat exchanger and piping contributing to the dose in the area. Therefore, the EPU 180-day accident dose shown is the contact dose for Source C on the Torus for conservatism. No CLTP accident dose shown due to methodology changes of RHR source basis.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-05-138	Cutler-Hammer Motor Starter/Control Transformer	3.65E+04	4.12E+04	2.34E+06	2.52E+06		2.56E+06	2.80E+06	9.4%	RB Vols. 14 and 19 only. Equipment installed in 2005 with 26 years of plant life remaining. Normal doses shown are based on 26 years at 160 mRem/hr dose rate per CA-08-067 [97] (EPU value is 1.13 times higher). Although margin does not meet the recommended +10%, it is observed that the accident dose prescribed for the areas is taken from the contact dose in the RWCU rooms. At 1-foot or more from the piping, the 180-day accident dose reduces to less than 1.00E+06 rad. Consequently, margin can be demonstrated for this equipment.

Table 3.4.7-1 Normal and Accident Radiation Evaluation for EPU

EQ Calc	Title	CLTP Normal Dose (rad) (CA-08-067)	EPU Normal Dose (rad) (CA-08-067)	CLTP Accident Dose (rad)	EPU Accident Dose (rad)	Beta Dose (rad)	Total EPU TID (rad)	Qual. Dose (rad)	Margin	Remarks
CA-05-140	ASCO Scram Solenoid Pilot Valves (SSPV)	2.10E+04	2.38E+04				2.38E+04	8.55E+04		RB Vols 14 and 18 only. SSPV valves are changed every 15 years. CLTP and EPU normal doses based on 160 mRem/hour dose rate (EPU dose rate adjusted by +13%) per CA-08-067 [97]. It is noted that the 600 mRem/hour contact dose rate now prescribed for RB Vol. 18 is not near the SSPVs, the sensitivity dose rate value for RB Vol. 18 at 30 cm distance is bounded by the 160 mRem/hour dose used above. Equipment serves scram function, no accident dose to consider.

3.5 Plant Performance / Equipment Out of Service Options

Item	Option	EPU Impact
1	None	N/A

4.0 Recommendations and Observations

4.1 Recommendations

Item	Subject	Recommendation
1	Radiation Doses	1. Provide additional material radiation documentation to support the qualification of the following items: • General Electric Fan Motors qualified by CA-98-024 • Microswitch Limit Switches qualified by CA-98-030 This equipment was discussed in Table 3.4.7-1 to extend qualification levels through additional material analyses for this DOR equipment. 2. Replace: • Rosemount Model 1153 Series A transmitters in the Torus compartment at functional locations LT-7338A/B, the higher EPU accident dose decreases their expected performance.
2	Turbine Building Areas Reclassified as EQ Harsh	As outcome from EWRA 1131374-20 [117], update the associated EQ Files to ensure the cables routed through the newly created harsh Turbine Building areas under EPU are addressed as detailed in Section 3.4.5.
3	Post-LOCA Heatup in RB	Incorporate new EPU post-LOCA heat up conditions in the EQ files and implement any necessary changes in replacement intervals due to qualified life changes. Also, replace Rosemount level transmitters LT-7338A/B as they would not posses adequate life margin when accounting for the higher PLHU conditions in the Torus compartment.

4	ITT Royal Cable	Add thermal lag analysis to CA-98-079 to demonstrate cable temperatures will remain below qualification temperatures for RWCU line breaks.
5	EQ Supporting Documentation (Configuration Management Issues)	REVISE EQ-Part-B for EPU conditions.

4.2 Observations

Item	Subject	Observation
1	Known Equipment Replacements (not EQ driven)	• The inboard MSIV air pack/solenoid cluster is being replaced as a result of EPU for differential operating pressure concerns on an air poppet valve. Although the modification also includes replacement of the solenoid cluster (the EQ portion), the same model/qualification pedigree as the currently installed units is being utilized. However, the electrical connector on the junction box is being changed from a Namco EC210 to an EGS quick disconnect. The modification is being implemented in the spring 2009 outage (EC 11988 [131]). The EGS connectors (EQ File CA-98-072 [55]) and the MSIV solenoid cluster (EQ File CA-98-008 [6]) are already qualified for Drywell conditions and are assessed for EPU impact in this evaluation. • SGTS flow switches (FT-2590/2591) are being replaced under work orders 335194 (FS-2951) and 345062 (FS-2950) due to equipment obsolescence issues. A new environmental qualification document file (QUAL-08-013 [135]) is being developed based on new IEEE Standard 323-1974 type testing completed in 2008 for Monticello. This evaluation only addressed the current EQ basis (EQ File CA-98-028 [21]) for EPU impact and not the new testing. The current DOR qualified switches are acceptable for EPU conditions if they do not get changed. • The SCRAM Solenoid Pilot Valves (SSPV) on all of the control rod drive units

Item	Subject	Observation
		are being changed from the current ASCO solenoid valve model (EQ Files CA-05-140 [81]) to an AVCO design under EC 12044 [132] (new EQ file QUAL-08-015). This evaluation only addressed the currently installed equipment for EPU impact. The qualification of the replacement AVCO valves is being appropriately considered under the normal design review processes of EC 12044. • As indicated in Sections 3.4.6 and 3.4.7 of this evaluation, the Torus wide range level transmitters are anticipated to have degraded performance under accident radiation dose and not maintain a sufficient qualified life and yet satisfy Reactor Building post-LOCA temperature requirements under EPU. These transmitters are being replaced under EC 13086 [133] (new EQ File QUAL-08-016). This evaluation does not address the qualification of the replacements transmitters for EPU. • The outboard MSIV Namco limit switches (EQ file CA-98-032 [23]) with integral Namco EC210 connectors (EQ file CA-98-033 [24]) are being replaced with similarly qualified limit switches, but new EGS quick disconnects under EC 11561 [134]. The EGS quick disconnects (EQ file CA-98-072 [55]) are currently qualified for Drywell conditions. This evaluation only addresses the currently installed MSIV limit switches and connectors for EPU impact.
2	EQ Equipment Distances from SGTS Charcoal Filter	• GAR 01150954 [136] is prepared to confirm the distances from the various EQ devices in the SGTS rooms from the charcoal filter. The CLTP analysis distance factors credited are listed in Table 3.4.7-1 where appropriate.

Appendix 2 – Thermal Aging/Qualified Lives (Continued)

Comparing terms of this result and the Fosta 512 life equation enables a determination of the activation energy for the Fosta 512 nylon material as follows:

use 50 years at 90°F
10 years at 90°F + 16°C
198 days at 124.2°F + 16°C

$$E_a = \frac{6233.876}{\log_{10}(e)} \times K_b = \frac{6233.876}{0.4343} \times 8.617x10^{-5} = 1.24\,\text{eV}$$

90°F

The normal temperature for Reactor Building Volume 37 is ~~80°~~F. The Allen Bradley terminal boards are used in SBGT power panels C-87A/B [19]. One terminal board for control and another for the heater power circuit. The SBGT functions to mitigate the consequences of a design basis LOCA. During this time, Volume 37 will experience a heatup to 1~~12.8°F [17]~~. Furthermore, the current flow due to SBGT heater energization will cause a temperature rise effect in the power circuit. Appendix 2 of CA-98-017 [20] analyzed the temperature rise effect for the cables serving EQ equipment. For the SBGT heaters, the cable temperature rise was determined to be 16°C when energized. Although the SBGT may be energized for 180 days post-LOCA, the system may also be energized for testing. To conservatively account for normal and post-accident aging effects for the terminal boards, 50 years at ~~80°F plus 11 years at 142°F (bounds 112.8°F + 16°C~~) is considered in the following aging analysis:

153°F (340.37K)

90°F (305.37K)

124.2°F

Letting,

t_{s1}	=	Normal 50-year life at ~~80°F~~
t_{s2}	=	Post-LOCA and testing duration (11-years for conservatism)
E_a	=	Activation Energy (1.24 eV)
t_a	=	Test Time (521 hours per Reference [16])
T_{s1}	=	Normal service temperature = ~~80°F (299.81K~~)
T_{s2}	=	Post-LOCA ~~and system testing~~ temperature ~~142°F (334.26K~~)
T_a	=	Test Temperature = 228°F (382.04K)
K_b	=	Boltzmann's Constant ($8.617x10^{-5}$ eV/K)

90°F + 16°C = 118.8°F (321.37K)

$$t_a = \sum_i t_{s_i} \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_a} - \frac{1}{T_{s_i}}\right)\right] = 50\text{ years} \times \exp\left[\frac{1.24}{8.617x10^{-5}}\left(\frac{1}{382.04} - \frac{1}{\cancel{299.81}}\right)\right] +$$

10

$$\cancel{11}\text{ years} \times \exp\left[\frac{1.24}{8.617x10^{-5}}\left(\frac{1}{382.04} - \frac{\cancel{1}}{\cancel{334.26}}\right)\right]$$

+ equivalent of 198 days at 124.2°F + 16°C = 153°F (340.37K)

$$= \cancel{15}\text{ hours} + \cancel{443}\text{ hours} = \cancel{458}\text{ hours}$$

As seen, the equivalent plant aging is bounded by the test duration of 521 hours. As such, the Allen Bradley terminal boards are qualified for 60-years service, plus 180-day accident operation. The analysis conservatively addresses the temperature rise effects due to system testing and post-LCOA operation.

= 152.9 hours

34.2 hours

71.5 hours

+ 47.2 hours (PLHU)

C. EQUIPMENT DATA SHEET

(1) System	(2) Plant ID/ (Mech. Comp.)	(3) Component Type	(4) Manufacturer/ Model No.	(5) Function	(6) Accident Requirement	(7) Environmental Specification
RSW	PS-7192	Pressure Switch	ASCO/ SA11AR-TG10A22R	RHR Aux Air Compressor Start Switch (K-10A)	Vol. 14 [9]	B.1.11 [9] Rx Bldg 935'SE
RSW	PS-7193	Pressure Switch	ASCO/ SA11AR-TG10A22R	RHR Aux Air Compressor Start Switch (K-10B)	Vol. 19 [9]	B.1.12 [9] Rx Bldg 935'SW
RHR	PS-10-105A	Pressure Switch	ASCO/ SB11AMR-TH10A32R	Logic for Auto Blowdown (Note 1)	Vol. 1 [9]	B.1.4 [9] A RHR Room 896'
RHR	PS-10-105B	Pressure Switch	ASCO/ SB11AMR-TH10A32R	Logic for Auto Blowdown (Note 1)	Vol. 3 [9]	B.1.4 [9] B RHR Room 896'
RHR	PS-10-105C	Pressure Switch	ASCO/ SB11AMR-TH10A32R	Logic for Auto Blowdown (Note 1)	Vol. 1 [9]	B.1.4 [9] A RHR Room 896'
RHR	PS-10-105D	Pressure Switch	ASCO/ SB11AMR-TH10A32R	Logic for Auto Blowdown (Note 1)	Vol. 3 [9]	B.1.4 [9] B RHR Room 896'
RCI	PS-13-87A	Pressure Switch	ASCO/ SB21AMR-TG23A42R	RCIC Turbine Shutdown	Vol. 14 [9]	B.1.9 [9] Rx Bldg 935'E
RCI	PS-13-87B	Pressure Switch	ASCO/ SB21AMR-TG23A42R	RCIC Turbine Shutdown	Vol. 14 [9]	B.1.9 [9] Rx Bldg 935'E
RCI	PS-13-87C	Pressure Switch	ASCO/ SB21AMR-TG23A42R	RCIC Turbine Shutdown	Vol. 14 [9]	B.1.9 [9] Rx Bldg 935'E
RCI	PS-13-87D	Pressure Switch	ASCO/ SB21AMR-TG23A42R	RCIC Turbine Shutdown	Vol. 14 [9]	B.1.9 [9] Rx Bldg 935'E
RHR	PS-2-128A	Pressure Switch	ASCO/ SB11AKR-TG13A42R	Shutdown Cooling Isolation	Vol. 14 [9]	B.1.9 [9] Rx Bldg 935'E
RHR	PS-2-128B	Pressure Switch	ASCO/ SB11AKR-TG13A42R	Shutdown Cooling Isolation	Vol. 14 [9]	B.1.9 [9] Rx Bldg 935'E

Only PS-10-105(A-D) in RB Volumes 1 and 3 and PS-13-87(A-D) in RB Volume 14 contain the "M" style switch, see thermal aging section later in this EPU evaluation for PLHU effects.

Add "Note 3"

PCT	PS-4664	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 9 [9]	B.1.6 [9] Torus Compartment
PCT	PS-4665	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol 12 [9]	B.1.6 [9] Torus Compartment
PCT	PS-4666	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 31 [9] (Note 2)	B.1.5 [9] RWCU Pump Room 962'-6"
PCT	PS-4667	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 11 [9]	B.1.6 [9] Torus Compartment
PCT	PS-4668	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 9 [9]	B.1.6 [9] Torus Compartment
PCT	PS-4669	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 9 [9]	B.1.6 [9] Torus Compartment
PCT	PS-4670	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 12 [9]	B.1.6 [9] Torus Compartment
PCT	PS-4671	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 31 [9] (Note 2)	B.1.5 [9] RWCU Pump Room 962'-6"
PCT	PS-4672	Pressure Switch	ASCO/ SB22BR-TG23A42R	Control Room Indication	Vol. 11 [9]	B.1.6 [9] Torus Compartment

Note 1: Pressure switches PS-10-105(A-D), located in the RHR pump rooms (Reactor Building (Volumes 1 and 3), function in the short term following a LOCA [47]. As such, a conservative operating time of 10 hours is considered in this analysis for these switches. ~~All other pressure switches are conservatively assumed to be required for 180 days post-accident.~~

These switches support ADS, Criterion #2 of CA-94-086 indicates up to 10-hours operability for ADS

Note 2: Pressure switches PS-4666 and PS-4671 serve Drywell vent isolation valves AO-2386 and AO-2387, respectively [48]. These isolation valves are in series on line CP2-18-HE which leads to the Standby Gas Treatment system [49]. These two pressure switches are located in Reactor Building Volume 31 which experiences a peak HELB pressure of 17.19 psia. However, per [48] and [49], it is observed that the safety function of these switches is for inside Drywell events. As such, the pressure switches do not require qualification for peak HELB pressure in Volume 31.

Note 3: Pressure switches PS-13-87(A-D) only serve isolation of RCIC steam line under RCIC system HELB. These switches are not required post-LOCA. Their operating time can thus be reduced to less than 1-hour.

Appendix 2 - Thermal Aging

Required: 47 years (Remaining Plant Life with a renewed license since 1984)

Reported: Various

Discussion: The qualified life of the ASCO pressure switches can be determined by extrapolating the simulated aging data [3] to the actual average temperature seen by the switches during normal operation. The Arrhenius methodology is an accepted methodology for extrapolating this data. The Arrhenius equation can be presented as follows [46].

$$t_a = t_t * e^{\left[E_a/K * \left(1/T_a - 1/T_t \right) \right]}$$

Where:

t_a = service life

t_t = test exposure time

E_a = activation energy - eV

K = Boltzmann's Constant (8.617 x 10^{-5} eV/K)

T_a = service temperature (Kelvin)

T_t = test temperatures (Kelvin)

As can be seen from the equation, it is necessary to determine the activation energy of the non-metallic components in the pressure switches. Reference [50] describes the nuclear grade pressure switches while Appendix B of Ref [3] presents a summary of the thermally degradable materials of construction that ASCO considers critical for satisfactory performance of the pressure switches. The summary includes the activation energies for those materials and addresses ASCO's entire line of pressure switches.

For the simulated aging testing [3], ASCO tested seven (7) different units. These were tested for different durations and temperatures based on the elastomeric materials. The goal of the testing was to qualify the components for 10 years @ 104°F. All thermal aging was conducted at 210°F [3, Table 5.1]. The following table summarizes the demonstrated thermal lives of the non-metallic materials used in the switches [3, Appendix B] and equates the testing data to ambient temperature conditions of ~~80°F, 85°F, and 114°F~~.

New Ambient temperatures:

85°F: RB Vols 1, 14, and 19
90°F: RB Vols 3, 9, 11, 12
114°F: RB Vol 31

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45.7 ("M" switch @ 85°F)

Demonstrated Lives of Pressure Switch Materials

97.8 (@90), >100 (@85), 18.7 (@ 114)

90°F

Material (Note 1)	Aging Time at 210°F		Plant Ambient Temperature ~~80°F~~	85°F	114°F
Urethane	30 days	0.82 eV	(----------Note 2----------)		
Silicone	18 days	0.91 eV	(----------Note 3----------)		
EPDM	15 days	0.94 eV	(----------Note 4----------)		
Viton	~~8.5~~ days	1.04 eV	~~57.64~~	~~39.80~~	~~5.30~~
Mylar	30 days	1.22 eV	>100	>100	47.90
Poly/glass	30 days	1.86 eV	>100	>100	>100
"M" Switch - MT-4R-A28 (Note 5)	15 days	0.98 eV	~~64.80~~	~~(--------Note 5--------)~~	
Non-M Switch - BZ-2R24-A2	30 days	1.00 eV	~~>100~~	>100	15.19

32.5

74.5

Note 5 now only applies to 114°F column

NOTES:

1. Materials list and activation energies taken from Appendix B of Reference [3], aging times developed from Tables 3.2 and 5.1 of Reference [3].

2. References [19] & [40] indicates that Urethane only used on high pressure models with transducers designation beginning with "TL" or "TM". There are no ASCO pressure switches installed at Monticello with Urethane, thus no qualified life is computed for this material.

3. Reference [18] indicates that Silicone is only used on transducers having a "44" in their model code. There are no ASCO pressure switches installed at Monticello with Silicone, thus no qualified life is computed for this material.

4. By review of Reference [50], it is observed that the EPDM material is only used with pressure transducers with having a "16", "26", "36", or "46" in the model number code. There are no such model installed at Monticello, thus no demonstrated thermal life is computed for this material.

5. Reference [14] indicates pressure switches with an "M" suffix designate micro-switch model type MT-4R-A28. Pressure switches with an "M" designator are only located in Volumes 1, 3, and 14 where the normal ambient is ~~80°F~~.

Can credit Specimen #3 which contained Viton and aged 30 days

85°F or 90°F

Post-LOCA Operability

"Insert A" new text, attached

~~The worst case post-LOCA heat-up (Figure 1, Appendix 1) is bounding for all pressure switches Reactor Building Volumes 9, 11, 12, 14, and 19. The pressure switches in Volumes 1 and 3 have short term functions and are bounded by the accident test (Appendix 1). The~~

pressure switches in the RWCU room (Volume 31) are not subject to increased post-LOCA heat-up effects.

As such, the bounding post-LOCA heat-up profile from the Torus compartment (peaks at 160.4°F) will be analyzed at the two normal ambient temperature conditions of 80°F and 85°F to address post-LOCA operability for the switches in Volumes 9, 11, 12, 14, and 19. The post-LOCA operability of the switches in Volumes 31 will simply have 198 days subtracted from the demonstrated life above (i.e. 5.30 years). The worst-case post-LOCA heat profile (Volume 9) can be expressed as 153 days at 104°F (duration when temperature is 104°F, or less), 39 days at 130°F (duration when temperature is >104°F), and 6 days at 161°F (duration when temperature is greater than 130°F through 160.4°F peak condition). The life limiting material was shown above to be Viton. The equivalent of the post-LOCA heat-up curve, using Arrhenius methodology and the 1.04 eV activation energy for Viton at the ambient temperatures of 80°F (299.81K) and 85°F (302.59K), is computed as:

Temperature Range	Days	Temp	Time (days) at 85°F	Time (days) at 80°F
104°F or less	153	104°F	587	849
Between 104°F and 130°F	39	130°F	820	1184
Greater than 130°F	6	161°F	793	1147
			2,200 days (6.0 years)	**3,180 days (8.7 years)**

CONCLUSION

The qualified life is, therefore, the difference of the demonstrated life minus the equivalent operability time for the switches: 6 years for those in an 85°F, 8.7 years for those at 80°F, and 0.54 years for those at 114°F. The qualified lives are as follows:

Plant ID	Ambient	Qualified Life
PS-7192, 7193, PS-10-105(A–D), PS-13-87(A–D), and PS-2-128(A & B)	80°F	48.9 years
PS-4664/4665/4667/4668/4669/4670/4672	85°F	33.8 years
PS-4666 / 4671	114°F	4.76 years (Viton limited)

INSERT "A"
New EPU Post-LOCA Operability and Conclusion Sections for EQ File 98-006

Post-LOCA Operability

The pressure switches in Volumes 1 and 3 (all) as well the "M" style switches in RB Volume 14 (PS-13-87A/B/C/D) have short-term functional durations which are bounded by the accident test (Appendix 1). Thus, only the non-"M" style switches need address for 180 day (198 days with margin) post-LOCA operation.

For the switches in the Torus compartment (RB Volumes 9, 11, and 12), the PLHU event can be expressed as 162 days at 125°F (duration when temperature is 125°F, or less), 20 days at 140°F (duration when temperature is >125°F, but less than or equal to 140°F), 11 days at 155°F (duration when temperature is greater than 140°F, but less than or equal to 155°F), and 5 days at 179°F (duration when temperature is greater than 155°F, or initial transient period to peak PLHU condition). For the switches in the Torus compartment, the life limiting material was shown to be the non-"M" style internal switches. The equivalent of the post-LOCA heat-up curve, using Arrhenius methodology and the 1.00 eV activation energy at the ambient temperature of 90°F (305.37K), is computed as:

Temperature Range	Days	Temp	Time (days) at 90°F
125°F or less	162	125°F	1,576
Over 125°F, but equal/under to 140°F	20	140°F	476
Over 140°F, but equal/under 155°F	11	155°F	612
Greater than 155°F	5	179°F	998
			3,662 days (10.0 years)

Thus, for the pressure switches in the Torus compartment, 10 years at 90°F for the equivalent accident duration will be taken from the previously computed minimal thermal life of 74.5 years resulting in a qualified life of >60 years.

For the pressure switches located in RB Volumes 14 or 19 (non-"M" variety), the worst-case peak PLHU temperature is 121.3°F. Assuming this PLHU temperature prevails for the entire 198 day post-accident duration (180 days + 10% margin), the equivalent of 121.3°F at the normal ambient temperature of 85°F is determined as 6.0 years. Subtracting this POAT equivalent life from the previously determined minimal thermal life of >100 years gives a qualified life greater than 60 years for the non-"M" variety pressure switches in RB Volumes 14 or 19.

For the pressure switches in RB Volume 31, (non-"M" variety), the worst-case peak PLHU temperature is 131.9°F. Assuming this PLHU temperature prevails for the entire 198 day post-accident duration (180 days + 10% margin), the equivalent of 131.9°F at the normal ambient temperature of 114°F is determined as 1.6 years. Subtracting this POAT equivalent life from

the previously determined minimal thermal life of 15 years gives a qualified life of 13.4 years for the non-"M" variety pressure switches in RB Volume 31.

CONCLUSION

The qualified life is, therefore, the difference of the demonstrated life minus the equivalent operability time for the switches determined above. The qualified lives, per component ID are as follows:

Plant ID	RB Volume	Qualified Life
PS-10-105(A/C) {"M" style switch}	1 (85°F)	45.7 years
PS-10-105B/D {"M" style switch}	3 (90°F)	32.5 years
PS-13-87(A–D) {"M" style switch}	14 (85°F)	45.7 years
PS-7192, 7193 {non-"M", long-term} PS-2-128(A & B)	14/19 (85°F)	>60 years
PS-4664/4665/4667/4668/4669/4670/4672	9/11/12 (90°F)	>60 years
PS-4666 / 4671	31 (114°F)	13.4 years

Qualified life change summary (comment for EPU evaluation, not to be part of revised EQ file):

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 the "M" or non-"M" style internal switch for all ASCO pressure switches. This explains the qualified life increase observed for the installed switches. In all RB cases, except Volume 31, the normal ambient temperature has increased, but due to the "doubling of the aging of the Viton, this effect was masked in this re-evaluation. In summary, the following qualified life changes are made:

PS-10-105(A/C) from 48.9 to 45.7 years ("M" switch and +5°F higher ambient)
PS-10-105(B/D) from 48.9 to 32.5 years ("M" switch and +10°F higher ambient)
PS-13-87(A-D) from 48.9 to 45.7 years ("M" switch and +5°F higher ambient)
PS-7192/7193 and PS-2-168(A/B) from 48.9 to >60 years (non-"M" switch in leu of Viton)
PS-4664 to PS-4672 (except 4666 & 4671) from 33.8 to >60 years (non-"M" switch in leu of Viton)
PS-4666/4671 from 4.76 to 13.4 years (non-"M" switch in leu of Viton)

So the significant reduction is for pressure switches in RB Volume 3. The earliest these switches were installed was 1984. Thus, assuming these were installed since that time, would require replacement by 2016.

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85°F (302.59K)

$$t_s = 1000 hours\, Exp^{\left[\frac{0.86}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.82}} - \frac{1}{380.4}\right)\right]}$$

"Aging" of Buna-N taken as 1,000 hours at 225°F (380.4K)

$$\cancel{= 1.15x10^6\ hours = 131.6}\ years$$

=97 years

For all switches in the RCI, REC, RHR, and MST systems, the accident testing of 212°F for 6 hours bounds their operating time of 1-hour or less. The switches in the PCT system (DPIS-2572/2573) are required for 180 days (198 days with 10% margin) in a worst-case post-LOCA temperature of ~~107.5°F (315.09K)~~ in Volume 14 [39]. The accident aging life for these switches can be determined in terms of normal life at ~~80°F (299.82K)~~ as follows:

115.6°F (319.59K)

$$t_s = 198 days * Exp^{\frac{0.86}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.82}} - \frac{1}{315.09}\right)}$$

$$t_s = \cancel{2.72}\ years$$

3.14 years

The Qualified Life can now be determined:

Service Life	Accident Profile (Vol. 14)
~~131.6~~ years	~~2.72~~ years
~~131.6~~ years - ~~2.72~~ years = ~~128.8~~ years	

Therefore, the Qualified Life of the ITT Barton pressure switches (Models 278, 288, 288A, and 289A) are greater than 60 years.

93.8 years

GE Cable Type	Normal Plant	Accident	Comment
SI-58007/SI-58136	5°C*	16°C	Applies to SI-58007 power cables only
SI-57275	5°C*	16°C	Not applicable to Drywell or Steam Chase (RB cables with load not routed in Steam Chase)
SI-58109/SI-58081	N/A	N/A	These are control cable types

* Rounded-up from the 2°C determined earlier for conservatism. Additionally, a +20°F panel temperature rise (assumed) will be included for GE Type SI-57275 SIS wire.

SI-58007 and SI-58136

Required: 60 Years
Reported: >60 years

From Reference 20, the SI-58136 sample was tested to the 268°F for 54 days. The non-metallic materials present are the Butyl rubber insulation and neoprene jacket [2] with activation energies of 1.08 eV and 0.87 eV, respectively [17]. Arrhenius methodology [17] to determine the qualified lives is demonstrated below. The maximum normal ambient temperature anywhere in the Reactor Building is ~~90°F~~ (excluding Steam Chase, RWCU rooms, and Drywell). The qualified life of the Butyl cables are:

95°F (308.15K)

Neoprene

t_s = Qualified life @ Service Temperature
E_a = Activation Energy = 0.87 eV [17]
t_a = Test Time = 54 days (Reference 20)
T_s = Normal Service Temperature = ~~90°F + 5°C (310.37K)~~ 95°F + 5°C =104°F (313.15K)
T_a = Test Temperature = 268°F (404.26K) (Reference 20)
K_b = Boltzmann's Constant (8.617 x 10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 54 \times \exp\left[\frac{0.87}{8.617x10^{-5}}\left(\frac{1}{\cancel{310.37}} - \frac{1}{404.26}\right)\right] = \cancel{282}\ 211.6 \text{ years}$$

Butyl Rubber

t_s = Qualified life @ Service Temperature
E_a = Activation Energy = 1.08 eV [17]
t_a = Test Time = 54 days (Reference 20)
T_s = Normal Service Temperature = ~~90°F + 5°C (310.37K)~~
T_a = Test Temperature = 268°F (404.26K) (Reference 20)
K_b = Boltzmann's Constant (8.617 x 10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 54 \times \exp\left[\frac{1.08}{8.617x10^{-5}}\left(\frac{1}{\cancel{310.37}} - \frac{1}{404.26}\right)\right] = \cancel{1,750}\ 1,222 \text{ years}$$

Compensating for the effects of thermal aging due to a Post LOCA heatup in the Reactor Building, the Arrhenius methodology will be applied to the Neoprene, as it is the limiting material of construction.

Peak EPU PLH 179.1°F + 16°C = 207.9°F (370.87K)

Neoprene

t_s	=	Qualified life @ Service Temperature
E_a	=	Activation Energy = 0.87 eV [17]
t_a	=	Accident Time = 197 days (Appendix 1)
T_s	=	Normal Service Temperature = ~~90°F +5°C (310.37K)~~
T_a	=	Accident Temperature = ~~160.4°F +16°C (360.48K), per Appendix 1~~
K_b	=	Boltzmann's Constant (8.617 x 10^{-5} eV/K)

95°F + 5°C =104°F (313.15K)

81.5

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 197 \times \exp\left[\frac{0.87}{8.617x10^{-5}}\left(\frac{1}{\cancel{310.37}} - \frac{1}{\cancel{360.48}}\right)\right] = \cancel{49.6} \text{ years}$$

Thus, subtracting the thermal aging due to a post LOCA environment from the thermal life yields a result of ~~232.4 years (282 Years – 49.6 Years)~~. As can be seen from the analysis above, the qualified life of SI-58007 and SI-58136 cables are well beyond the 60 year anticipated life of the plant (with extended operation under a renewed license).

211.6 - 81.5 = 130.1 years

GE:SI-57275 Switchboard Wire

Required:	60 Years
Reported:	> 60 Years

The only non-metallic on the SI-57275 switchboard wire is cross-linked polyethylene (XLPE). From Reference 17, Appendix B, the most conservative listed activation energy for XLPE is 1.13 eV. Using this information and the Arrhenius equation, the thermal life is as follows:

ambient remains valid

Cross Linked Polyethylene

t_s	=	Qualified life @ Service Temperature
E_a	=	Activation Energy = 1.13 eV [17]
t_a	=	Test Time = 54 days (Reference 20)
T_s	=	Normal Service Temperature = 120°F + 5°C +20°F (338.15K) (bounds +5°C rise on 135°F in Steam Chase/Drywell, no panels in these areas)
T_a	=	Test Temperature = 268°F (404.26 K) (Reference 20)
K_b	=	Boltzmann's Constant (8.617 x 10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 54 \times \exp\left[\frac{1.13}{8.617x10^{-5}}\left(\frac{1}{338.15} - \frac{1}{404.26}\right)\right] = 83.9 \text{ years}$$

179.1°F

Extrapolating the thermal degradation experienced from a post LOCA condition in the Drywell or post-LOCA heatup in the Reactor Building needs to be determined. From Appendix 1, the post LOCA condition in the Drywell was stated to be 175°F for 195 days. However, the worst-case Reactor Building post-LOCA heat-up condition (with temperature rise effect) is ~~160.4°F~~

~~(Volume 9)~~ + 16°C (SBGT heaters), or ~~189.2°F~~. As such, extrapolating 198 days at ~~189.2°F~~ will bound all installations of the GE SI-57275 cable type throughout the plant. This will be determined at the previously considered normal ambient as follows:

179.1°F + 16°C = 207.9°F (370.87K)

t_s	=	Qualified life @ Service Temperature
E_a	=	Activation Energy = 1.13 eV [17]
t_a	=	Accident Time = 198 days (see text above)
T_s	=	Normal Service Temperature = 120°F +5°F + 20°F (338.15K)
T_a	=	Accident Temperature = ~~189.2°F (360.48K) (see text above)~~
K_b	=	Boltzmann's Constant (8.617 x 10^{-5} eV/K)

16.6

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 198 \times \exp\left[\frac{1.13}{8.617x10^{-5}}\left(\frac{1}{338.15} - \frac{1}{\cancel{360.48}}\right)\right] = \cancel{6.0}\text{ years}$$

Thus, subtracting the thermal aging due to a post LOCA environment from the thermal life yields a result of ~~77.9 years (83.9 Years – 6.0 Years)~~. As shown, the SI-57275 cable with XLPE insulation is qualified for the 60 year life of the plant, considering license renewal.

67.3 years (83.9 - 16.6)

SI-58109

Required:	60 Years
Reported:	>60 Years

The only non-metallics on the SI-58109 is cross linked polyethylene (XLPE) insulation and the Neoprene jacket. From Reference 17 Appendix B, the most conservative listed activation energy for XLPE is 1.13eV. The Neoprene jacket does not support the safety function of the cable to maintain class 1E circuit integrity. Using this information and the Arrhenius equation, the service life is as follows:

ambient remains valid, PLHU for this cable is addressed in Appendix 1. Short operating time basis remains bounding for EPU.

Cross Linked Polyethylene

t_s	=	Qualified life @ Service Temperature
E_a	=	Activation Energy = 1.13 eV [17]
t_a	=	Test Time = 54 days (Reference 20)
T_s	=	Normal Service Temperature = 138.3°F (332.20K) (Ref. [71])
T_a	=	Test Temperature = 268°F (404.26K) (Reference 20)
K_b	=	Boltzmann's Constant (8.617 x 10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 54 \times \exp\left[\frac{1.13}{8.617x10^{-5}}\left(\frac{1}{332.20} - \frac{1}{404.26}\right)\right] = 168\text{ years}$$

In conclusion, the SI-58109 cable has demonstrated a thermal life at 138.3°F of 168 years which is sufficient to last through the plant life of 60 years (with a renewed license).

SI-58081

Required:	60 Years
Reported:	>60 Years

For cable model SI-58081, the qualification test reports a thermal aging test of 340 hours @ 212°F. The following non-metallics comprise the SI-58081. The activation energies were determined by Wyle in [44] from a literature search to determine the required thermal aging test times/temperatures.

Material	Use	Activation Energy
Polyethylene	Inner conductor	1.14
PVC-Flamenol	Outer Conductor	1.15
PVC-Flamenol	Jacket	1.15

The Arrhenius methodology [17] can be used to calculate the equivalent life of the limiting cable material at ~~90°F (maximum~~ normal ambient of Reactor Building, excluding Volumes 16 and 28-32), as follows:

95°F (308.15K)

Polyethylene

t_s = Qualified life @ Service Temperature
E_a = Activation Energy = 1.14 eV
t_a = Test Time = 340 Hours (Reference 44)
T_s = Normal Service Temperature = ~~90°F (305.37K)~~
T_a = Test Temperature = 212°F (373.15K) (Reference 44)
K_b = Boltzmann's Constant (8.617 x 10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 340 \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{\cancel{305.37}} - \frac{1}{373.15}\right)\right] = \cancel{101}\ \text{years}$$

68.6

PVC-Flamenol

t_s = Qualified life @ Service Temperature
E_a = Activation Energy = 1.15 eV
t_a = Test Time = 340 Hours (Reference 44)
T_s = Normal Service Temperature = ~~90°F (305.37K)~~
T_a = Test Temperature = 212°F (373.15K) (Reference 44)
K_b = Boltzmann's Constant (8.617 x 10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 340 \times \exp\left[\frac{1.15}{8.617x10^{-5}}\left(\frac{1}{\cancel{305.37}} - \frac{1}{373.15}\right)\right] = \cancel{108}\ \text{years}$$

73.3

~~Compensating for the effects of thermal aging due to a Post LOCA heatup in the Reactor Building, the Arrhenius methodology will be applied to the Polyethylene as it is the limiting material of construction. The worst-case post-LOCA heat-up occurs in the Torus area~~

~~(Volumes 9-12) with a peak temperature of 160.4°F. This event in these areas gradually reaches the peak, then begins declining. The post-LOCA heat-up event in the RHR rooms, however, is mostly a sustained temperature at 145°F for the full 180-day operability period. After 12 days, the post-LOCA heat-up event in Reactor Building Volumes 9 to 12 is thus bounded by the conditions of the RHR room. As such, considerations for post-LOCA heat-up are taken as 12 days at 160.4°F, followed by 186 days at 145°F for conservatism. These time/temperature conditions will be equated to the normal ambient temperature of 90°F. No temperature rise effects are considered for this control cable type.~~

~~Polyethylene~~

~~t_s = Qualified life @ Service Temperature~~
~~E_a = Activation Energy = 1.14 eV~~
~~t_{a1} = First accident time = 12 days~~
~~t_{a2} = Second accident time = 186 days~~
~~T_s = Normal Service Temperature = 90°F (305.37K)~~
~~T_{a1} = First accident temperature = 160.4°F (344.48K)~~
~~T_{a2} = Second accident temperature = 145°F (335.93K)~~
~~K_b = Boltzmann's Constant (8.617 x 10^{-5} eV/K)~~

$$t_s = t_{a1} \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_{a1}}\right)\right] + t_{a2} \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_{a2}}\right)\right]$$

$$= 12 \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{305.37} - \frac{1}{344.48}\right)\right] + 186 \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{305.37} - \frac{1}{335.93}\right)\right] = 30.7 \text{ years}$$

~~Qualified Life = Equivalent Thermal Life - Accident Life~~

~~70.3 Years = 101 Years – 30.7 Years~~

~~Therefore, the qualified life of the SI-58081 cables in greater than 60 years.~~

The qualified life above exceeds plant operation of 60 years under an extended license period. Combined with the HELB analysis in Appendix 1, the GE cable type SI-58081 is shown to be qualified for greater than 60 years life and Reactor Building HELB conditions.

Post-LOCA Operation

The worst-case PLHU temperature for all of the Reactor Building occurs in the Torus area (RB Volume 10), gradually reaching a peak temperature of 179.1°F. By review of all RB post-LOCA heat temperatures (CA-08-085), the following conservative representation of the PLHU temperature conditions can be made:

Duration Temperature
0 through 7 days: 180°F
7 through 23 days: 150°F
23 through 198 days: 131°F

(PLHU analysis with this profile is amended on the next page)

CA-98-017 GE Cable type SI-58081 PLHU analysis for EPU

All following analyses on polyethyelene with an activation energy of 1.14eV.

Thermal Aging

Duration	Temperature		Equivalent @ 95°F
340 hours	212 °F	373.1478 K	68.64461 years
395 hours	212 °F	373.1478 K	79.74889 years

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature		Equivalent @ 95°F
7 days	180 °F	355.37 K	5.75647 years
16 days	150 °F	338.7033 K	2.106698 years
175 days	131 °F	328.1478 K	6.559286 years
		total	14.42245 years

PLHU Qualified life is then 65.32643 years ← 79.7 minus 14.4

For post-LOCA operation, a revised qualified life can be determined when considering the additional aging imparted to the cable specimens during accident testing. Using only the last 55 hours of accident testing at 217°F, a modified thermal aging period of 395 hours at 212°F can conservatively be established.

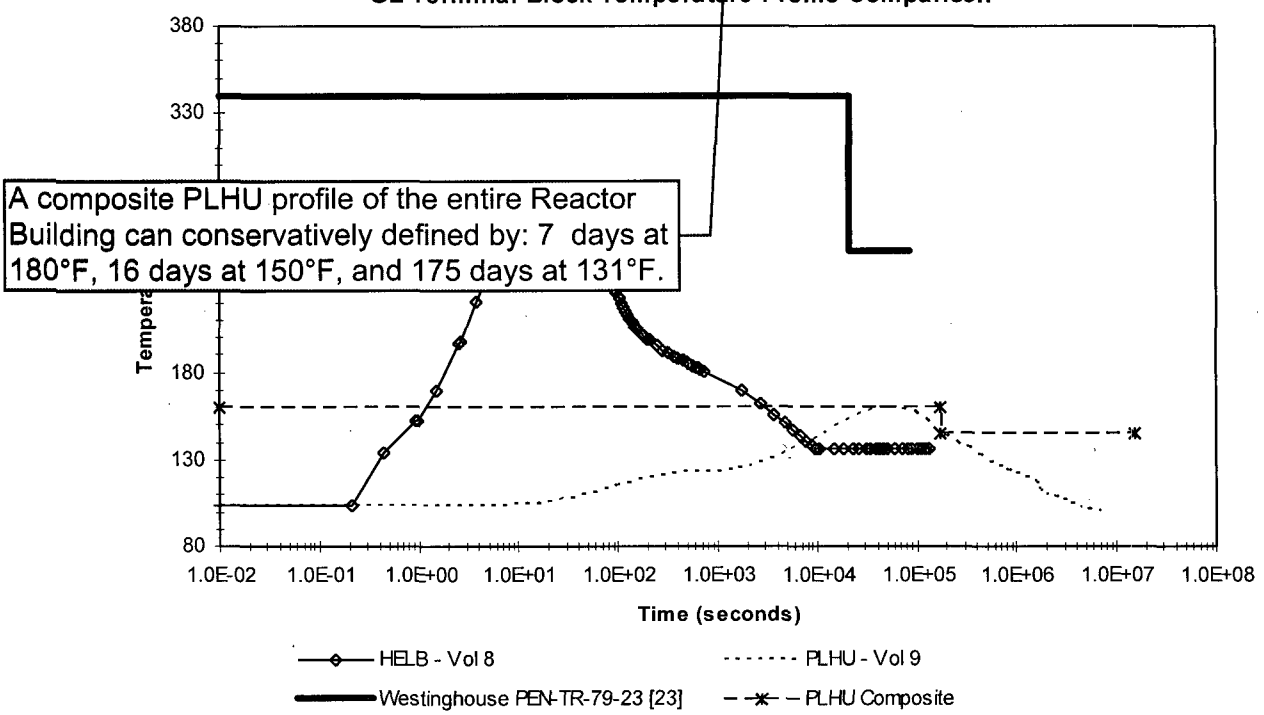
The fundamental basis for allowing the modified thermal aging period for PLHU operation is that both a HELB and design basis LOCA will not occur simultaneously. Thus the actual thermal aging test conditions of 340 hours at 212°F followed by the accident test conditions of 59 minutes at 230°F followed by 55 hours at 217°F demonstrate qualification for greater than 60 years in the Reactor Building plus HELB conditions (excluding RB Volumes 16 and 28 to 32) as illustrated in Appendix 1 of the EQ file. While the modified thermal aging period using the 340 hours at 212°F plus an additional 55 hours (conservatively reduced to 212°F from 217°F) can be utilized to demonstrate a qualified life of greater than 60 years plus 180 days (198 days with margin) service for PLHU conditions in the Reactor Building.

APPENDIX 1
ACCIDENT DEGRADATION EQUIVALENCY

is worst-case.

The Reactor Building HELB condition for Volume 8 ~~and post-LOCA heat-up temperature for Volume 9 are shown in Figure 1-1 along with the LOCA test profile from Reference [23]. In the figure, a composite post-LOCA heat-up profile is also shown consisting of a 2-day period at 160.4°F and 178 days at 143.8°F (Volume 3, 145°F for conservatism). This composite profile bounds the worst-case peak post-LOCA heat-up condition of Volume 9, plus the sustained post-LOCA heat-up temperature of less than 145°F occurring in the RHR pump rooms (Volume 1 or 3) [24].~~

Figure 1-1
GE Terminal Block Temperature Profile Comparison



The test profile [23] envelops the worst-case HELB condition outside the Steam Chase. The composite post-LOCA heat-up profile is evaluated to an equivalent duration at the 250°F test condition using Arrhenius methods to assess post-LOCA operability.

Arrhenius equation is given as follows:

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

Appendix 1 (Continued)

Where:

t_a = Equivalent Accident Time
t_t = Time duration for each profile step
E_a = Activation energy - (1.05 eV) [8, B-7]
K = Boltzmann's Constant (8.617 x 10^{-5} eV/K)
T_a = Equivalent Temperature = 250°F (394.26K)
T_t = Profile step Temperature ~~160.4°F (344.48K) and 145°F (335.93K)~~

~~Using this equation for each step of the composite accident profile yields:~~

~~2 days @ 160.4°F (344.48K)~~	~~= 0.55 hours @ 250°F (394.26K)~~
~~178 days @ 145°F (335.93K)~~	~~= 19.95 hours @ 250°F (394.26K)~~
~~Total Equivalent Time~~	~~= **20.5** hours @ 250°F~~

These results show that the test profile [23], lasting 24-hours, envelops 180 days of the composite post-LOCA profile. Based on this the GE terminal blocks are qualified for the accident condition.

see analysis on next page

CA-98-021 GE CR151 D Terminal Blocks (DOR)

Testing for these terminal blocks included a last stage accident test of 24 hours at 250°F

All following analyses use an activation energy of 1.05 eV.

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature			Equivalent @ 250°F
7 days	180 °F	355.37 K		5.71 hours
16 days	150 °F	338.7033 K		2.41 hours
175 days	131 °F	328.1478 K		8.30 hours
			total	16.42 hours

Therefore, testing of 24 hours at 250°F will bound the EPU PLHU conditions of the Reactor Building

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Appendix 2 - Thermal Aging/Qualified Lives

The Arrhenius Methodology was used for all calculations in the evaluation. The equation used for calculations is based on the Arrhenius equation from Reference [19]. The Arrhenius equation is:

$$t_a = t_t \times \exp\left\{\frac{E_a}{K_b}\left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right\}$$

Where:

t_a = Thermal aging life at Temperature (T_a)
t_t = Test exposure time
E_a = Activation energy (eV)
K_b = Boltzmann's constant (8.617×10^{-5} eV/K)
T_a = Specified temperature (Kelvin)
T_t = Test temperature (Kelvin)

Table 1 below lists all of the non-metallic materials which could be susceptible to thermal degradation. The Arrhenius methodology' will be utilized to evaluate the thermal degradation for the materials during normal service and during the accident operation (See Appendix 1).

Thermal aging of the polyester varnish will not be addressed since Reference [5] states that "deterioration of the varnish will not impair transformer operation." The cellulose acetate paper is used as a barrier to dirt and other contaminates and as a secondary partial backup insulator for the copper wires. The cellulose acetate therefore will not be analyzed for degradation since it will not affect transformer operation.

124.2°F (51.3°C)

Self heating during operation is a factor for transformers. Reference [16] indicates that the Hevi-Duty Model SZO transformer has a 31.5°C temperature rise and a 2°C hot spot allowance when loading on the secondary coils is 3.3 Amps @ 120 Volts. Since the normal and accident ambient temperature is a maximum of ~~112.8°F (44.9°C~~), this value will be used as a normal maximum operating temperature of the transformer:

Max accident temperature + Heat Rise + Hot Spot = Max operating Temperature

(~~44.9°C~~) + (31.5°C) + (2°C) = ~~78.4°C = 351.55K~~ 84.8°C (357.95K)

The above listed load is the maximum load on the SBGT system (Reference [18]). The following evaluation will be conservatively based on the transformer being continuously energized at the maximum load of 3.3 Amps @ 120 Volts. This is conservative since the normal operation of the system is at 2.3 Amps @ 120 Volts, and the higher load is applied during system testing for only 720 hours a year (Reference [18]).

Substituting in the Arrhenius equation:

$$60{,}000 = 6{,}000 \times \exp\left\{\frac{E_a}{8.617x10^{-5}\,\text{eV/K}}\left(\frac{1}{393.15\text{K}} - \frac{1}{423.15\text{K}}\right)\right\}$$

$$E_a = 1.09\,\text{eV}$$

Acrylic:

From Reference [17, Page 2]: 100,000 hours @ 140°C (284°F) = 413.15K
70 hours @ 210°C (410°F) = 483.15K

$$100{,}000 = 70 \times \exp\left\{\frac{E_a}{8.617x10^{-5}\,\text{eV/K}}\left(\frac{1}{413.15\text{K}} - \frac{1}{483.15\text{K}}\right)\right\}$$

$$E_a = 1.78\,\text{eV}$$

Therefore, the activation energy for the non-metallics are:

Formvar = 1.09 eV
Acrylic = 1.78 eV

The transformer will be evaluated at the maximum temperature to determine a conservative qualified life. The qualified life of the transformer can be calculated as:

84.8°C (357.95K)

For the Acrylic wire:

Substituting values in the Arrhenius equation to determine service life at the maximum operating temperature of ~~78.4°C (351.55K)~~ gives:

$$t_a = 100{,}000 \times \exp\left\{\frac{1.78}{8.617x10^{-5}\,\text{eV/K}}\left(\frac{1}{\cancel{351.55\text{K}}} - \frac{1}{413.15\text{K}}\right)\right\}$$

$$>> 100 \text{ years}$$

For the Formvar magnet wire:

Substituting values in the Arrhenius equation to determine service life at the maximum operating temperature of ~~78.4°C (351.55K)~~ gives:

$$t_a = 60{,}000 \times \exp\left\{\frac{1.09}{8.617x10^{-5}\,\text{eV/K}}\left(\frac{1}{\cancel{351.55\text{K}}} - \frac{1}{393.15\text{K}}\right)\right\}$$

357.95

$$>> 100 \text{ years}$$

Therefore, since the available thermal life normalized to the maximum temperature the transformer experiences is greater than the 60 year plant life plus the 180 day accident duration (198 days with margin), the transformer is qualified for thermal aging and accident aging for this application.

Polyester Varnish
This material is used for motor insulation.

Material	Activation Energy	Thermal Capabilities
Polyester Varnish	1.04 eV [25]	20,000 hrs. @ 356°F [18]

The listed activation energy is the lowest value found in Reference [25] for ungelled polyester resin. This data is conservative since this form of polyester is inferior to the gelled type. The thermal capability data is taken from testing performed by the Navy on materials used in the same application, i.e., motor coils.

Neoprene
Reference [29] confirms that due to the similar vintage of various motors used throughout MNGP, the power lead insulation for the motors would be neoprene. Aging data contained in Reference [37] states that Neoprene has a maximum continuous operating temperature of 70°C. Based upon the Arrhenius plots contained in Reference [37], neoprene has a thermal life of 40 years at 40°C with 60% retention of initial elongation. The 40°C (104°F) is well above the normal ambient temperature of ~~80~~°F [2] for the SBGT Room.

90°F

Alkanex (Polyester)
This material is used for insulation on the coil wire.

Material	Activation Energy	Thermal Capabilities
Alkanex (Polyester)	1.0 eV [25]	20,000 hrs. @ 356°F [18]

The thermal capability data is taken from testing performed by the Navy on materials used in the same application, i.e., motor coils.

Lubricant (Mobil EP-2)
Lubricant qualification is not addressed in this analysis. Generally, lubricants are routinely replaced during plant preventive maintenance activities. Reference [30] addresses lubricants used at MNGP.

Thermal Aging Calculations
The Arrhenius Method will be used to evaluate the life of each material at the service temperature of a non-energized motor during normal operation and at running temperatures during normal plant life and accident conditions. The motor is conservatively assumed to be run during plant life for testing and operational concerns for 24 hours per month.

The motor data sheet [22] identifies the insulation as Class B. The standard heat rise of the windings in a Class B insulated motor is 90°C [26]. The normal ambient temperature in the SBGT room is ~~80°F (26.7°C)~~. The running motor temperature, during normal plant life of ~~80~~°F

90°F (32.2°C)

ambient, is ~~117°C (90°C + 26.7°C)~~. The total number of assumed running hours during 60 years of plant life is ~~17,280 (24 running hours/month x 60 years x 12 months/year~~).

124.2°F

For post-accident operation, the peak temperature of ~~112.7°F~~ in Volume 37 for post-LOCA heatup will be conservatively assumed for the entire 180 days of operation. The resulting motor temperature at an ambient of ~~112.7°F is 134.8°C (44.8°C + 90°C)~~.

The qualified life is given as:

Qualified Life (Motor not running) = Service Life - 60 yr Normal Condition (Motor Running) - 180-days of Accident Condition

122.2°C (90°C + 32.2°C)

Using the previous data for each material, the equivalent life for each condition can be calculated using the Arrhenius Method as described below.

see Note next page , duration reduced to 10,800 hours

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

Where:

t_a = Qualified Life
t_t = Service Life at service temperature
E_a = activation energy - (eV)
K = Boltzmann's Constant (8.617×10^{-5})
T_a = Service temperature – (°K)
T_t = Aging temperature – (°K)

124.2°F is 141.2°C (51.2°C + 90°C)

In Table 3-1 the thermal data for each material is extrapolated to a service life at a temperature of ~~135°C~~. The accident time of 4320 hours is subtracted from this life to determine the remaining available service life.

Table 3-1

Material	Activation Energy	Thermal Capabilities	Service Life @ ~~135~~°C (hrs)	Remaining Service Life Hrs @ ~~135~~°C after subtracting 4320 hrs for Accident Conditions
Dacron	1.15 eV	10,000 hrs. @ 302°F	~~31,865~~ 19,536	~~27,545~~ 15,216
Mylar	1.18 eV	10,000 hrs. @ 302°F	~~32,843~~ 19,880	~~28,523~~ 15,560
Polyester Glass	0.87 eV	5,000 hrs. @ 300°F	~~11,284~~ 7,793	~~6,964~~ 3,473
Polyester Varnish	1.04 eV	20,000 hrs. @ 356°F	~~376,941~~ 24,217	~~372,621~~ 19,897
Alkanex (Polyester)	1.0 eV	20,000 hrs. @ 356°F	~~336,686~~ 22,002	~~332,366~~ 17,682

122.2°C

90°F

In Table 3-2 the remaining service life is extrapolated for the normal 60-year life with the motor running ~~(117°C~~) and the motor idle ~~(80°~~F).

Table 3-2

Material	Equivalent Life @ ~~117~~°C for Remaining Service Life (Hrs)	Remaining Service Life @ ~~117°C~~ after subtracting ~~17,280~~ hrs for Motor Running Conditions (Hrs)	Qualified Life @ ~~80°~~F for Remaining Service Life (after subtracting normal running and accident conditions) (Yrs)
Dacron	~~124,654~~ 71,539	~~107,374~~ 60,739	>100
Mylar	~~134,265~~ 76,172	~~116,985~~ 65,372	>100
Polyester Glass	~~21,827~~ 11,202	~~4,547~~ 402	~~>100~~ 84
Polyester Varnish	~~1,461,227~~ 80,672	~~1,443,947~~ 69,872	>100
Alkanex (Polyester)	~~1,236,644~~ 67,934	~~1,219,364~~ 57,134	>100

Therefore, the Fan Motors are qualified for greater than 60 years plus the post-accident operating time of 180 days.

Note, assumed run-time during normal plant operation can be reduced based on Technical Specification Surveillance Requirement in Section 3.6.4.3.1, the SGTS must be tested for ≥10 hours every 31 days. Thus, the former assumed duration of 17280 hours is reduced to 10800 hours over 60 years (15 hour/month) and still have conservatism.

(D.3) Namco Model EA180 Series Limit Switches Without EC210 Receptacles

Plant ID: AO-2377, 2378, 2379, 2380, 2381, 2383, 2386, 2387, 2541A/B, 2561A/B, 2896, AO-2982, CV-2791, 3267, 3268, 3269, 3311, 3312, 3313, 3314, and 7956

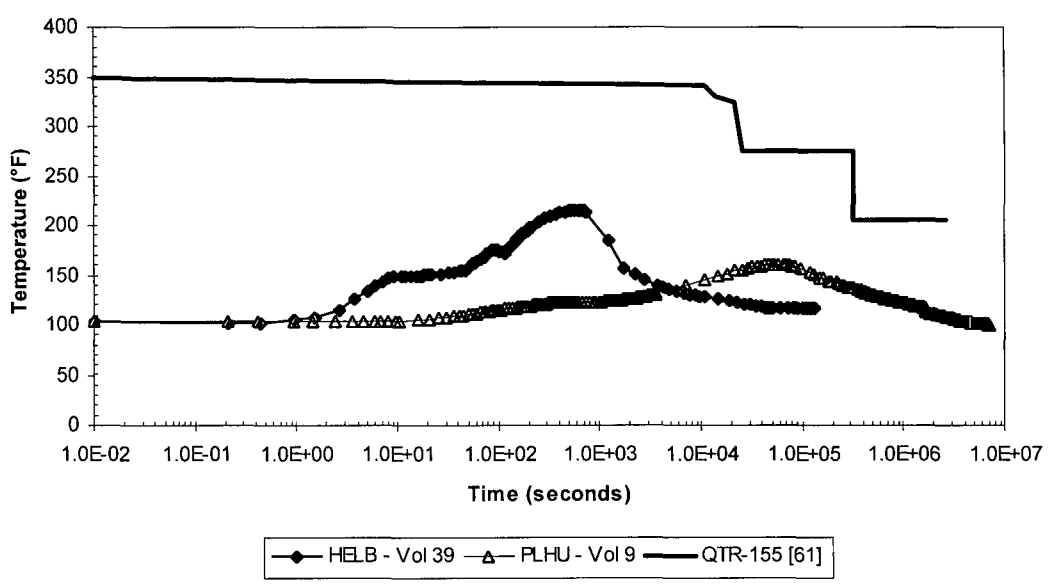
Plant Location: Reactor Building Volumes 9, 11, 12, 18, 19, 31, and 39

Models: Namco EA180-14302, EA-15302, and EA180-24302

Test Reports: Namco QTR-105 [4] or QTR-155 [61], per Appendix 2

Although these switches do not mitigate HELB events (except for CV-2791 in RB Volume 31), the worst-case HELB event in RB Volume 39 is considered for conservatism. The peak HELB temperature is 215.3°F for this area. The worst-case post-LOCA heat-up temperature is ~~160.4°F (RB Volume 9)~~ 179°F (RB Volumes 9, 11, 12). Depending on B/M code of the switch, the limit switches are qualified by Namco Reports QTR-105 [4] or QTR-155 [61]. As illustrated in Appendix 2, the conditions of Namco Report QTR-155 [61] are minimal amongst the testing for accident temperature qualification of the Model EA180 series limit switches without an integral EC210 receptacle. The accident testing [61] versus RB Volume 39 HELB and post-LOCA heat-up [37] are shown in Figure 3. The test curve begins at the peak temperature of the second transient of the dual transient test [61, Page 8-53].

Figure 3
Namco Model EA180 Without EC210 Receptacles
Versus Reactor Building HELB and Post-LOCA Heat-Up Conditions



As shown, the peak HELB and post-LOCA temperature conditions are bounded by the test. To demonstrate that the Model EA180 limit switches without EC210 receptacles will survive

post-accident operation, ~~an extrapolation of the test time (84 hours) at 275°F (408.15K)~~ is made using ~~the following~~ Arrhenius analysis:

a comparison of the final test duration of 26 days at 205°F (369.25K)

~~Letting,~~

~~t_e = Equivalent operating time in years at T_e~~
~~t_a = Test duration = 84 hours (i.e. time period at 275°F)~~
~~T_e = Service temperature = 160.4°F (344.48K)~~
~~T_a = Test temperature = 275°F (408.15K)~~
~~E_a = Activation energy = 0.8 eV [61, Page 6-5]~~
~~K_b = Boltzmann's constant (8.617×10^{-5} eV/K)~~

See EPU evaluation next page

~~Then,~~

$$\cancel{t_e = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_e} - \frac{1}{T_a}\right)\right] = 84\text{ hours} \times \exp\left[\frac{0.8}{8.617x10^{-5}}\left(\frac{1}{344.48} - \frac{1}{408.15}\right)\right] = 234\text{ days}}$$

~~Thus, the limit switches are qualified for accident and post-accident operation for greater than the required 180 days (198 days with margin). It should be noted that the last 26 days of testing at 205°F were not used in the analysis above, this simply adds additional margin.~~

(D.4) Namco Model EA180 Series Limit Switches With EC210 Receptacles

Plant ID: CV-2790

Plant Location: Drywell (Volume 49)

Models: Namco EA180-31602 and EA-32602

Test Report: Namco QTR-157 [79]

These limit switches are installed inside the Drywell. The accident testing and Drywell composite accident [37] are shown in Figure 4. The test curve begins at the peak temperature condition of the second transient of the dual transient test [79, Page 8-73 through 8-78, and Page 8-90].

Not relevant to the Model EA180 without EC210 receptacles.

CA-98-032 Namco EA 180 Limit Switch without an EC210 connector

Testing for these limit switches included a last stage accident test of 26 days at 205°F

All following analyses use an activation energy of 0.8 eV.

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature			Equivalent @ 205°F
7 days	180 °F	355.37 K		2.62 days
16 days	150 °F	338.7033 K		1.66 days
175 days	131 °F	328.1478 K		7.50 days
			total	11.78 days

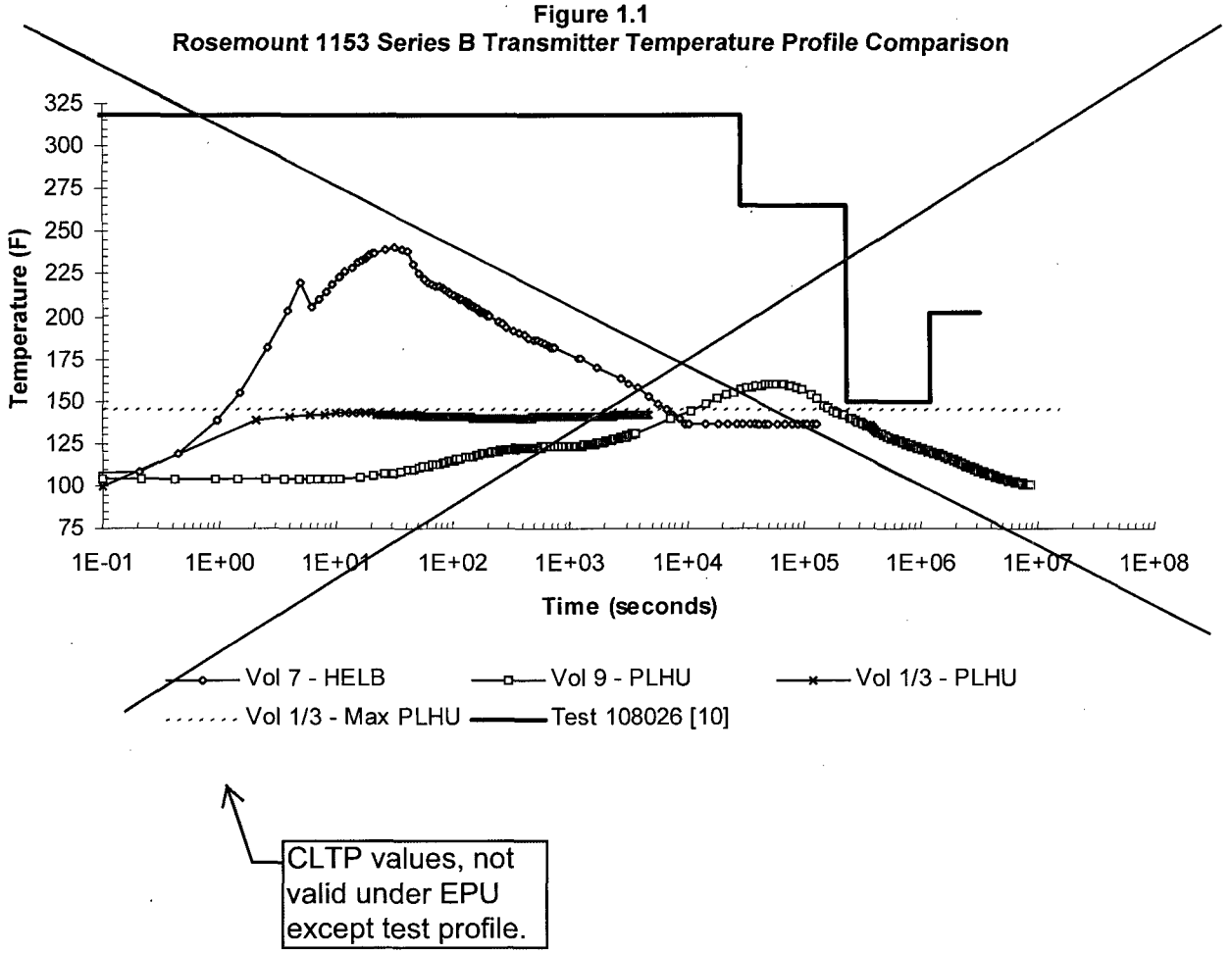
Therefore, testing of 26 days at 205°F will bound the EPU PLHU conditions of the Reactor Building

APPENDIX 1
ACCIDENT DEGRADATION EQUIVALENCY

Rosemount 1153 Series B Transmitter Accident Degradation Equivalency Analysis

The first 64-hours of the test profile [10, App. II, p.110 (Reference [5] summarizes the actual testing of Reference [10])] completely envelops the worst-case Reactor Building HELB profile (of Volumes 1, 3, 7, 9, 12, 14, 17, 18, 19, 22, 27, 33, and 34). Following the accident test, the transmitter specimens were exposed to 14 days at 150°F, then 23 days at 203°F [10, Section 16]. The Reactor Building HELB events are short lived and easily bounded by the dual transient accident test as shown in Figure 1.1 (the test profile begins at peak conditions of second transient). ~~Worst-case post-LOCA heat-up temperature is 160.4°F [49] in Reactor Building Volume 9, but long-term post-LOCA heat-up temperature does not exceed 145°F (bounds Reactor Building Volumes 1/3)~~. As shown in Figure 1.1, the plant conditions (HELB ~~and post-LOCA heat-up)~~ are fully bounded by the 64 hour + 14 days of testing. The Arrhenius method [1] is used to extrapolate the last 23 days of testing at 203°F to the post accident operating time of 198 days.

Figure 1.1
Rosemount 1153 Series B Transmitter Temperature Profile Comparison



Arrhenius equation is given as follows:

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

See next page for EPU evaluation of Reactor Building bounding PLHU conditions.

Where:

t_a = Equivalent Accident Time @
t_t = Time duration = 23 days [10, Section 16]
E_a = Activation energy - (0.78 eV) [5, p.29]
K = Boltzmann's Constant (8.617 x 10^{-5})
T_a = Plant Temperature 145°F(335.93K)
T_t = Test Temperature = 203°F (368.15K)

Using the above equation on the test profile data (last 23 days at 203°F) yields:

$$t_a = 23\text{ days} \times \exp\left[\frac{0.78}{8.617\text{x}10^{-5}}\left(\frac{1}{335.93} - \frac{1}{368.15}\right)\right] = 243\text{ days}$$

These results show that the Rosemount Series B transmitters are qualified for 198 days of worst-case accident conditions.

CA-98-040 Rosemount Model 1153 Series B

Testing for these limit switches included a last stage accident test of 23 days at 203°F

All following analyses use an activation energy of 0.78 eV.

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature			Equivalent @ 203°F
7 days	180 °F	355.37 K		2.89 days
16 days	150 °F	338.7033 K		1.89 days
175 days	131 °F	328.1478 K		8.74 days
			total	13.52 days

Therefore, testing of 23 days at 203°F will bound the EPU PLHU conditions of the Reactor Building

Appendix 1 - Accident Degradation Equivalency

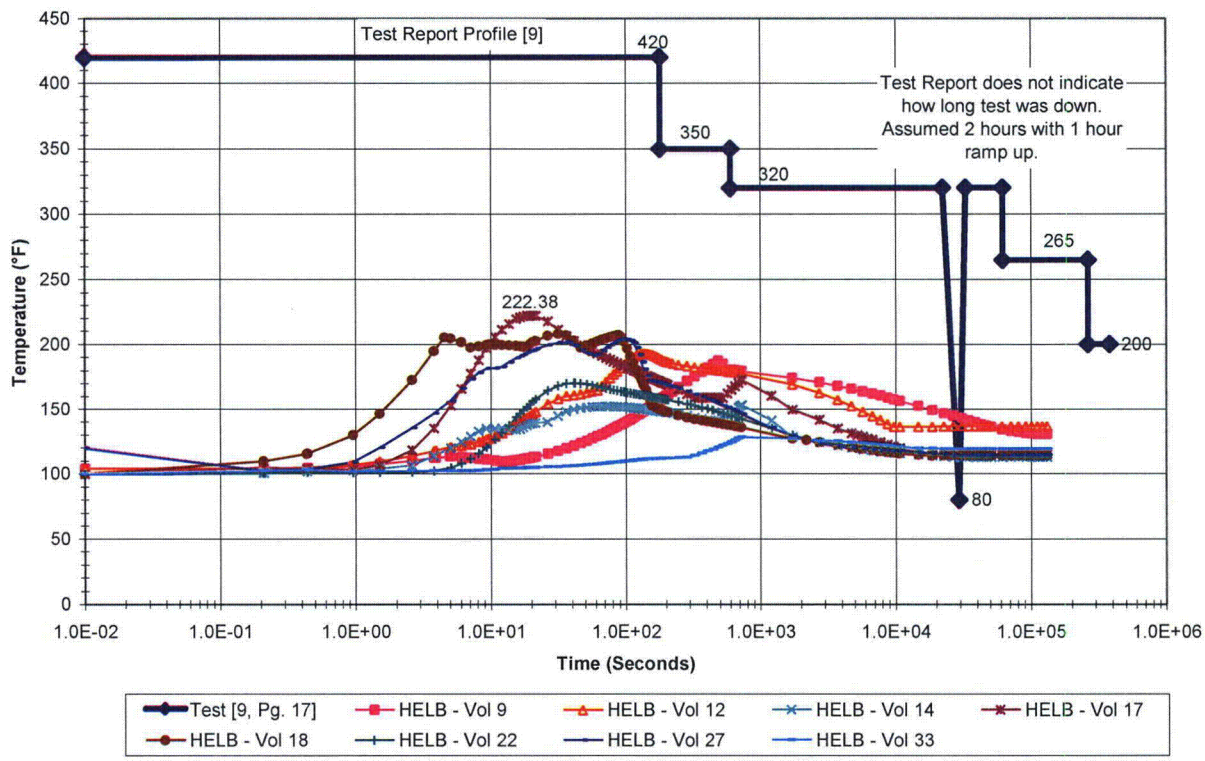
Required: 180 Days + 18 Days (Margin) = 198 Days

Reported: 102 hours LOCA Test

for the entire Reactor Building can conservatively be defined as 7 days at 180°F, 16 days at 150°F, and 175 days at 131°F.

The LOCA test profile [9, Page 17] was plotted against the composite accident profile [21] where the conduit seals are installed. The test profile fully envelopes the composite HELB accident profiles for Reactor Building Volumes: 9, 12, 14, 17, 18, 22, 27 and 33. The worst-case post-LOCA Heatup condition ~~is 160.4°F occurring in Reactor Building Volume 9 [21].~~ To satisfy the post accident operating time requirement of 198 days, the test conditions of 56 hours at 265°F will be evaluated using Arrhenius methods. Although the test included a post-HELB aging simulation of 32 hours at 200°F, this part of the test is treated as margin.

Rosemount Conduit Seals Test Report vs. Accident Conditions (Temperature)



As shown, the peak HELB conditions are bounded by the test. To demonstrate that the conduit seals will survive post-LOCA operation, ~~an extrapolation of~~ the test time (56 hours) at 265°F (402.59K) is made using the following Arrhenius analysis:

comparison of the composite Reactor Building PLHU profile to

Appendix 1 - Accident Degradation Equivalency (continued)

Letting,

t_e = Equivalent operating time in years at T_e
t_a = Test duration = 56 hours (i.e. time period at 265°F)
T_e = Service temperature ~~= 160.4°F (344.48K)~~ see analysis next page
T_a = Test temperature = 265°F (402.59K)
E_a = Activation energy = 0.97 eV (see Appendix 2)
K_b = Boltzmann's constant (8.617×10^{-5} eV/K)

Then,

$$\cancel{t_e = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_e} - \frac{1}{T_a}\right)\right] = 56\text{ hours} \times \exp\left[\frac{0.97}{8.617x10^{-5}}\left(\frac{1}{344.48} - \frac{1}{402.59}\right)\right] = 260\text{ days}}$$

Thus, the Rosemount conduit seals are qualified for accident and post-accident operation for greater than the required 180 days (198 days with margin). It should be noted that the last 32 hours of testing at 200°F were not used in the analysis above, this simply adds additional margin.

CA-98-041 Rosemount 353C Conduit Seals

Testing for these seals included accident testing of 56 hours at 256°F

All following analyses use an activation energy of 0.97 eV.

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature			Equivalent @ 256°F
7 days	180 °F	355.37 K		5.81 hours
16 days	150 °F	338.7033 K		2.80 hours
175 days	131 °F	328.1478 K		10.50 hours
			total	19.11 hours

Therefore, testing of 56 hours at 256°F will bound the EPU PLHU conditions of the Reactor Building

APPENDIX 1
ACCIDENT EVALATION / THERMAL AGING

Accident Evaluation

The installed level switch is a 4418EC, which includes a transmitting coil and armature assembly for remote indication. However, the remote indicating feature is not used on these switches. The 4418EC is identical to the tested 4418TC for the required switch function. The installed Yarway 4418EC level transmitter uses two mercury switches rather than a snap-action switch as in the test unit [7]. The Yarway level indicating switches are only exposed to a post-LOCA temperature of less than ~~110~~°F through the assumed 180-day operating time. This period will be address under the thermal aging analysis. The switches are not required for mitigating any Reactor Building HELB. Furthermore, the switches only provide a permissive signal, containment spray will not actuate if the Yarway switches fail during a Reactor Building HELB [22 through 24].

121.3°F

Thermal Aging Analysis

Since there is no test data available for this equipment, thermal aging was addressed by evaluating the critical non-metallic materials of construction [17] susceptibility to thermal degradation. Indicator 4400 Series, Items # 1, 1A, 3A, 3B, and 9 of the 09/08/1980 non-metallic materials list (included in [17]) are not critical materials for the function of the level indicating switch in a radiation only harsh environment per review of Yarway drawing 021-934800 [17] and Reference [18]. The 4400 Series Control module (with mercury switches) contains phenolic terminal block and HT 105 vinyl tubing. A literature search was performed to identify the thermal properties of each non-metallic as evaluated below. Non–metallic components are identified with their thermal aging properties.

Table 3-1
Thermal Aging Analysis of Yarway 4400 Series Level Switch

Component*	Material	Activation Energy (eV)	Ref	Thermal Aging Data (Hrs @ °F)	Ref
Diaphragm (item # 19)	Dacron Polyester EPT**	1.13 1.05	8c 8c	1000 @ 300 1000 @ 250	12 8e
O-Ring (item # 32B)	Buna-N	0.86	8c	1000 @ 225	8e
Back Plate Seal (item # 14)	EPT**	1.05	8c	1000 @ 250	8e
Terminal block	Phenolic	1.05	8c	1000 @ 212	20
Insulating Tube	Vinyl	~~0.5~~	~~See Analysis~~	10,000 @ 221	17
Mercury Switch	Glass/metal	NAS	N/A	N/A	N/A

* Item # for 4400 Series indicator components per non-metallic material list 09/08/1980 [17]
**Ethylene Propylene Terpolymer

1.69 eV, see revised basis provided later in mark-up

Appendix 1 (Continued)

Mercury Switch

The mercury switch is a glass envelop containing the mercury and metallic contacts. These materials are not age-sensitive. However, the lead wires are insulated with the vinyl tubing (see above).

Qualified Life Determination

The equivalent service life of each of the non-metallic materials used in the construction of the Yarway level switch can be determined based on a service temperature ~~80°~~ using the Arrhenius equation:

85°F (302.59K)

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

Where:

t_a = Equivalent aging at ~~80°F~~
t_t = Thermal Life Time
E_a = activation energy - (eV)
K = Boltzmann's Constant (8.617 x 10^{-5})
T_a = Equivalent temperature - °K ~~(80°F)~~
T_t = Thermal Life temperatures - °K

The qualified life can be determined as follows:

Qualified Life = Service Life - Accident Aging

The equivalent years at ~~80°F~~ for each material has been calculated using the equation above and presented in the following table.

Table 3-2
Equivalent Age of Non-Metallic Materials at ~~80°F (299.81K)~~

Material	Ae	Thermal Life (hr)	Thermal Temp (°F)	Thermal Temp (K)	Equivalent Years at ~~80°F~~
Vinyl	~~0.50~~	10,000	221	378.15	~~63~~
Buna-N	0.86	1,000	225	380.37	~~>100~~
Phenolic	1.05	1,000	212	373.15	>100
EPT	1.05	1,000	250	394.26	>100
Dacron Polyester	1.13	1,000	300	422.04	>100

>100

96.8

1.69 (see revised activation energy basis next page)

MONTICELLO NUCLEAR GENERATING PLANT		CA-98-046, Add. 0
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Appendix 1 (Continued)

Buna-N

96.8 years at 85°F

From the table above ~~Vinyl~~ is the limiting material with an equivalent life of ~~63 years @ 80°F~~. The effect of 180-days at ~~110°F (316.48K)~~ for accident operation is determined using the Arrhenius equation. The equivalent of ~~110°F~~ for ~~198~~ days (180 days plus 10% margin) at ~~80~~°F for the weak-link ~~vinyl tubing~~ material is:

121.3°F (322.76K)

$$t_a = 198\text{ days} \times \exp\left[\frac{0.50}{8.617\text{x}10^{-5}}\left(\frac{1}{299.81} - \frac{1}{316.48}\right)\right] = 1.5\text{ years}$$

4.3 years

85°F (302.59K)

The qualified life is equal to:

Qualified Life = ~~63 years - 1.5 years = 61.5 years~~

Therefore, Yarway level switches LIS-2-3-73A/B are qualified for 60 years plus 198 days of post-LOCA operation.

96.8 years - 4.3 years = 92.5 years

Activation energy for the Vinyl sleeve can be updated by using data from Markel (Page 5 of Reference 17): 105°C continuous use (i.e. 10,000 hours) temperature and 400 hours of aging at 130°C. The resulting activation energy is 1.69 eV (see attached System 1000 worksheet). From the data and understanding of UL temperature indices, this above method is justified. UL temperature index, or continuous use temperature is that temperature at which 50% of material changes occur in about 10,000 hours. The thermal aging of 400 hours at 130°C per ASTM 876 methods revealed less than 35% material changes. For this EPU analysis, the next higher activation energy of 0.86 eV is chosen.

RCM Technologies
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System 1000 Revision 17.0.c TTR Report

TIME-TEMPERATURE REGRESSION CALCULATION

Description: No Description Provided

CALCULATION INPUTS

Time to Failure	Temperature
400 Hours	130 Celsius
10000 Hours	105 Celsius

CALCULATION RESULTS

Activation Energy:	1.6901
Slope:	19,613.76919621
Intercept:	-42.67793792
Correlation Coefficient:	1.00000000

RB Vol 19, 121.3°F PLHU

RB Vol 11, 179°F PLHU

APPENDIX 1
ACCIDENT EVALUATION

Solenoid Valves: SV-4001A/B, 4002A/B, 4003A/B, 4004A/B, 4005A/B, 4020A/B, 4081, 4082, 3307, and 3308

RB Vol 19

By review of P&ID drawings NH-91197 [34] and NH-96042-1 [35], these Valcor solenoid valves are part of the H_2/O_2 analyzer systems; and by USAR Table 5.2-3a, they are also containment isolation valves. Modification 03Q145 has removed the Combustible Gas Control system and downgraded the H_2 analyzer to Regulatory Guide 1.97, Category 3 status (environmental qualification not required, see the May 21, 2004 NRC Safety Evaluation Report, ADAMS #ML041180612). Thus, for environmental qualification purposes, these solenoid valves now only serve as containment isolation valves and perform a Regulatory Guide 1.97 Containment Isolation Valve Position Indication function under a design basis LOCA condition (Group 2 isolation per USAR Table 5.2-3b). The solenoid valves are not required to function under HELB events occurring in the Reactor Building. During a design basis LOCA, the Reactor Building (location of the solenoid valves) will only experience relatively moderate post-LOCA heat-up temperature (~~158.7~~°F) ~~and harsh~~ radiation conditions. Therefore, qualification for these parameters is addressed in Appendices ~~2 and 3.~~

179°F

RB Vol 18, 121.3°F PLHU

Solenoid Valves: SV-4234 and SV-4235

TB Vol 21 (new, TB25 old), PLHU unaffected by EPU

These solenoid valves (original Model V526-5295-159) were reconfigured per Modification 93Q305, Part C (film location [MO4493-0007]) to Model V526-5891-54. They now serve as alternate Nitrogen supply isolation valves if there were a system breach inside the Drywell due to large pipe break jet impingement (i.e. large break LOCA). The Alternate Nitrogen system supplies back-up pneumatic pressure for the Safety Relief Valves (SRV) and/or Main Steam Isolation Valves (MSIV) inside the Drywell. With implementation of Modification 93Q305, Part C, solenoid valves SV-4234 and SV-4235 are now normally de-energized and remain de-energized until the Alternate Nitrogen line breach condition. The solenoid valves are not required to change position during HELB events outside the Drywell as the Alternate Nitrogen lines inside the Drywell are not susceptible to these events. Thus, solenoid valves SV-4234/4235 are not required to function under the HELB events at their respective locations (Reactor building Volume 18 and Turbine Building Volume 25). As such, only the post-LOCA heat-up temperature of ~~109.3~~°F and radiation dose for RB Volume 18 and the normal ambient temperature of ~~80°~~F of TB Volume 25 are considered for these valves, see the analyses in Appendices 2 and 3. (See future needs for SV-4235).

Based on the above assessments, the housings do not need to be sealed.

100°F

APPENDIX 3
THERMAL AGING ANALYSIS

The solenoid valves are located in Reactor Building Volumes 11, 18, and 19 and Turbine Building Volume 25. The solenoid valves are not required to function under HELB conditions in these areas. The solenoid valves located in the Reactor Building will be exposed to post-LOCA heat-up conditions while the solenoid valve in the Turbine Building (SV-4235) only experiences normal ambient temperature conditions (see Future Needs for SV-4235). Some solenoid valves in the Reactor Building are installed on lines that are heat-traced, and some valves are normally energized. Since steam conditions are not prevalent during the time in which they are required to function, the qualified life, accident, and post-accident operation of the Valcor solenoid valves can be determined using the Arrhenius Methodology. There are three different situations that must be considered:

- Heat traced valves (SV-3307, 3308, 4001A/B, 4002A/B, 4003A/B, 4020A/B, 4081, & 4082 [34, 35, and 36]). Reference [37] indicates a maximum heat-trace temperature of 291°F.
- Normally energized valves (SV-3307, & 3308 [36])
- Valves that are energized for 180 days (assumed) post accident (SV-4234/4235)

The Valcor solenoid valves addressed by this EQ Calculation include common functional sub-assemblies as: valve seats, solenoid coils, solenoid housing electrical components, and housing seals. The housing seals (O-Rings, Silicone [38] or EPR [29]) on the solenoid valves serve no purpose since the housings are not required to be sealed. As such, these items do not require environmental qualification. The Model V526-5295-67 solenoid valve has Vespel seats [29] while the Model V526-5891-54 has EPR seats [38]. The remaining components of the valve (solenoid coils and solenoid housing electrical components) are similar between the two valve models. Also common between the solenoid valve models, are the standoff spacers between the valve body and solenoid housing ([29] or [38]). These standoff spacers minimize the heat flow from normally energized coils to the valve seats; and, similarly, the heat tracing effects on the valve body into the solenoid housings.

Based on Appendix 4 and References [2], [28], [31], and [32], the following thermal aging parameters are considered for the Valcor solenoid valves (see notes for details):

	Activation Energy	Aging Data	Operating Temperature
Coil	2.096 eV (1)	1773 hours at 420°F (2)	309°F (3)
Solenoid housing electrical components	1.136 eV (1)	3395 hours at 320°F (2)	183-226°F (3)
Vespel seats	Arrhenius plotted data at 50% tensile strength (4)		291°F (4)
EPR seats	1.038 eV (5)	614 hours at 318°F (6)	110°F (7)

Data Sources/Notes:

1) The activation energy for the coil and solenoid housing electrical components is per References [2, Page 13], [31, Page 4], and [32, Table V].

2) Thermal aging data for the coil and solenoid housing electrical components per References [2, Table 3.2.2-III] and [32, Table V].

3) All of the installed solenoid valves are not exposed to hot process media. Solenoid valves SV-3307, 3308, 4001A/B, 4002A/B, 4003A/B, 4004A/B, 4005A/B, 4020A/B, 4081, and 4082 are used for isolating containment atmosphere (Drywell or Torus). Solenoid valves SV-4234 and 4235 isolate the Alternate Nitrogen gas supply. Various temperature rise measurement tests were conducted by Valcor, see References [2, App. VI], [28, Addendum 1], or [32, App. VI]. Of these test, Reference [28] is most relevant as it measured the temperature rise of the coil and housing electrical components while the valve was energized at 72°F, 120°F, and 150°F ambient temperature conditions and no flow through the valve, the other tests had additional influences from hot process media during their tests. The normally energized solenoid valves SV-3307 and 3308 have Drywell atmosphere as the process media. Solenoid valves SV-4234 and 4235, assumed energized during a LOCA, will not have any Alternate Nitrogen process flow as they are energized closed valves. As such, the heat-rise testing in Reference [28, Addendum 1, Page 6 and Table I] is most appropriate for the solenoid valves at Monticello.

4) The Arrhenius aging characteristics for the Vespel valve seats in the Model V526-5295-67 solenoid valves (SV-3307, 3308, 4001A/B, 4002A/B, 4003A/B, 4004A/B, 4005A/B, 4020A/B, 4081, and 4082) is provided in Reference [28, App. IX, Page 27 and App. XII of App. IX]. The operating temperature for the valves seats of these solenoid valves is taken as maximum of either the Containment Atmospheric Monitoring or Post-Accident Sampling system heat-tracing control data per Reference [37]. Although all valves are not heat-traced, the worst-case operating temperature for the Vespel is being applied to all of the Model V526-5295-67 solenoid valves for conservatism.

5) The activation energy for EPR is per Reference 31, Page 4]. The Arrhenius plot for EPR in Reference [28, App. IX, Page 27] yields a higher value. For conservatism, the lower activation from [31] is considered.

6) The thermal aging data for specimen V52600-5291-2 (the one with EPR seat) is considered for solenoid valves SV-4234 and 4235. The thermal aging program for this specimen is explained in Reference [28, Page 21].

7) Solenoid valves SV-4234 and 4235 are located in RB Volume 18 and TB Volume 25, respectively. The worst-case post-LOCA heat-up condition of either valve is ~~109.3°F~~ 121.3°F [1]. As such, a ~~110°F~~ 121.3°F temperature is considered for the EPR valve seat.

Based on the above discussion, the qualified lives of each sub-component (coil, solenoid housing electrical component, and valve seat) can be calculated using the Arrhenius equation as follows:

$$t_a = t_t \times \exp\left[\frac{E_a}{K}\left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]$$

Where:

t_a = Qualified Life
t_t = Aging Time (per table above)
E_a = activation energy - (per table above)
K = Boltzmann's Constant (8.617 x 10^{-5} eV/K)
T_a = Service temperature (per table above)
T_t = Aging temperature (per table above)

Coil – all solenoid valves

85°F to 100°F

179°F

The solenoid valves are located in RB Volumes 11, 18, and 19, and TB Volume 25 where the normal ambient temperature is either ~~80°F or 85~~°F [1]. Under post-LOCA operation, the worst-case ambient temperature occurs in RB Volume 11 of ~~158.7°F~~ [1]. Only solenoid valves SV-3307 and 3308 are energized during normal plant operation while SV-4234 and 4235 are only energized during the post-LOCA period. For conservatism, all solenoid valves are assumed to be continuously energized during normal and post-LOCA operation. Reference [28, Addendum 1] provides a maximum operating temperatures of 309°F (427.04K) for the coil when energized at 120Vac and 150°F ambient conditions. This simulates the conditions postulated for the solenoid valves at Monticello. The thermal aging for the coil per Reference [2, Table 3.2.2-III] or [32, Table V] was 1,773 hours at 420°F (488.70K). The qualified life of the coil is thus:

$$t_a = 1{,}773 \text{ hours} \times \exp\left[\frac{2.096 \text{ eV}}{8.617\text{x}10^{-5} \text{ eV/K}}\left(\frac{1}{427.04\text{K}} - \frac{1}{488.70\text{K}}\right)\right] = 268 \text{ years}$$

Calculation still ok under EPU as valves in Torus that experience the 179°F are de-enerized.

Solenoid Housing Electrical Components – all solenoid valves

The solenoid valves are located in RB Volumes 11, 18, and 19, and TB Volume 25 where the normal ambient temperature is either ~~80°F or 85°F~~ [1]. Under post-LOCA operation, the worst-case ambient temperature occurs in RB Volume 11 of ~~158.7°F~~ [1]. Only solenoid valves SV-3307 and 3308 are energized during normal plant operation while only SV-4234 and 4235 are energized during the post-LOCA period.

RB Vol 19 (121.3°F PLHU)

Delete added conservatism, assumption of all valves energized and experiencing 180°F continuous accident temperature is conservative enough. Calculation remains valid.

There is no combination of energized/normal plus energized/accident conditions for any of the solenoid valves. Reference [28, Addendum 1, Table I] provides maximum operating temperatures of 183°F and 226°F for the solenoid housing electrical components when energized at 120Vac and 72°F and 120°F ambient conditions, respectively.

Model V526-5295-67 Solenoid Valves

90°F 180°F

These valves (SV-3307, 3308, 4001A/B, 4002A/B, 4003A/B, 4004A/B, 4005A/B, 4020A/B, 4081, and 4082) are located in RB Volumes 11, 18, or 19 and experience a worst-case normal ambient temperature of ~~85°F~~ [1]. Interpolating the data from Valcor yields an operating temperature of 200°F ~~(with +5°F for conservatism)~~ for the solenoid housing electrical components at an ~~85°F~~ ambient temperature. All of the Model V526-5295-67 solenoid valves are de-energized post-LOCA (for their environmentally qualified required function: containment isolation and valve position indication) during which time the worst-case post-LOCA ambient is less than ~~160°F~~. As such, the worst-case bounding operating temperature of the solenoid housing electrical components in the Model V526-5295-67 valves is 200°F (366.48K) for normal (valves energized) and post-LOCA operation (valves de-energized, ambient ~~158.7°~~F). The thermal aging for the solenoid housing electrical components per Reference [2, Table 3.2.2-III] or [32, Table V] was 3,395 hours at 320°F (433.15K). The qualified life of the solenoid housing electrical components is thus:

$$t_a = 3{,}395 \text{ hours} \times \exp\left[\frac{1.136 \text{ eV}}{8.617\text{x}10^{-5} \text{ eV/K}}\left(\frac{1}{366.48\text{K}} - \frac{1}{433.15\text{K}}\right)\right] = 98 \text{ years}$$

Model V526-5891-54 Solenoid Valves

121.3°F 100°F

These valves (SV-4234 and 4235) are located in RB Volume 18 and TB Volume 25, respectively, where the normal ambient temperature is ~~80°~~F [1]. These valves are normally de-energized. Under post-LOCA operation, the valves are assumed to be continuously energized in a worst-case ambient temperature of ~~109.3°~~F [1]. ~~For conservatism, the ambient temperature during this time is taken as 120°F~~. From the Valcor data stated earlier, the operating temperature of the solenoid housing electrical components at 120°F ambient is 226°F ~~(380.93K)~~. This value is applicable to the post-LOCA operating period while ~~80°F (299.81K)~~ is considered for normal plant operation of the Model V526-5891-54 solenoid valves. The thermal aging for the solenoid housing electrical components per Reference [2, Table 3.2.2-III] or [32, Table V] was 3,395 hours at 320°F (433.15K). The LOCA/post-LOCA operating time and qualified life of the solenoid housing electrical components is determined as follows:

100°F (310.93K)

Demonstrated LOCA/post-LOCA time (valve energized):

$$t_a = 3{,}395 \text{ hours} \times \exp\left[\frac{1.136 \text{ eV}}{8.617\text{x}10^{-5} \text{ eV/K}}\left(\frac{1}{\cancel{380.93\text{K}}} - \frac{1}{433.15\text{K}}\right)\right] = \cancel{25} \text{ years}$$

23.5

For conservatism, the 1.3°F difference between the higher PLHU ambient in RB Volume 18 will be added resulting in a service temperature of 227.3°F (381.65K).

Assuming that post-LOCA operation lasts a year (only need 198 days to bound 180 days worst-case plus margin), this leaves 24 years of demonstrated service life of the solenoid housing electrical components while energized. The ~~24~~ years time remaining ~~at 226°F (380.93K)~~ can be converted to an equivalent time duration at the normal plant temperature of ~~80°F (299.81K)~~, when the valves are de-energized, as follows:

100°F (310.93K)

$$t_a = \cancel{24}\text{ years} \times \exp\left[\frac{1.136\text{ eV}}{8.617\text{x}10^{-5}\text{ eV/K}}\left(\frac{1}{\cancel{299.81\text{K}}} - \frac{1}{\cancel{380.93\text{K}}}\right)\right] >> 100\text{ years}$$

23 years, 227.3°F (381.65K)

Result still valid, although inputs changed due to EPU and newer ambient temperature.

Vespel Valve Seats (Model V526-5295-67)

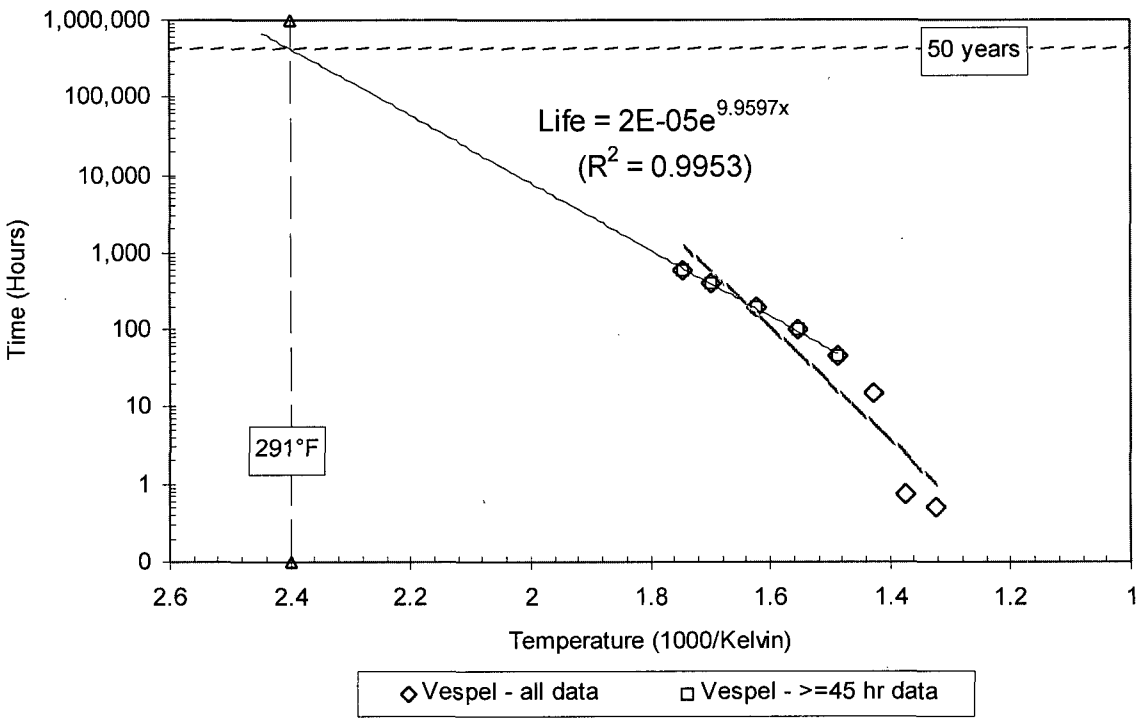
The Model V526-5295-67 solenoid valves (SV-3307, 3308, 4001A/B, 4002A/B, 4003A/B, 4004A/B, 4005A/B, 4020A/B, 4081, and 4082) are located in RB Volumes 11, 18, and 19. Most of these valves are installed on lines that are heat-traced [34, 35, 36]. Per Reference [37], the maximum temperature specified for the heat-trace temperature controllers is 286°F±4.5°F. Since the heat tracing is temperature controlled, the ambient temperature has no influence on the valve seat operating temperature. As such, a continuous temperature of 291°F will be considered for all periods of normal and post-LOCA operation.

Valcor did not thermally age Vespel valve seats. However, Reference [28, Appendix XIII of Appendix IX] provides DuPont's Arrhenius aging data for an end-point life condition of 50% original tensile strength for this material. The aging data considered is the following:

Time (hours)	Temperature (F)	Temperature (K)	1000/Temp (1000/K)
0.5	900	755.37	1.323855
0.75	850	727.59	1.374400
15	800	699.81	1.42895
45	750	672.04	1.488000
100	700	644.26	1.552168
200	650	616.48	1.62211
400	600	588.70	1.698658
600	572	573.15	1.744744

By plotting Time (ordinate) versus the Inverse Temperature (abscissa) using Microsoft Excel software, the Arrhenius plot shown in Figure 1 was created. There are two trend line curves shown in Figure 1, the shorter (faint) line is based on all of the data points. The longer trend line (darker) only uses the data points from 45 hours or greater thermal aging. This curve results in a more conservative representation or Arrhenius life characteristic line that can be extended down to the normal operating temperature of 291°F for the Vespel material.

Figure 1
Arrhenius Aging Data for Vespel (50% Original Tensile Strength)



Also shown in Figure 1 is the intercept of the extrapolated life curve to the seat operating temperature of 291°F. The life equation (determined by the Excel software) indicates the following life at 291°F (417.04K) (Note, exponent corrected by the factor of 1,000 applied to the 1/T scale):

$$Life = 2x10^{-5} \times \exp\left(\frac{9959.7}{417.04}\right) = 53.7 \text{ years}$$

Calculation still valid for EPU, heat tracing has more influence on service temperature than normal or PLHU ambient.

The "50 year" line shown in Figure 1 is provided as a reference. The Model V526–5295-67 solenoid valves (SV-3307, 3308, 4001A/B, 4002A/B, 4003A/B, 4004A/B, 4005A/B, 4020A/B, 4081, and 4082) were purchased under Cherne Purchase Order M5073. The earliest these valves were available for installation was 1981 [29]. The current plant license expires September 2010, with extended plant operation under a renewed license, this would be September 2030 (reference Monticello License Renewal Application). As such, the Model V526-5295-67 solenoid valves only need to have a qualified life of 50 years (2030 – 1980, for margin). Accordingly, the Vespel valve seats posses adequate life to survive 50 years of normal plant operation, plus an additional 3.7 years. This bounds operation under post-LOCA conditions (note, valves will still be heat-traced during post-LOCA operation).

EPR Valve Seats (Model V526-5891-54)

121.3°F

100°F

The Model V526-5891-54 solenoid valves (SV-4234 and 4235) are located in RB Volume 18 and TB Volume 25. These valves are installed on Alternate Nitrogen gas supply lines. The valves were originally installed as normally closed, energize to open Valcor Model V526-5295-159 under Modification 87Z002 [15]. Under Modification 93Q305, Part C, the valves were re-configured as normally open, energize to close Valcor Model V526-5891-54. During the retro-fit, the EPR valve seats from the original valve were re-used for the new model [39]. Regardless, the solenoid valves were not, and are not, energized during normal plant operation. Under a design basis LOCA, the valves may be energized, but the solenoid coil heat is not greatly influential on the valve seat due to the stand-off isolation of the coil assembly [38]. The valve are not installed on lines that are heat-traced. Under normal plant operation, the ambient temperature is ~~80°F~~ [1], during post-LOCA, the worst-case ambient temperature is ~~109.3°~~F [1]. As such, a continuous temperature of ~~110°F (316.48K)~~ will be considered for all periods of normal and post-LOCA operation for the EPR valve seats. The thermal aging for the EPR valve seats, per Reference [28, Addendum 1, Page 6] was 614 hours at 318°F (432.04K). The qualified life of the EPR valve seat is thus:

122°F (323.15K)

$$t_a = 614\text{ hours} \times \exp\left[\frac{1.038\text{ eV}}{8.617\text{x}10^{-5}\text{ eV/K}}\left(\frac{1}{\cancel{316.48\text{K}}} - \frac{1}{432.04\text{K}}\right)\right] >> 100\text{ years}$$

Result still valid although inputs changed under EPU

Aging Summary

Based on the above calculations, the Model V526-5295-67 and V526-5891-54 solenoid valves have a qualified lives greater than that required to satisfy extended plant operation under a renewed license, plus 180 day operation under a design basis LOCA.

Appendix 2 - Thermal Aging & Qualified Life

Required: 49 years

Reported: >49 years

90°F 124.2°F

Transmitter FT-2942 is located in Reactor Building Volume 37. The normal average temperature for this room is ~~80~~ °F. As shown in Appendix 1 (Figure 1.1), the Post LOCA heatup results in a maximum temperature of ~~112.8~~ °F. As this is strictly an increase in temperature and has no effect on the pressure or humidity conditions, this temperature excursion will be evaluated as part of the thermal aging analysis.

The test units were thermally aged for 51 days @ 105°C (221 °F) [6/p.30]. Based on an analysis of the limiting component, the transformer, it can be shown that these transducers are qualified in excess of the remaining design life of the plant, including the accident (198 days, includes margin).

In order to evaluate the accelerated aging on the Tavis Flow Transmitters, the Arrhenius method will be applied [25]. This is an accepted methodology for extrapolating accelerated aging data to the plant normal environmental conditions. The Arrhenius equation is as follows:

$$t_s = t_a \, Exp^{\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right]}$$

Where:

t_s	= Qualified Life @ Service Temperature
t_a	= Accelerated Test Time
E_a	= Activation Energy
K_b	= Boltzmann's Constant (8.617x10-5 eV/°K)
T_s	= Normal Service Temperature
T_a	= Accelerated Test Temperature

As stated previously, the accelerated test time (t_a) is 51 days. The accelerated test temperature (T_a) is 221 °F. The normal service temperature (T_s) is ~~80~~ °F. Reference [6] states that the applicable activation energy for the transformer is 0.89 eV [6/p. 8]. Therefore, applying the Arrhenius equation:

90°F (305.37K)

$$t_s = 51\,\text{days} \times \exp\left[\frac{0.89}{8.617\text{x}10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{378.15}\right)\right]$$

$$= \cancel{175}\ \text{years}$$ 93.7 years

This is the qualified life not including the accident conditions. However, as stated in Appendix 1, the accident Post LOCA Heatup will be taken into account in the thermal aging analysis. To do this, the equivalent thermal degradation must be calculated for the accident. The peak temperature during the Post LOCA is ~~112.8 °F [23]. To add additional conservatism, it will be assumed that the transmitters are exposed to 120 °F (322.04K) conditions~~ for the duration of the accident (198 days). Applying Arrhenius to the accident duration.

124.2°F (324.37K)

$$t_s = 198\,\text{days} \times \exp\left[\frac{0.89}{8.617\text{x}10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{\cancel{322.04}}\right)\right]$$

$$= \cancel{5.8}\ \text{years}$$ 3.9 years

Therefore, the total qualified life for the Tavis flow transmitters is:

~~175 years - 6 years (conservative) = **169 years**~~ 93.7 - 3.9 = 89.8 years

This is much greater than the 49 years required for these devices. The Tavis flow transmitters are considered qualified for remainder of MNGP's design life based on thermal aging.

Note:
Transformer activation energy = 0.89 eV, which was identified to be the most age-sensitive non-metallic material (Reference 6). The function of the potting compound is to harden the assembly to increase shock resistance. The potting compound is extremely stable at elevated temperatures. Therefore, the potting compound is not age sensitive (References 6 & 22) and is not included in the aging calculation.

(a sub-component of the transmitter)

APPENDIX 2
THERMAL AGING ANALYSIS

Thermal Aging Analysis

The Conoflow I/P Transducer test specimen was aged for 190 hours at 280°F [3, p.16] prior to the LOCA test. For the most limiting condition the normal service temperature of the SBGT Room, ~~80°F~~, will be used. Heat rise is not required to be considered since these devices use very low voltage/current circuits.

90°F (305.37K), RB Vol 37, new ambient temperature

Qualified life can be calculated using the Arrhenius equation [1] as follows:

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

Where:

t_a = Qualified Life
t_t = Aging Time (190 hrs)
E_a = activation energy - (0.8 eV [11])
K = Boltzmann's Constant (8.617×10^{-5})
T_a = Service temperature - ~~80°F (299.81K)~~
T_t = Aging temperature - 280°F (410.93K)

Qualified Life = $190 \times \exp\left[\frac{0.80}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{410.93}\right)\right]$

Qualified Life @ ~~80°F = 8.23 x 10^5 hours = 93.9 years~~ 53.4 years at 90°F

As discussed in Appendix 1 the accident testing was not extended long enough to envelop the entire 198 day duration of the post-accident operation. The excess portion of the thermal aging test, after demonstrating qualified life, will be analyzed for demonstrating this post accident operating time. This is acceptable since the required environment is harsh due to thermal effects alone and since the thermal aging test temperature envelopes the accident temperature. To determine the amount of thermal aging is required for post accident operation the 198-day profile will be extrapolated to ~~80°F~~ and then it will be subtracted from the demonstrated qualified life.

To simplify the calculation the Post-LOCA Heatup conditions in the SBGT Rooms will be conservatively assumed as ~~112.8°F (318.04K)~~ 124.2°F (324.37K) for 198 days. The equivalent accident aging for the transducer is given as:

$$t_a = 4{,}752 \text{ hours} \times \exp\left[\frac{0.80}{8.617\text{x}10^{-5}}\left(\frac{1}{299.81} - \frac{1}{318.04}\right)\right]$$

305.37K (90°F)

ta = ~~28,037 hours =~~ 3.20 years

Therefore the qualified life is equal to:

Qualified Life = ~~93.9 - 3.20 = 90.7 years~~ 53.4 - 3.2 = 50.2 years

Therefore, the qualified life of the Conoflow I/P Transducers is greater than 48 years (required to bound remaining plant life, including an additional +20 years under a renewed license) plus a 198 day post accident operating time.

Appendix 2 - Thermal Aging/Qualified Lives

Required: 47 Years at ~~80°F~~
Reported: >50 Years at ~~80°F~~

new RB Vol 37 ambient of 90°F (305.37K)

Contactor/Breaker

The critical degradable materials of construction are identified by the manufacturer in [4:P8].

The "weak link" material of the contactor and circuit breaker was identified as Phenolic, and was thermally aged for 107 hours at 100°C[4:P10]. The Durez 791 Phenolic used in these devices has an activation energy of 1.357eV [4], based upon retention of 50% of the original impact strength. This criterion is realistic and acceptable. The Arrhenius plot for this material is presented in [4:P9].

Using Arrhenius methodology outlined in [9], at normal operating temperature of ~~80°F~~ for the installed location, the qualified life of the Phenolic is greater than 47 years:

t_s	=	Qualified life @ Service Temperature
E_a	=	Activation Energy (1.357 eV)
t_a	=	Test Time (107 Hours per reference 4)
T_s	=	Normal Service Temperature (~~80°F or 299.82K~~)
T_a	=	Test Temperature (100°C or 373.15K per reference 4)
e_b	=	Boltzmann's Constant (8.617×10^{-5} eV/K)

142 years

$$t_s = t_a Exp^{\frac{E_a}{K_b}\left(\frac{1}{T_s}-\frac{1}{T_a}\right)} = (107)Exp^{\frac{1.357}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.82}}-\frac{1}{373.15}\right)} = \cancel{3253865 Hours = 374.52 Years}$$

Considering post LOCA heat up of the Reactor Building and effects it has on the contactor/breaker, the ~~112.8°F~~ maximum temperature [10] for 198 days will be equated to ~~80°F~~ to subtract from service life. The resulting number will be the service life available prior to accident aging.

124.2°F (324.37K)

t_s	=	Qualified life @ Service Temperature
E_a	=	Activation Energy (1.357 eV)
t_a	=	Accident Duration (198 Days including margin)
T_s	=	Normal Service Temperature (~~80°F or 299.82K~~)
T_a	=	Accident Temperature (~~112.8°F or 318.04K [10]~~)
e_b	=	Boltzmann's Constant (8.617×10^{-5} eV/K)

11.1 years

$$t_s = t_a Exp^{\frac{E_a}{K_b}\left(\frac{1}{T_s}-\frac{1}{T_a}\right)} = (198)Exp^{\frac{1.357}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.82}}-\frac{1}{\cancel{318.04}}\right)} = \cancel{4013 Days = 10.99 Years}$$

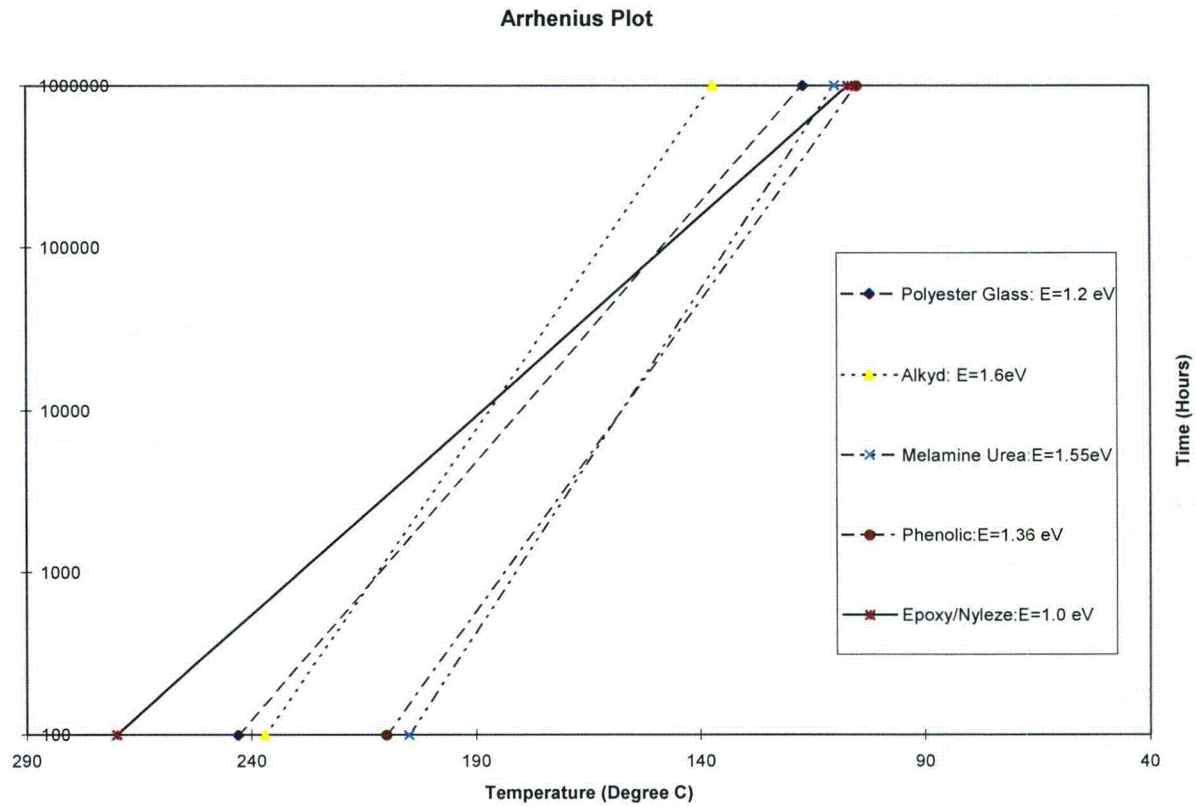
305.37

Subtracting the accident aging from the artificial aging yields ~~363.5 Years (374.52 Years - 10.99 Years)~~. Therefore, the critical weak link material is qualified in excess of the remaining 47 years of plant life (includes extended plant operation under a renewed license).

142 - 11.1 = 130.9 years

Epoxy Coils

The epoxy encapsulated coils may experience temperatures of 85°C (185°F) in the ambient plant environment due to self heating effects [4:p13]. Thermal aging qualification for the coil is achieved through material analysis and Arrhenius methodology of test data. The analysis was performed since the time and temperature required to age the coils could not be met without potentially damaging the coils unrealistically at excessive temperatures. Extrapolating the Arrhenius plot for the epoxy coils, Nyleze wire insulation service life for the coils at 85°C is theoretically about 240 years.



A substantial safety margin exists between the amount of degradation expected to occur in 47 years and the time to theoretical end of life at 85°C. Additionally, the coils are not likely to be energized continuously for the duration of the plant so the average operating temperatures would realistically be reduced, increasing the theoretical margin of safety.

Based on analysis, the qualified life of the coils is justified to be in excess of the 47 year remaining design life of the plant (includes extended operation under a renewed license).

Disconnect Switches

The disconnect switches contain three materials of construction, (identified by the manufacturer) that are subject to thermal degradation; melamine, glass filled polyester, and alkalyd/urea. See [4:p63-64] for further discussions on thermal aging of these components.

These three materials have been thermally aged in other devices for 670 hours at 110°C with no detrimental effects [4:p64]. Using the limiting activation energy for these materials (1.238 eV-Glass filled polyester [4:p66]), the qualified life of the disconnect is greatly in excess of the 47 year remaining plant life (includes extended plant operation under a renewed license).

t_s = Qualified life @ Service Temperature
E_a = Activation Energy (1.238 eV)
t_a = Test Time (670 Hours per reference 4)
T_s = Normal Service Temperature ~~(80°F or 299.82K)~~ 90°F (305.37K)
T_a = Test Temperature (110°C or 383.15K per reference 4)
e_b = Boltzmann's Constant (8.617×10^{-5} eV/K)

1,074 years

$$t_s = t_a Exp^{\frac{E_a}{K_b}\left(\frac{1}{T_s}-\frac{1}{T_a}\right)} = (670 Hours) Exp^{\frac{1.238}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.82}}-\frac{1}{383.15}\right)} = \cancel{2.25x10^7 Hours = 2568 Years}$$

Determining the effect of the post LOCA heatup on the equipment, the maximum temperature of ~~112.8°F~~ will be considered for the 198 day accident duration.

t_s = Qualified life @ Service Temperature
E_a = Activation Energy (1.238 eV)
t_a = Accident Duration (198 Days including margin)
T_s = Normal Service Temperature ~~(80°F or 299.82K)~~
T_a = Accident Temperature ~~(112.8°F or 318.04K [10])~~ 124.2°F (324.37K)
e_b = Boltzsman Constant (8.617×10^{-5} eV/K)

8.5 years

$$t_s = t_a Exp^{\frac{E_a}{K_b}\left(\frac{1}{T_s}-\frac{1}{T_a}\right)} = (198 Days) Exp^{\frac{1.238}{8.617x10^{-5}}\left(\frac{1}{\cancel{299.82}}-\frac{1}{\cancel{318.04}}\right)} = \cancel{3082 Days = 8.4 Years}$$

Therefore, subtracting the accident aging equivalent from the artificial aging equivalent yields a service life of ~~2559.6 years (2568 Years – 8.4 Years)~~. Thus, the disconnect is qualified in excess of the required 47 years from time of installation until the end of extended plant operation under a renewed license.

greater than 100 years

APPENDIX 2
THERMAL AGING ANALYSIS

The relays, 94–5A, B & C, are normally de-energized and are located in a service temperature environment of ~~80°F~~. The test program [3] did not thermally age the subject relays. However, the thermal aging analysis performed in Appendix VI of Reference [3] established the necessary thermal aging data for each sub-component of the relays to evaluate a qualified life.

Table 2-1 provides a list of the non-metallic materials along with the applicable failure mechanism. The data in Table 2-1 is taken from the Appendix VI of Reference [3]. The equivalent life of each sub-component of the relay at ~~80°F~~ is calculated using the following form of the Arrhenius equation [3, App VI] to determine life at the service temperature of ~~80°F~~:

$$\text{Service Life} = e\left[\frac{E_a}{K}\cdot\left(\frac{1}{T}\right)+C\right]$$

where,

Service Life = Sub-component life (hrs) at specific service temperature
Ea = Activation energy (eV)
T = Service temperature ~~(80°F/299.81K)~~ 85°F (302.59K)
C = Intercept, [3, App VI]
K = Boltzmann's Constant 8.617×10^{-5} eV/K

The results of this analysis is given below in Table 2-1 with all the components having much greater than a 60-year life in a de-energized state.

Table 2-1
List of Non-Metallic Sub-components in the Eaton Cutler–Hammer D26MR04 Relays

Material	Description	Activation Energy	Intercept	Failure Mechanism	Service Life at ~~80°F~~ (yrs)
Adhesive	---	0.8044	–16.092	50% reduction of bond strength	390
Adhesive	---	0.8044	–16.092	50% reduction of bond strength	390
Sealant	---	0.8044	–16.092	50% reduction of bond strength	390
Adhesive/sealant	---	0.8044	–16.092	50% reduction of bond strength	390
Nylon 6:6, Zytel 101	Front attachment component	0.8415	–17.373	50% reduction of tensile strength	455
Nylon 6:6, Zytel 101	Pneumatic Timer component	0.8415	–17.373	50% reduction of tensile strength	455
Polycarbonate Resin Lexan 121R–112	Unit pole housing cover	0.6734	–10.635	50% reduction of tensile impact strength	574
Loctite 36590	Magnet frame adhesive	0.8577	–17.312	50% reduction of bond strength	905
Plenco 03509 - Cotton & Mineral filled Phenolic	Pneumatic timer component	0.6374	–8.9681	50% reduction of flexural strength	754

see last sheet for revised thermal lives at 85°F

Table 2-1
List of Non-Metallic Sub-components in the Eaton Cutler–Hammer D26MR04 Relays

Material	Description	Activation Energy	Intercept	Failure Mechanism	Service Life at 80°F (yrs)
Polyacrylic	Magnet cushion	0.7706	–13.326	50% reduction of dielectric withstand strength	>1,000
Plenco 07500 Durez 123	Cotton filled Phenolic Housing	0.7168	–11.316	50% reduction of flexural strength	>1,000
Acetal copolymer Celcon M90–04	Pneumatic timer component	0.9501	–19.656	50% reduction of tensile strength	>1,000
Acetal Delrin 500	Pneumatic timer component	0.9501	–19.656	50% reduction of tensile strength	>1,000
PTFE Filled Acetal LNP Fulton 404	Pneumatic timer component	0.9501	–19.656	50% reduction of tensile strength	>1,000
PVC	Tubing	1.155	–26.434	50% loss of elongation	>1,000
Molybdenum Disulfide Filled Nylon Nylatron GS JF3–10166	Pneumatic timer component	0.8529	–15.070	50% reduction of tensile strength	>1,000
Teflon Filled Polysulfone LNP GL4040	Unit pole spring retainer	0.7211	–9.6924	50% reduction of tensile strength	>1,000
33% Short Fiber Glass Filled Nylon 6:6 Zytel 70G–33I–BK031	Push bar	0.8590	–14.713	50% reduction of tensile strength	>1,000
Glass Fortified Nylon 6:6 Zytel 70G–33L	Pneumatic timer component	0.8590	–14.713	50% reduction of tensile strength	>1,000
Glass Fortified Nylon Well–Sphere GS25–66	Connecting Rod	0.8590	–14.713	50% reduction of tensile strength	>1,000
Silicone rubber		0.7755	–11.550	50% loss of elongation	>1,000
Polycarbonate Lexan 141–701	Transient suppressor housing	1.154	–24.017	50% reduction of tensile impact strength	>1,000
Polyethylene	Pneumatic Timer component	1.176	–24.811	50% reduction of dielectric withstand strength	>1,000
Epoxy Hardener		1.034	–19.129	50% reduction of dielectric withstand strength	>1,000
Epoxy Resin		1.034	–19.129	50% reduction of dielectric withstand strength	>1,000
Mineral Filled Epoxy	Magnet bobbin and encapsulant	0.9605	–16.128	50% reduction of electrical strength	>1,000 (>100,000)
Polyterephthalate Valox 325–1001 Orgater TMN	Unit pole housing	1.255	–26.718	50% reduction of electrical strength	>1,000
40% Glass Filled PPS Resin, Ryton R–4	Push bar	0.7650	–7.1683	50% reduction of impact strength	>1,000
Heat Stabilized Nylon Zytel 103HS–L Celanese Nylon 1003–1	Insulator between contact carrier in rear and front deck	1.250	–25.372	50% loss of elongation	>1,000
Liquid rubber	Casting compound	1.403	–29.200	10% loss of weight	>1,000

Table 2-1
List of Non-Metallic Sub-components in the Eaton Cutler–Hammer D26MR04 Relays

Material	Description	Activation Energy	Intercept	Failure Mechanism	~~Service Life at 80°F (yrs)~~
General Latex & Chemical IP712	Casting compound	1.403	–29.200	10% loss of weight	~~>1,000~~
Kraft–Mylar–Kraft	Crossover insulation	1.455	–28.544	50% reduction of dielectric withstand strength	~~>1,000 (>1.0x10⁶)~~
Polyester–Polyamide–Imide Overcoat	Magnet wire insulation	1.600	–27.686	50% reduction of dielectric withstand strength	~~>1,000 (>1.0x10⁶)~~
Grease			See Note 1		
Powdered cement			See Note 2		
Permacel 248 Tape	Tape is used prior to encapsulation. No function after magnet assembly. No safety function				
Paper Tape	Tape is used prior to encapsulation. No function after magnet assembly. No safety function				
Double Adhesive Mylar Tape	Tape is used prior to encapsulation. No function after magnet assembly. No safety function				

Notes:

1. The Mobilgrease 28 activation energy varies significantly with temperature. The life of the Mobilgrease 28 will be assumed to be equal to the most limiting material.

2. The Sauereisen powdered cement forms a ceramic material upon drying. Therefore, aging is not applicable in the range of operating temperatures for the EATON Cutler–Hammer relays.

Further thermal analysis is required to determine the capability of energized relay in an accident condition. Where:

Accident Aging = Service Life – Qualified Life

121.2°F (322.70K)

Sixty years of qualified life will be removed from the service life in Table 2-1. The remaining life will be extrapolated to the accident temperature of ~~115.6°F~~ plus the associated worst-case heat rise of each sub-component. The heat rise for each sub-component is given in Appendix VI of Reference [3] and shown on Table 2-2 below.

The equivalent accident aging for each sub-component is calculated using the Arrhenius equation as follows:

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

where,

t_a = Life available for accident conditions
t_t = Service Life at ~~80°F~~ from Table 2-1 minus 60 years
Ea = Activation energy (eV)
T_a = Service temperature - ~~80°F (299.81K)~~
T_t = Accident conditions - ~~115.6°F (319.59K)~~ + Worst-Case Heat Rise
K = Boltzmann's Constant - 8.617×10^{-5} eV/°K

85°F (302.59K)

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Not specifically shown in the EPU re-evaluation on the last sheet. This column is simply thermal life remaining after subtracting 60 years from Table 2-1 results.

Table 2-2
Analysis for Life at Accident Conditions for Eaton Cutler–Hammer D26MR04 Relays

Material	Description	Activation Energy	t_L (years)	Worst-Case Heat Rise	Life (yrs) Available for Accident Condition
Adhesive	---	0.8044	330	15°C	12.9
Adhesive	---	0.8044	330	15°C	12.9
Sealant	---	0.8044	330	15°C	12.9
Adhesive/sealant	---	0.8044	330	15°C	12.9
Nylon 6:6, Zytel 101	Front attachment component	0.8415	395	15°C	13.4
Nylon 6:6, Zytel 101	Pneumatic Timer component	0.8415	395	15°C	13.4
Polycarbonate Resin Lexan 121R–112	Unit pole housing cover	0.6734	514	36°C	8.6
Loctite 36590	Magnet frame adhesive	0.8577	992	36°C	5.4
Plenco 03509 - Cotton & Mineral filled Phenolic	Pneumatic timer component	0.6374	807	15°C	62
Polyacrylic	Magnet cushion	0.7706	>940	36°C	>8.7
Plenco 07500 Durez 123	Cotton filled Phenolic Housing	0.7168	>940	36°C	>12.1
Acetal copolymer Celcon M90–04	Pneumatic timer component	0.9501	>940	15°C	>20.6
Acetal Delrin 500	Pneumatic timer component	0.9501	>940	15°C	>20.6
PTFE Filled Acetal LNP Fulton 404	Pneumatic timer component	0.9501	>940	15°C	>20.6
PVC	Tubing	1.155	>940	28°C	>2.0
Molybdenum Disulfide Filled Nylon Nylatron GS JF3–10166	Pneumatic timer component	0.8529	>940	15°C	>30.4
Teflon Filled Polysulfone LNP GL4040	Unit pole spring retainer	0.7211	>940	36°C	>11.8
33% Short Fiber Glass Filled Nylon 6:6 Zytel 70G–33I–BK031	Push bar	0.8590	>940	36°C	>5.1
Glass Fortified Nylon 6:6 Zytel 70G–33L	Pneumatic timer component	0.8590	>940	15°C	>29.6
Glass Fortified Nylon Well–Sphere GS25–66	Connecting Rod	0.8590	>940	36°C	>5.1
Silicone rubber	---	0.7755	>940	15°C	>41.5
Polycarbonate Lexan 141–701	Transient suppressor housing	1.154	>940	28°C	>2.0
Polyethylene	Pneumatic Timer component	1.176	>940	15°C	>8.3
Epoxy Hardener	---	1.034	>940	28°C	>3.8
Epoxy Resin	---	1.034	>940	28°C	>3.8
Mineral Filled Epoxy	Magnet bobbin and encapsulant	0.9605	>100,000	82.5°C	>7.8

see last sheet for re-evaluation under EPU.

Table 2-2
Analysis for Life at Accident Conditions for Eaton Cutler–Hammer D26MR04 Relays

Material	Description	Activation Energy	t_L (years)	Worst-Case Heat Rise	Life (yrs) Available for Accident Condition
Polyterephthalate Valox 325–1001 Orgater TMN	Unit pole housing	1.255	>940	36°C	>160 (days)
40% Glass Filled PPS Resin, Ryton R–4	Push bar	0.7650	>940	36°C	>9.0
Heat Stabilized Nylon Zytel 103HS–L Celanese Nylon 1003–1	Insulator between contact carrier in rear and front deck	1.250	>940	36°C	>170 (days)
Liquid rubber	Casting compound	1.403	>940	28°C	>190 (days)
General Latex & Chemical IP712	Casting compound	1.403	>940	28°C	>190 (days)
Kraft–Mylar–Kraft	Crossover insulation	1.455	$>1.0 \times 10^6$	82.5°C	>219 (days)
Polyester–Polyamide–Imide Overcoat	Magnet wire insulation	1.600	$>1.0 \times 10^6$	82.5°C	>52 (days)

The post accident operating time of 30 days is enveloped by the remainder of the thermal life capability (after subtracting 60-years) for each of the sub-components analyzed in Table 2-2. Therefore the Qualified Life of the Eaton Cutler-Hammer D26MR04 relays is greater than 60 years plus a 33 day post accident operating time.

	AE (eV)	Intercept	Life @ 85°F	Temp Rise (°C)	Accident Life @ 121.2°F
Adhesive/Sealant	0.8044	-16.092	293	15	9.42
Nylon 6:6, Zytel 101	0.8415	-17.373	337	15	9.68
Polycarbonate ResinLexan 121R–112	0.6734	-10.635	451	36	6.89
Loctite 36590	0.8577	-17.312	668	36	3.54
Plenco 03509 - Cotton & Mineral filled Phenolic	0.6374	-8.9681	601	15	42.60
Polyacrylic	0.7706	-13.326	1,273	36	11.92
Plenco 07500Durez 123	0.7168	-11.316	1,207	36	15.56
Acetal copolymerCelcon M90–04	0.9501	-19.656	2,216	15	48.79
AcetalDelrin 500	0.9501	-19.656	2,216	15	48.79
PTFE Filled AcetalLNP Fulton 404	0.9501	-19.656	2,216	15	48.79
PVC	1.155	-26.434	6,527	28	14.85
Molybdenum Disulfide Filled NylonNylatron GS JF3–10166	0.8529	-15.07	5,227	15	172.29
Teflon Filled PolysulfoneLNP GL4040	0.7211	-9.6924	7,219	36	94.64
33% Short Fiber Glass Filled Nylon 6:6Zytel 70G–33I–BK031	0.859	-14.713	9,439	36	54.21
Glass Fortified Nylon 6:6Zytel 70G–33L	0.859	-14.713	9,439	15	305.21
Glass Fortified NylonWell–Sphere GS25–66	0.859	-14.713	9,439	36	54.21
Silicone rubber	0.7755	-11.55	9,074	15	409.24
PolycarbonateLexan 141–701	1.154	-24.017	70,430	28	162.39
Polyethylene	1.176	-24.811	74,022	15	679.98
Epoxy Hardener	1.034	-19.129	93,729	28	406.41
Epoxy Resin	1.034	-19.129	93,729	28	406.41
Mineral Filled Epoxy	0.9605	-16.128	112,454	82.5	9.99
PolyterephthalateValox 325–1001Orgater TMN	1.255	-26.718	227,503	36	122.19
Heat Stabilized NylonZytel 103HS–LCelanese Nylon 1003–1	1.25	-25.372	721,551	36	399.40
40% Glass Filled PPS Resin, Ryton R–4	0.765	-7.1683	485,152	36	4927.76
Liquid rubber	1.403	-29.2	5,548,224	28	3454.38
General Latex & Chemical IP712	1.403	-29.2	5,548,224	28	3454.38
Kraft–Mylar–Kraft	1.455	-28.544	78,552,974	82.5	57.30
Polyester–Polyamide–Imide Overcoat	1.6	-27.686	48,184,336,592	82.5	8596.76

Replacement column for Table 2-1 (values in years)

Replacement column for Table 2-2 (values in years).

NOTE: this EPU analysis used the full thermal life instead of shortening the lives to 1,000, 100,000, or 1,000,000 years as done in the CLTP analysis. This results in all accident lives being now expressed in years rather than shortening some to "days" as done in the CLTP analysis.

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Appendix 2 - Thermal Aging/Qualified Lives

Required: 60 Years
Reported: > 60 Years

85°F (302.59K)

The test sample was removed from the plant and had already been subjected to 20 years of natural aging [2] in the plant environment. Accelerated aging was performed to extend the service life further. The accelerated aging test was performed at 234°F for 25 hours [2]. Using the Arrhenius equation, an extrapolation is made to determine an equivalent life at the normal service temperature of 80°F (no temperature rise is applicable for the pressure switch instrument circuits). An activation energy of 1.14 eV will be used [2].

t_s = Qualified life @ Service Temperature
E_a = Activation Energy (1.14 eV)
t_a = Test Time (25 Hours per Reference [2])
T_s = Normal Service Temperature ~~(80°F or 299.81K)~~
T_a = Test Temperature (234°F or 385.37K per Reference [2])
e_b = Boltzmann's Constant (8.617×10^{-5} eV/K)

34.2 years

$$t_s = t_a \times \exp\left[\frac{E_a}{e_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 25 \text{ hours} \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{299.81} - \frac{1}{385.37}\right)\right] = \cancel{51 \text{ years}}$$

Combining the 20-years of actual aging inside the plant before testing and the accelerated aging of the cable simulated in [2], the resulting service life would be ~~71 years.~~ 54.2 years

~~To account for 180 days of post LOCA operation at 115.6°F, an equivalence of these conditions will be correlated to the 80°F service temperature and subtracted from the 71 year service calculated above.~~

t_s = Qualified life @ Service Temperature
E_a = Activation Energy (1.14 eV)
t_a = Accident Time (180 days)
T_s = Normal Service Temperature (80°F or 299.81K)
T_a = Accident Temperature (115.6°F or 319.59K per Reference [1])
e_b = Boltzmann's Constant (8.617×10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{e_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 180 \text{ days} \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{299.81} - \frac{1}{319.59}\right)\right] = 7.6 \text{ years}$$

~~Therefore, a qualified life of 63.4 years is determined (71 years service life – 7.6 year accident equivalence).~~

(attached)

The limited thermal aging of Reference [2] testing combined with the natural aging of 20 years is not sufficient to demonstrate a qualified life of 60 years. Information provided in Reference [4] from ITT Royal indicates that their PVC insulation has undergone physical property testing which included 168 hours of thermal aging at 136°C. Using this thermal aging and the revised ambient temperature of 85°F, a thermal life can be computed as follows:

t_s	=	Thermal life @ Service Temperature
E_a	=	Activation Energy (1.14 eV)
t_a	=	Test Time (168 hours per Reference [4])
T_s	=	Normal Service Temperature (85°F or 302.59K)
T_a	=	Test Temperature (136°C or 409.15K per Reference [4])
e_b	=	Boltzmann's Constant (8.617×10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{e_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 168 \text{ hours} \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{302.59} - \frac{1}{409.15}\right)\right] => > 100 \text{ years}$$

To account for 180-days of post-LOCA operation at 121.3°F, an equivalence of these conditions will be correlated to the 85°F service temperature and subtracted from the thermal life calculated above.

t_s	=	Equivalent Duration
E_a	=	Activation Energy (1.14 eV)
t_a	=	Accident Time (180 days)
T_s	=	Normal Service Temperature (85°F or 3022.59K)
T_a	=	Accident Temperature (121.3°F or 322.76K)
e_b	=	Boltzmann's Constant (8.617×10^{-5} eV/K)

$$t_s = t_a \times \exp\left[\frac{E_a}{e_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right] = 180 \text{ days} \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{302.59} - \frac{1}{322.76}\right)\right] = 7.6 \text{ years}$$

Therefore, a qualified life of greater than 60 years is determined for the ITT cables.

SENT BY:GATEWAY WIRE AND CABLE;10-27-89 10:55AM ; -15635568→ ;# 9,415 845 8453;# 3

ITT ROYAL ELECTRIC

ITT CORP. • 95 GRAND AVENUE • PAWTUCKET, RHODE ISLAND 02862

PRODUCT DATA
WC-6 REV. E
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B1022056

191905-2-3-054
Rev 0
Reference # 4
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THHN UNIVERSAL BUILDING WIRE
600 VOLT

Stranded—ITT # 12 AWG Type MTW or THHN or THWN to Gasoline and Oil Resistant II (UL) or AWM Styles 1318 & 1430 600 Volts VW-1

UNDERWRITERS' LABORATORIES LISTED THHN/THWN, TFFN, MTW, AWM

FEATURES

- Smaller overall diameter
- Rated 90°C Dry 75°C Wet
- Gasoline and Oil (75°C) Resistant
- Abrasion and chemical resistant
- Protected with tough slick nylon jacket over PVC insulation
- Flame Retardant, UL VW-1

USES

- 90°C Building wire Dry locations
- 105°C Appliance wire
- 90°C Machine tool wire — stranded only
- 90°C Fluorescent ballast hook-up wire

DESCRIPTION

ITT THHN/THWN offers outstanding cost and space-saving advantages to the contractor, the electrician, the manufacturer and the design engineer. It is the smallest diameter of any building wire for general purpose application, either new work or rewiring. Using THHN/THWN can result in substantial space savings, increased circuit capacity, and lower installation cost because it allows smaller sizes of conduit, fewer straps and connectors. ITT THHN/THWN meet the UL "VW-1" Vertical Flame Test and is surface marked "VW-1" for flame retardant recognition. It is ideal for appliance, lighting and machine tool applications. With reduced wire size, lightness, and slick nylon jacketing, THHN/THWN can be pulled through conduits and raceways with great ease. It has excellent abrasion, chemical, gasoline and oil resistance. ITT THHN/THWN also conforms to the NEC regulation requiring wires within three inches of a fluorescent ballast to be 90°C type wire. 18 and 16 AWG are UL listed as TFFN.

MATERIAL SPECIFICATION DATA

Conductor: Soft drawn annealed copper conductors. Available in solid or stranded in sizes 14, 12 and 10 AWG. Sizes 18, 16 and 8 AWG and larger available in stranded only.

Insulation: Extruded 105°C polyvinyl chloride insulation with slick polyamide nylon overall jacket that meets the UL "VW-1" Vertical Flame Test.

Colors: Smaller gauge sizes available up to 13 colors. See page 4 for specific information.

CHEMICAL RESISTANCE

THHN has excellent resistance to most chemicals, solvents or fumes. Underwriters' Laboratories, Inc. recognizes the use of nylon jacketed PVC insulated wires, similar to THHN, in areas where gasoline is prevalent, such as auto service stations. Since the jacket contains no plasticizers, there are no potential exuding problems, even at high temperatures.

WATER ABSORPTION

When tested in accordance with UL standard 1581, the water absorption results were no more than 20 milligrams per square inch of exposed surface.
TEST: With nylon jacket removed, the vinyl insulation is conditioned for seven days at 82°C after which the amount of water absorption is determined. ITT THHN/THWN wire has passed this test in all cases.

REVISED INFORMATION

EFFECTIVE JANUARY 28, 1985

PHYSICAL PROPERTIES

PHYSICAL REQUIREMENTS OF WIRE

Unaged Sample:

Min. Tensile Strength, P.S.I.	2000
Min. Elongation at Rupture	150% (2 to 5)

Accelerated Aging Test: In a full draft circulating air oven for 168 hours at 136°C (276.8°F.) with nylon jacket removed before aging

Tensile 75% of Unaged Sample — Die cut or other specimen
Elongation 45% of Unaged Sample — Die cut specimens
Elongation 65% of Unaged Sample — Other specimens

GASOLINE RESISTANCE

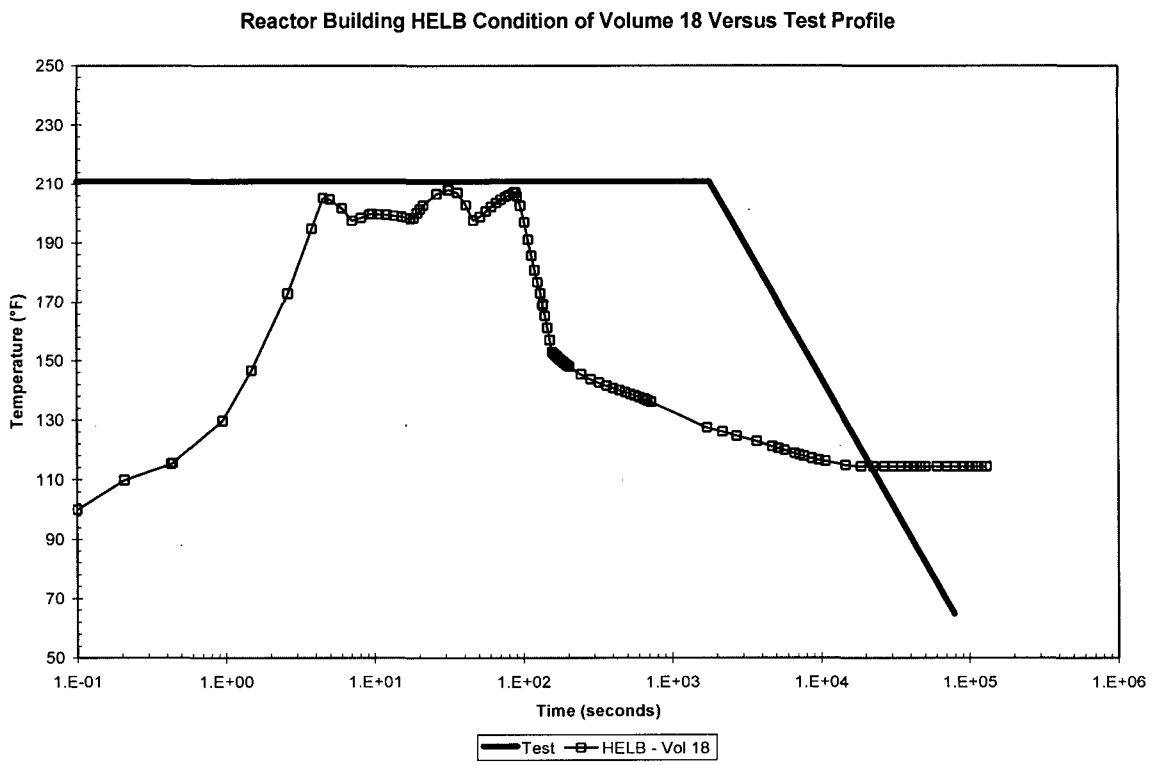
ITT THHN/THWN and TFFN wire marked gasoline resistant has been subjected to 30 days in gasoline at 30°C test. There was no appreciable increase in thickness of the nylon or insulation, nor was there any adverse effects upon either as result of the immersion in gasoline. The physical properties of the gasoline-aged wire are required to be 75% percent that of wire that is shelf-aged.

MICROFILMED

Appendix 1 - Accident Evaluation

Temperature

The Okonite PE/PVC control cables are installed in Reactor Building Volumes 5, 14, 18, 19, and 22. For Volume 5 (RCIC pump room), the Okonite cables are used on the temperature switches (TS-13-79C/D through TS-13-82C/D [15]) that sense and provide trip signals for RCIC steam line break detection. The set-points or these switches must be ≤200°F per Table 3.2.1 of the Technical Specifications [16]. As such, the bounding HELB temperature conditions for the Okonite cables occur in Volume 18. The HELB profile for Volume 18 and the test temperature profile from the Reference [2] test are shown below:



As shown, the test bounds the worst-case HELB for the areas in which the Okonite cable is installed. During post-LOCA operation, the ambient temperature of Reactor Building Volumes 5, 14, 18, 19, or 22 remains below ~~110°F~~ [14]. Although this is a non-harsh condition, an exposure of 180-days at ~~110°F~~ will be analyzed as part of the Thermal Aging/Qualified Life in Appendix 2.

121.3°F

Appendix 2 - Thermal Aging/Qualified Life

Required: 60 Years

Reported: >60 Years

95°F

85°F

The thermal aging & qualified life for the Okonite cable can be determined by using the Arrhenius equation to demonstrate the total qualified life at the normal service temperature. The normal service temperature where these cables are installed is ~~80~~°F (except for Volume 5 where the normal ambient is ~~90~~°F [14]). The Okonite cable has Polyethylene insulation and a PVC jacket (Reference 2). The activation energy was researched to determine a reasonable yet conservative value to use for this analysis. EPRI research (Reference 8) reports values for Polyethylene insulation between 1.13 and 3.10 eV. Other manufactures of Polyethylene insulated cable (GE and Rockbestos) also report activation energies within this range. Therefore, a conservative value of 1.13 eV was used for this analysis of the polyethylene insulation. The PVC jacket is not credited as an insulator for the cable and not evaluated.

A piece of the cable was removed from the Monticello plant and used as the test sample (Reference 2). The test sample was already naturally aged for 20 years at 90°F (normal operating conditions). The specimens were then subjected to 25 hours at 234°F. ~~Therefore, the~~ Arrhenius method will be used to determine the qualified life ~~plus the 20 years of natural aging~~.

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$$t_s = t_a \, Exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right]$$

95°F (308.15K)

Where:

t_s	Qualified Life @ Service Temperature (~~80°F~~)
t_a	Accelerated Test Time (~~25 hours~~) 340 hours
E_a	Activation Energy (1.13 eV)
K_b	Boltzmann's Constant (8.617×10^{-5} eV/K)
T_s	Normal Service Temperature (~~80°F or 299.81K~~)
T_a	Accelerated Test Temperature (~~234°F or 385.37K~~)

212°F (373.15K)

Qualified Life at ~~80°F (Reactor Building Volumes 14, 18, 19, and 22)~~

$$t_s = \cancel{25}\,\text{hours} \times \exp\left[\frac{1.13}{8.617\text{x}10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{\cancel{385.37}}\right)\right] = \cancel{47\text{ years}}$$

64.3 years

Appendix 2 – Thermal Aging/Qualified Life (Continued)

121.3°F (322.76K)

~~With the 20 years of natural aging, this becomes 67 years of expected service life for those Okonite PE/PVC cables in Volumes 14, 18, 19, and 22.~~ As stated in Appendix 1, the worst-case post-LOCA temperature for these Reactor Building areas is less than ~~110°F~~. Using the Arrhenius equation to equate 180-days at ~~110°F (316.48K)~~ to an equivalent duration at ~~80°F (299.81K)~~ is determined as follows:

3.4 years

$$t_s = 180\text{ days} \times \exp\left[\frac{1.13}{8.617\text{x}10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{\cancel{316.48}}\right)\right] = \cancel{4.9\text{ years}}$$

95°F (308.15K)

The qualified life is then calculated to be ~~62.1 years~~ for the cables ~~in Volumes 14, 18, 19, and 22 of the Reactor Building~~.

60.9 years (64.3 - 3.4)

~~Qualified Life at 90°F (Reactor Building Volume 5)~~

$$\cancel{t_s = 25\text{ hours} \times \exp\left[\frac{1.13}{8.617\text{x}10^{-5}}\left(\frac{1}{305.37} - \frac{1}{385.37}\right)\right] = 21.2\text{ years}}$$

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~~With the 20-years of natural aging, this becomes 41.2 years of expected service life for those cables in Volume 5.~~ The aging in the Reference [2] test was not intended to achieve an end-of-life condition for the cable insulation. ~~The combined 20-years of natural aging and the limited accelerated thermal aging time bounds the original plant design and licensing period of 40 years~~. Other polyethylene insulated and PVC jacketed cables used at Monticello (General Electric) had longer accelerated thermal aging tests. Reference [19], for example exposed polyethylene insulated, PVC jacket control cables to 340 hours at 212°F (373.15K) [19]. ~~Using this test data in the Arrhenius equation yields the following service life at 90°F (305.37K):~~

$$\cancel{t_s = 340\text{ hours} \times \exp\left[\frac{1.13}{8.617\text{x}10^{-5}}\left(\frac{1}{305.37} - \frac{1}{373.15}\right)\right] = 94.7\text{ years}}$$

~~The cables in Volume 5 serve temperature switches that detect and help mitigate a RCIC steam line break (high area temperature). These temperature switches do not function long-term post-HELB and are not required for a LOCA. Therefore, it is reasonable to conclude that there is no significant thermal aging expected at an ambient temperature of 90°F over an extended plant operating time of 60 years (original plant design plus renewed license).~~

APPENDIX 3
THERMAL AGING ANALYSIS

The CR151A212 terminal block is made of wood flour filled phenolic material [6, 8]. The activation energy of wood flour filled phenolic is given as 1.05 eV in Reference [5, p.B-7]. The test report in Reference [3, Page 4] aged the terminal blocks at 150°C for 100 hours prior to LOCA testing. The normal service temperature is ~~80~~°F with a peak accident temperature of ~~115.6°F in Volume 33~~. For conservatism the entire service life and accident period will be calculated at ~~115.6°F~~.

The qualified life of the terminal blocks can be determined utilizing the Arrhenius equation as follows:

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

Terminal blocks only located in RB Volumes 22, 33, and 34. Normal ambient is 85°F (302.59K) while worst-case PLHU temperature is 121.2°F (322.70K)

Where:

t_a = Service Life at ~~115.6°F~~
t_t = Test Time
E_a = activation energy - (1.05 eV)
K = Boltzmann's Constant (8.617×10^{-5})
T_a = Service temperature – ~~115.6°F (319.59K)~~
T_t = Test Temperature - 150°C (423.15K)

Qualified Life = $100 \times \exp\left[\frac{1.05}{8.617x10^{-5}}\left(\frac{1}{\cancel{319.59}} - \frac{1}{423.15}\right)\right]$

Qualified Life = ~~128.8 years @ 115.6°F~~ 89 years at 121.2°F

Therefore, the GE terminal blocks are qualified for >60 years plus the 33 days of post-accident operation.

MONTICELLO NUCLEAR GENERATING PLANT		CA-98-128, Add. 0
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Appendix 1 - Accident Evaluation

EC10413

The UCI-003XS tape splices are located in the RHR and Core Spray Pump Rooms, the SBGT Room, and Steam Chase (Reactor Building Volumes 1, 3, 37 and 16). As indicated in Section C of this EQ calculation, the splices inside the Steam Chase are used on equipment which is only used for post-accident monitoring of Containment Isolation Valve Position Indication for inside Drywell events. As such, they only function under harsh radiological conditions. ~~The medium voltage~~ splices in the RHR and Core Spray Pump Rooms are not exposed to harsh steam temperature, pressure, and humidity conditions but only exposed to post-LOCA heat-up and harsh radiological conditions. This is also true for devices located in the SBGT Room.

The evaluation in this appendix evaluates the post-accident temperature conditions for the installed UCI-003XS splices. The worst-case temperature condition during the time which the subject splices must function exists in Reactor Building Volume 3 of 143.8°F [1]. While operating, a temperature rise effect of 18°C was determined for the RHR and CS pump motor cables/splices [4]. This effect will be considered in this evaluation. The splice specimens were exposed to simulated 30 day accident test. The last portion, and lowest temperature condition, of the test maintained a temperature of 225°F [2, Sheet 122]. Accordingly, it is conservative to extrapolate 30 days at 225°F to the worst-case post-accident temperature of 143.8°F (plus rise), using Arrhenius method, to determine a demonstrated operating time for the splices installed at Monticello.

Letting,

t_e = Equivalent post-accident operating time in years at T_e
t_a = Test duration = 30 days [2, Sheet 122]
T_e = Post-accident temperature = 143.8°F + 18°C rise = 176.2°F (353.26K)
T_a = Test temperature (minimum) = 225°F (380.37K) [2, Sheet 122]
E_a = Activation energy = 1.22 eV [2, Sheet 11 & 226]
K_b = Boltzmann's constant (8.617×10^{-5} eV/K)

$$t_e = t_a \times \exp\left[\frac{E_a}{K_b}\left(\frac{1}{T_e} - \frac{1}{T_a}\right)\right] = 30 \text{ days} \times \exp\left[\frac{1.22}{8.617\text{x}10^{-5}}\left(\frac{1}{353.26} - \frac{1}{380.37}\right)\right]$$
$$= 1.4 \text{ years}$$

EC10413

As such, the UCI-003XS tape splices in Reactor Building Volumes 1, 3, 37 and 16 have demonstrated post-accident operating times significant greater than 180 days (198 days with +10% margin).

The above analysis remains bounding for the medium voltage ECCS pump motor splices in RHR rooms (RB Volumes 1 and 3) under EPU. For all other general splice use at 600 Volt class and lower throughout the Reactor Building, a re-evaluation under EPU conditions is conducted on the next sheet.

13/19

CA-98-128 UCI Tape Splices

Testing for these ssplices included accident testing of 30 days at 225°F

All following analyses use an activation energy of 1.22 eV.

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature	Rise	Service Temp (K)	Equivalent @ 225°F
7 days	180 °F	16 °C	371.37 K	2.84 days
16 days	150 °F	16 °C	354.7033 K	1.08 days
175 days	131 °F	16 °C	344.1478 K	3.48 days
			total	7.40 days

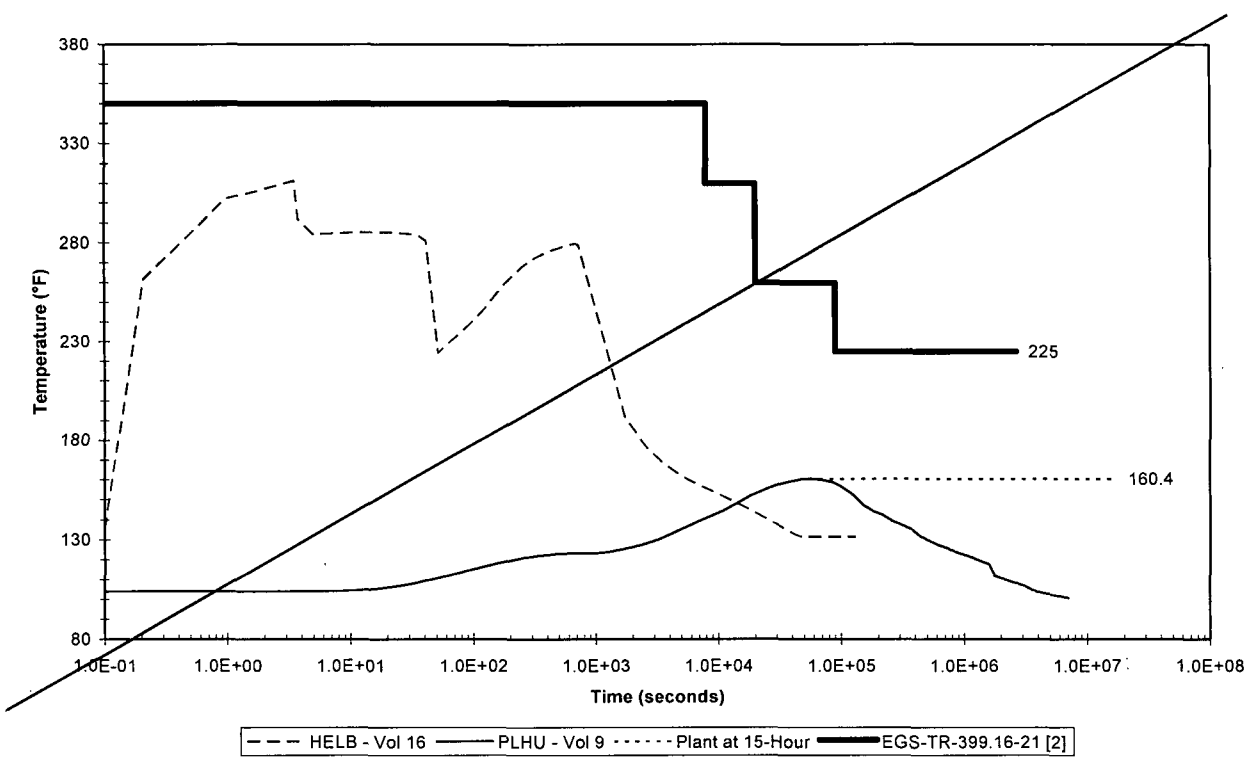
Therefore, testing of 30 days at 225°F will bound the EPU PLHU conditions of the Reactor Building

The temperature rise above was determined in EQ File CA-98-017, Appendix 2 after reviewing energized EQ devices post-accident. The worst-case rise was calculated to be 16°C for the SGTS heater circuit. Thus, using the worst-case rise with the bounding composite Reactor Building PLHU curve under EPU is conservative. The 30 day testing assumed to be at a minimal 225°F confirms that the splices will survive post-LOCA operation anywhere in the Reactor Building.

APPENDIX 1
ACCIDENT DEGRADATION EQUIVALENCY

The composite HELB profile for the Steam Chase (Volume 16) is bounding for all areas of the Reactor Building outside the Drywell. During post-LOCA operation, the highest peak temperature condition at ~~160.4~~ 179.1°F occurs in Reactor Building Volume ~~9~~ 10. The worst-case operating time for any splice installation is assumed to be 198 days (includes 10% margin). The bounding MNGP composite HELB ~~and post-LOCA heat (PLHU) are~~ shown in Figure 1 below along with the accident test profile. The test profile [2/Appendix VII] consisted of a dual transient steam exposure lasting more than 30-days (740 hours, [2, App. VII, Pg. 6]). The first transient, and the first hour of the second transient are not shown in Figure 1 for conservatism. As seen, the test exposure completely envelops both the bounding HELB accident ~~and PLHU~~ profiles of Reactor Building.

Figure 1
Test Versus Bounding Reactor Building Accident Conditions



As seen, the 740 hour test duration does not bound the required 198 days of assumed post-accident operation at MNGP. The last 725 hours (740 hours – 15 hours) of testing at 225°F will be extrapolated to bounding post-LOCA temperature conditions using Arrhenius methods.

APPENDIX 1
ACCIDENT DEGRADATION EQUIVALENCY
(Continued)

Temperature Rise Effect Considerations

Appendix 2 of CA-98-017 [8] (GE Cables) evaluates the expected temperature rise effects in field run cables based on a review of the energized EQML equipment. Equipment that may cause a temperature rise affect in its associated cable, and have the potential to contain the EGS supplied Scotch tape splices, are the motors for V-AC-4, V-AC-5, K-10A/B, V-EF-17A/B, or the SBGT heaters (panels C-87A/B), located in Reactor Building Volumes 1, 3 14, 19, and 36 to 39. The worst-case temperature rise effect determined [8] for this equipment was 16°C under accident/post-accident operation. ~~Per EQ Part B [3],~~ the worst-case post-LOCA heat-up temperature ~~among these areas is less than 145°F (143.8°F for Volume 3). As such, a bounding post-LOCA service temperature for any EQ splice in the Reactor Building would be less than 175°F (145°F + 16°C, rounded up).~~

anywhere in the Reactor Building can be conservatively modeled as 7 days at 180°F, 16 days at 150°F, and 175 days at 131°F. To this profile the +16°C rise can be added for PLHU analysis.

Letting:

t_a = Equivalent aging at 175°F
t_t = Test Time (725 hours)
E_a = Activation energy (1.14 eV) [2/6]
K = Boltzmann's Constant (8.617×10^{-5} eV/K)
T_a = Service Temperature ~~(175°F or 352.59K)~~
T_t = Test Temperature (225°F or 380.37K)

see analysis, next sheet

$$\cancel{t_a = t_t \times \exp\left[\frac{E_a}{K}\left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right] = 725 \text{ hours} \times \exp\left[\frac{1.14}{8.617x10^{-5}}\left(\frac{1}{352.59} - \frac{1}{380.37}\right)\right] = 468 \text{ days}}$$

As such, the EGS testing demonstrates a post-accident operating time greater than the required 198 days.

CA-03-105 Scotch Tape Splices

Testing for these ssplices included accident testing of 725 hours at 225°F

All following analyses use an activation energy of 1.14 eV.

The PLHU temperature throughout the Reactor Building can be conservatively bounded by the following:

Duration	Temperature	Rise	Service Temp (K)	Equivalent @ 225°F
7 days	180 °F	16 °C	371.37 K	72.32 hours
16 days	150 °F	16 °C	354.7033 K	31.00 hours
175 days	131 °F	16 °C	344.1478 K	107.99 hours
			total	211.31 hours

Therefore, testing of 725 hours at 225°F will bound the EPU PLHU conditions of the Reactor Building

The temperature rise above was determined in EQ File CA-98-017, Appendix 2 after reviewing energized EQ devices post-accident. The worst-case rise was calculated to be 16°C for the SGTS heater circuit. Thus, using the worst-case rise with the bounding composite Reactor Building PLHU curve under EPU is conservative. The 725 hour testing assumed to be at a minimal 225°F confirms that the splices will survive post-LOCA operation anywhere in the Reactor Building.

Appendix 2 - Thermal Aging/Qualified Lives

Required: 47 years (See Section D)

Reported: >>47 years [11]

The ASCO temperature switch Test Report No. AQR-020184 / Rev. 1 [11] states that thermal aging simulation was completed on the subject switches. Specifically, the switches were subjected to 210°F for 15 days [11/11]. Reference [9] states that the average temperature of the Standby Gas Treatment Room is ~~80°F~~. The thermal life of the component can be calculated by using the Arrhenius methodology. The Arrhenius methodology is a method of extrapolating the test data at 210°F to the average room temperature at ~~80°F~~ and determine the qualified life of the component at that temperature. The Arrhenius equation is as follows.

90°F

$$t_a = t_t * e^{\left[E_a/K * \left(1/T_a - 1/T_t \right) \right]}$$

Where:

t_a = service life

t_t = test exposure time

E_a = activation energy - eV

K = Boltzmann's Constant (8.617×10^{-5})

T_a = service temperature - °K

T_t = test temperatures - °K

In order to do this evaluation, the limiting activation energy of the pressure switches must be determined. The following table outlines the materials of construction for the temperature switch being evaluated in this file.

Model No. SB11AR / QJ11A4R		
Component	Material	Reference
Microswitch	BZ-2R24-A2	[14]
Diaphragm	Beryllium - Copper	[18]
Body	Aluminum	[16/5]
Capillary / Bulb	Stainless Steel	[16/6]
Fill Fluid	Ethyl - Ether	[11]
Switch Enclosure Seal	BUNA-N	[16/5]
Switch Enclosure	Aluminum (sheet)	[16/5]

Appendix B of References [3] and [11] present a summary of the thermally degradable materials of construction that ASCO considers critical for satisfactory performance of its entire line of pressure and temperature switches. The activation energy for these materials are included in the summary.

The fill fluid was not included in Appendix B of [3] and [11] because it is sealed from the air and is not considered thermally age susceptible [11/19], [14].

The BUNA-N seal cover o-ring was not included in Appendix B of [3] and [11] because it was not considered critical based on qualification of the unit for use outside containment While in many cases, a unit outside containment might be exposed to saturated steam or submergence and the seal could be critical, in this case it is not. Moisture is not a factor in the isolated Standby Gas Treatment Room [9]. For this reason, the BUNA-N seal is considered non-essential to the switch's ability to function.

From the list of materials of construction, it can be seen that the only non-metallic item contained in the temperature transducer portion of the complete unit, is the fill fluid. Since the fill fluid is not thermally degradable [14], the temperature transducer is not susceptible to thermal aging.

The only age susceptible items in the switch mechanism/housing are the BUNA - N seal and microswitch. Since the BUNA-N seal is non-critical, only the microswitch need be evaluated for thermal aging.

From Appendix B of References [3] and [11], the activation energy of the switch is 1.0 eV (Phenolic). This activation energy will be used for thermal aging calculations involving the temperature switch.

Now that the activation energy has been established, the Arrhenius equation can be applied for this scenario. As stated previously, the test specimens were thermally aged for 15 days at 210 °F. Following is the Arrhenius calculation based on an average ambient temperature ~~of 80~~ °F where:

t_a = service life

t_t = 15 days

E_a = 1.0 eV

K = 8.617 x 10^{-5} eV/K

T_a = ~~80°F (299.81K)~~ 90°F (305.37K)

T_t = 210°F (372.04K)

Therefore:

37.2

$$t_a = 15\text{ days} \times \exp\left[\frac{1.0}{8.617\text{x}10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{372.04}\right)\right] = \cancel{75.4}\text{ years}$$

90°F (305.37K)

POST LOCA HEATUP

The maximum temperature resulting from the Post LOCA Heatup conditions is ~~134.4°F (330.04K) in Volume 36.~~ For conservatism, it will be assumed that this temperature exists for the entire 180 days post accident. The thermal degradation resulting from this temperature increase will be subtracted from the previously calculated qualified life to account for the Post LOCA Heatup.

The thermal degradation for the accident is calculated as follows using the Arrhenius Method:

7.6

$$t_a = 180\text{ days} \times \exp\left[\frac{1.0}{8.617\text{x}10^{-5}}\left(\frac{1}{\cancel{299.81}} - \frac{1}{\cancel{330.04}}\right)\right] = \cancel{17}\text{ years}$$

Therefore, the total qualified life is as follows:

132.6°F (329.04K) in RB Volumes 36/39

~~**75.4 years - 17.0 years = 58.4 years**~~

~~**This is in excess of the 47 years required for the remaining design life of the plant, including extended plant operation under a renewed license. The thermal aging requirement is considered satisfied.**~~

37.2 years - 7.6 years = 29.6 years.

Note: the temperature switches were installed in early 1984 and would require replacement in 2012. The reduced qualified life is being tracked under the Corrective Action Program (CAP 01106163), no further action need for EPU as this is a current plant issue with normal abient temperature change. The CAP derived qualified life was 28.8 years based on the higher CLTP post-accident temperature of 134.4°F. The qualified life is thus increased slightly using the PLHU temperature calculated under EPU conditions.

APPENDIX 3
THERMAL AGING ANALYSIS

Thermal Aging Analysis

Since there is no test data available for this equipment, thermal aging was addressed by evaluating the critical non-metallic materials of construction [9] susceptibility to thermal degradation. A literature search was performed to identify the thermal properties of each non-metallic as evaluated below.

Thermal Aging Analysis of Micro Switch OP-AR Limit Switch					
Component	Material	Activation Energy (eV)	Ref	Thermal Aging Data (Hrs @ °F)	Ref
BZ switch block [9]	GP Black Phenolic	1.05	[2]	1000 @ 190°C	[20]
Electrical Insulator	Black Varnish Fiberglass	0.73	[2]	4 years @ 109°C	[2]
Housing Seal	Synthetic Rubber	Performs no critical function - See Appendix 2			
Plunger Seal	Buna-N	Performs no critical function - See Appendix 2			

Material Analysis

Black Varnish Fiberglass

The fiberglass cloth is used as an electrical insulator and is not susceptible to significant thermal aging. The thermal capability of varnish (epoxy) is based on a 5% deterioration per year at a 109°C continuous duty temperature [2, p.6-20]. This gives a 4 year life assuming that 20% deterioration constitutes end of life. The lowest activation energy presented for use of epoxy as an insulation is 0.73 eV [2, p.B-5].

GP Phenolic

The GP Phenolic is used to make the switch case and cover where retention of physical strength is important. The activation energy is based on 50% retention of impact strength [2, p.B-7]. The thermal aging data is based on Arrhenius curve for Durez 791 black wood/flour filled general purpose phenolic (1/8" sample – flexural strength) [20, p.C-8].

Service Life Determination

The service life for the Black Varnish Fiberglass and GP Phenolic at the normal service temperature of ~~80°F~~ 90°F in the SGTS Room, is determined by using the Arrhenius equation as follows:

$$t_a = t_t * e^{\left[\frac{E_a}{K} * \left(\frac{1}{T_a} - \frac{1}{T_t}\right)\right]}$$

90°F (305.37K)

Where: t_a = Equivalent aging at ~~80°~~F
t_t = Thermal Life Time
E_a = activation energy - (1.05, 0.73 eV)
K = Boltzmann's Constant (8.617 x 10^{-5})
T_a = Equivalent temperature - ~~299.82°K (80°F)~~
T_t = Thermal Life temperatures - °K

The result are:

Service Life for Black Varnish Fiberglass = ~~1759.8~~ years @ ~~80°F~~
Service Life for GP Phenolic = ~~193766~~ years @ ~~80°F~~

>1,000

Qualified Life Determination

The qualified life of the pressure switch is based on the weak-link material, Black Varnish Fiberglass with a service life of 1759.8 years. The qualified life is given as:

132.6°F (329.04K)

Qualified Life = Service Life - Accident Aging

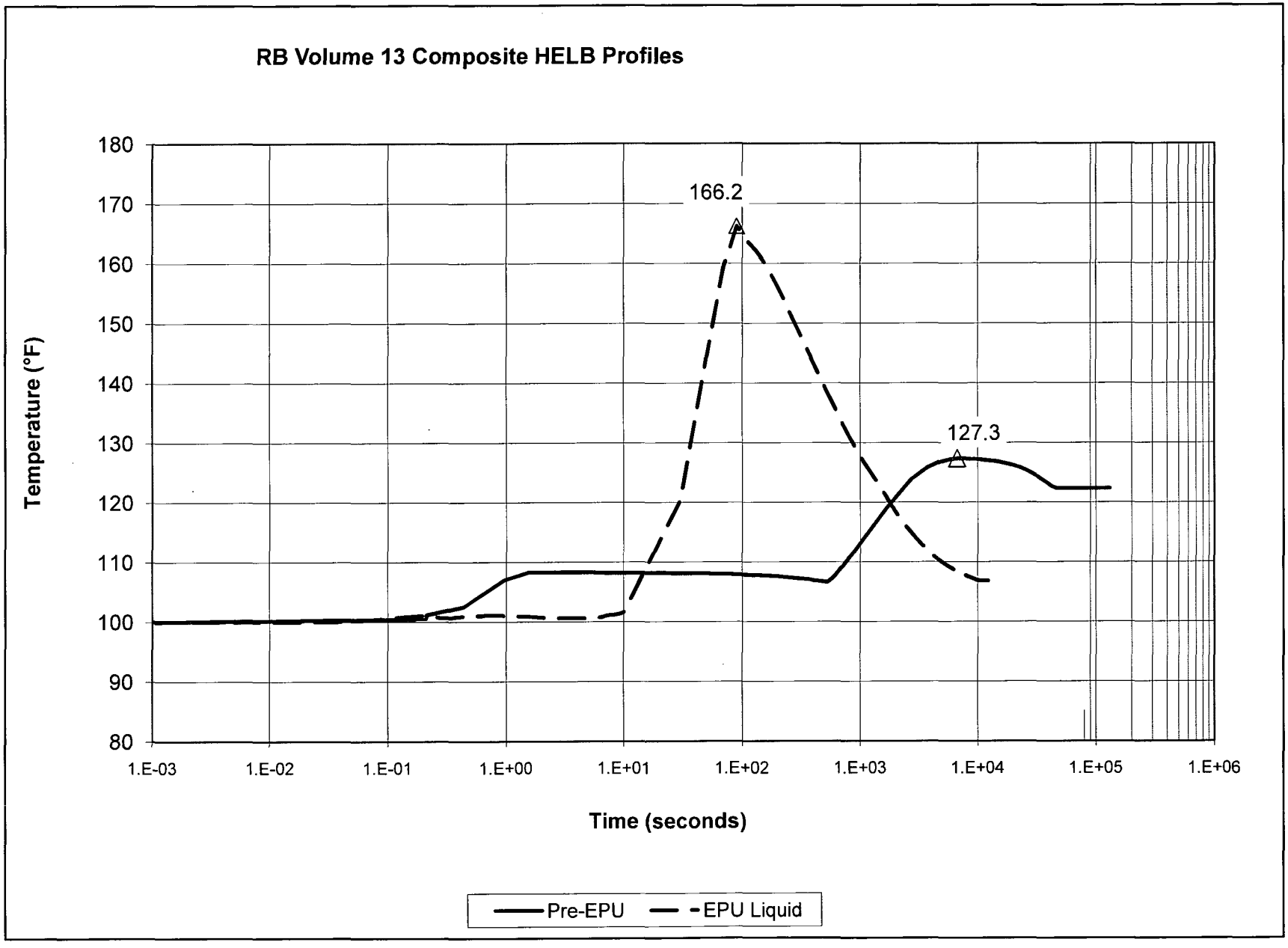
The Post-LOCA Heatup conditions in the SGTS Rooms is assumed as ~~134.4°F (330.04K)~~ for 180 days in Appendix 1. This is conservative since ~~134.4°F is actually only~~ the peak temperature ~~and~~ does not last for 180 days. The equivalent aging for the Black Varnish Fiberglass is given as:

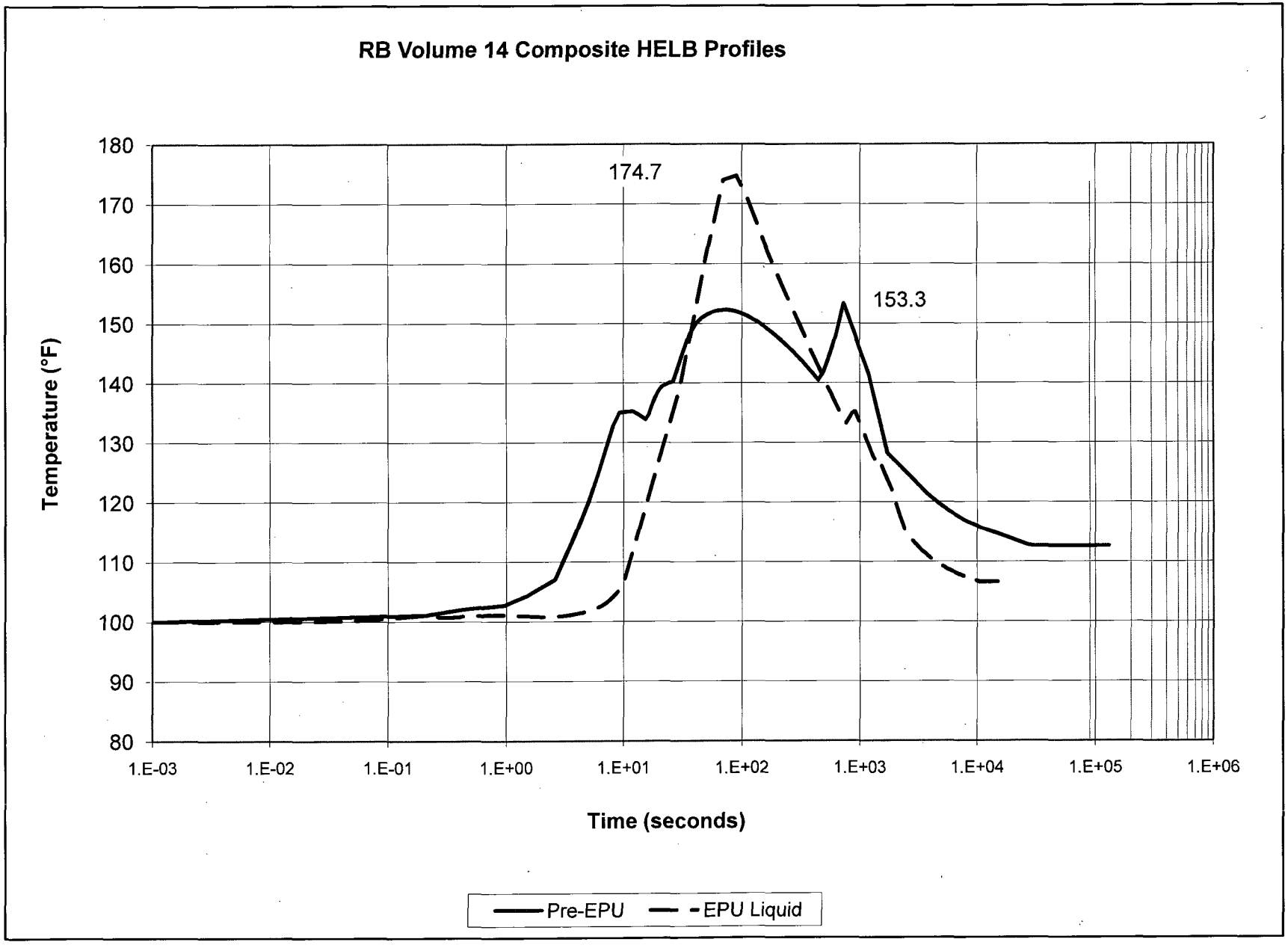
$$t_a = 4320 \text{ hours} \times \exp\left[\frac{0.73}{8.617\text{x}10^{-5}}\left(\frac{1}{299.82} - \frac{1}{330.04}\right)\right] = 6.6 \text{ years}$$

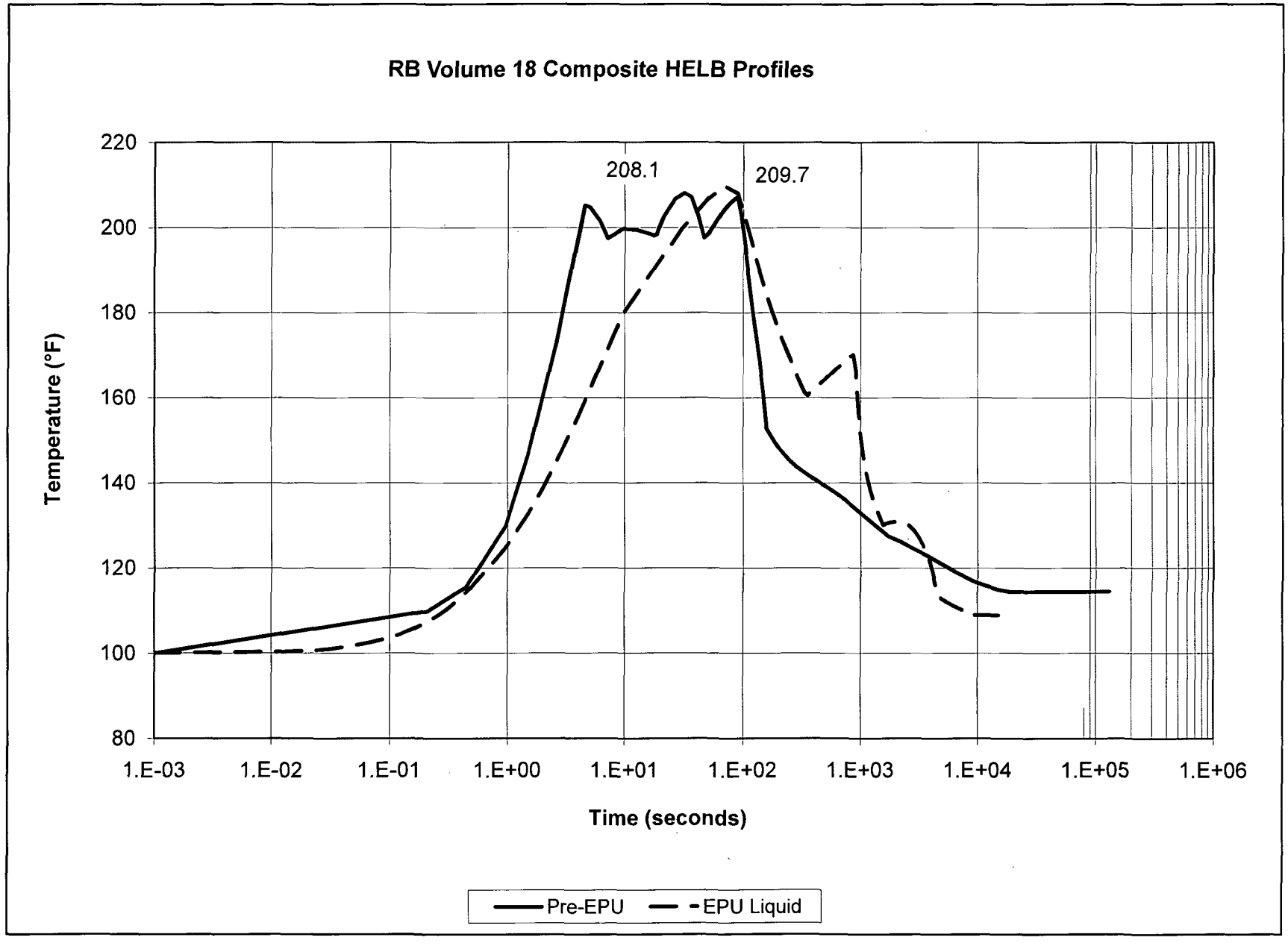
Therefore the qualified life is equal to:

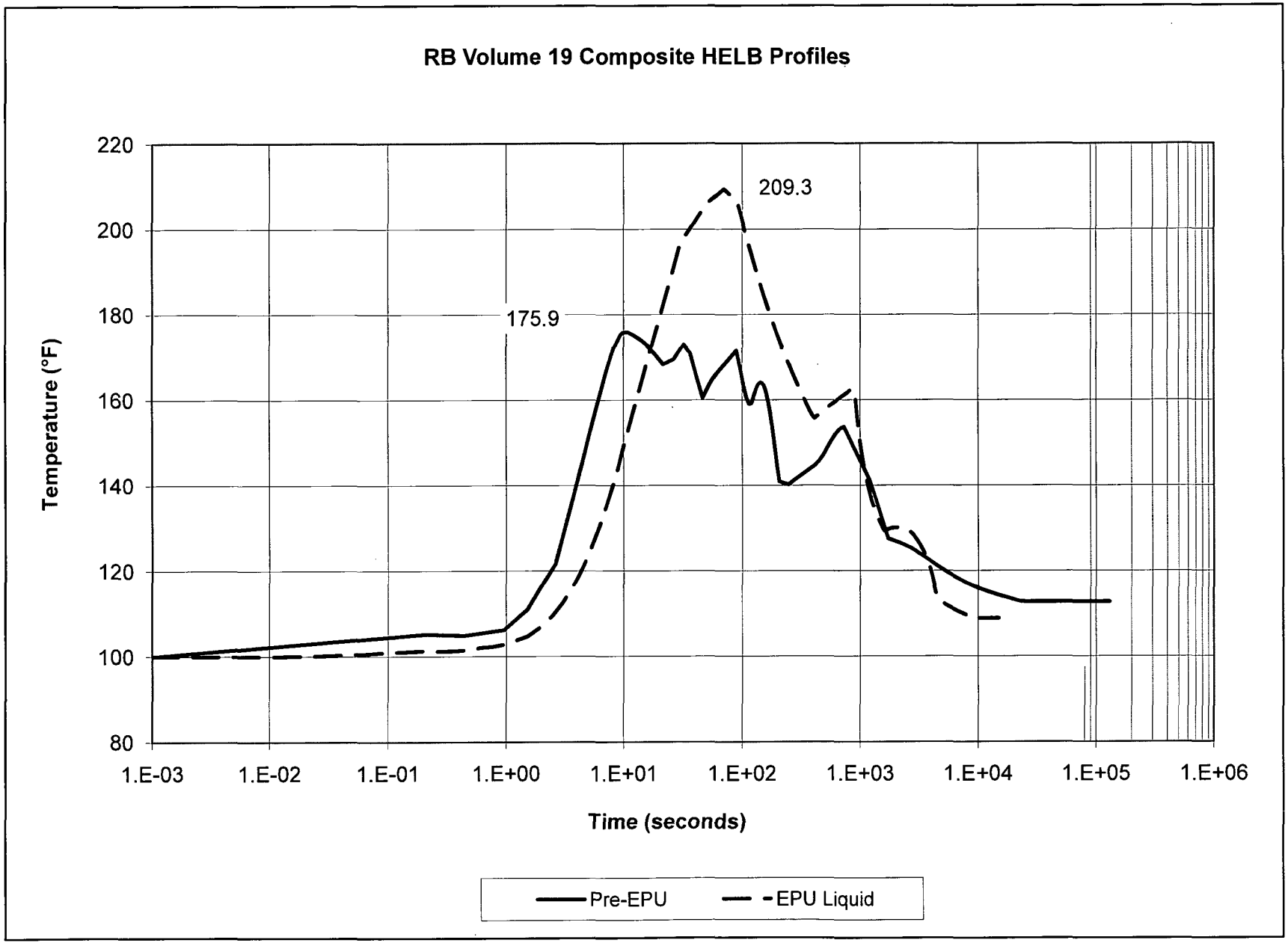
Qualified Life = ~~1759.8 - 6.6~~ > 100 years

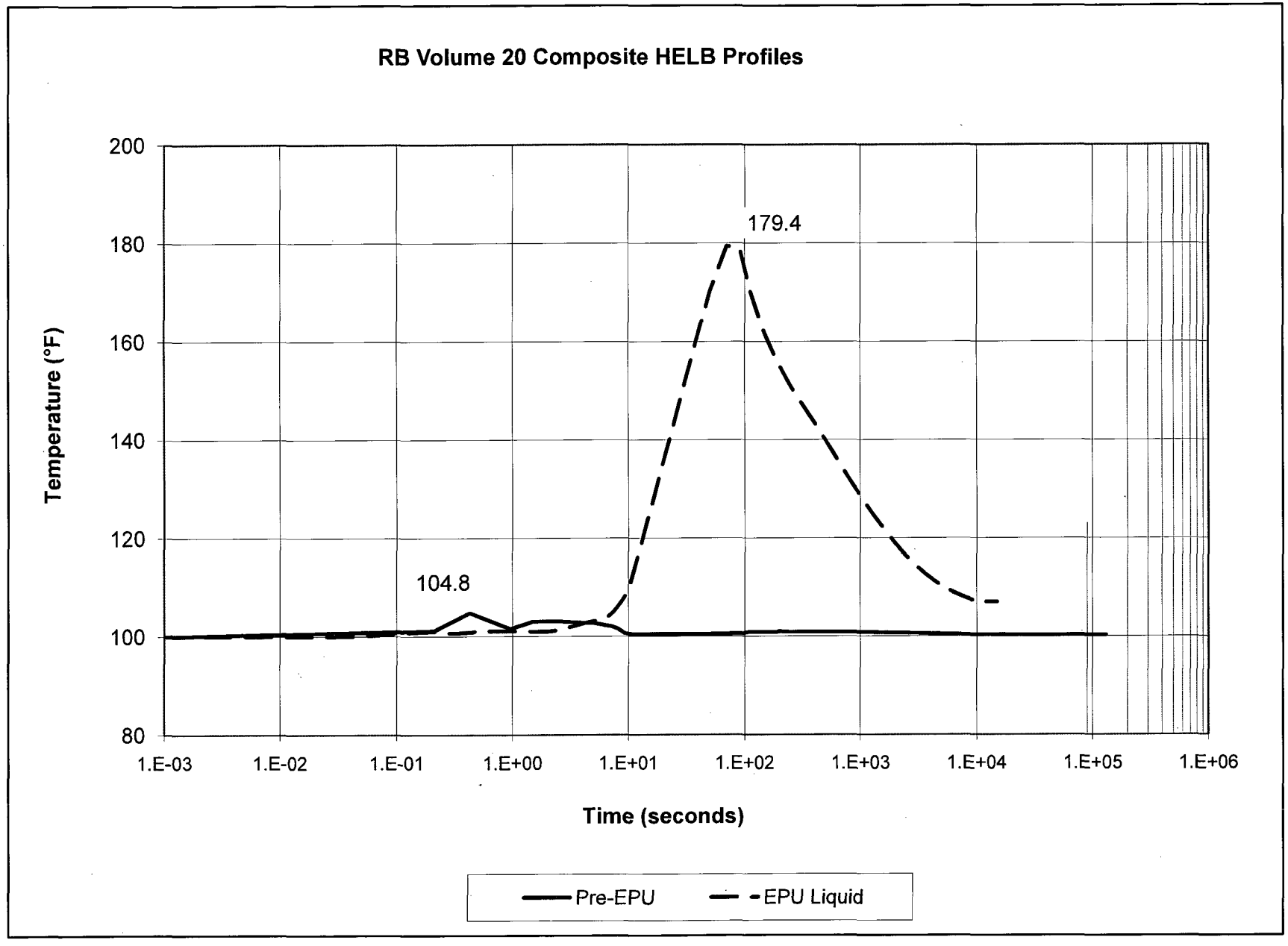
Therefore, Micro Switch OP-AR limit switches are qualified for 60 years plus 180 days of post accident conditions.

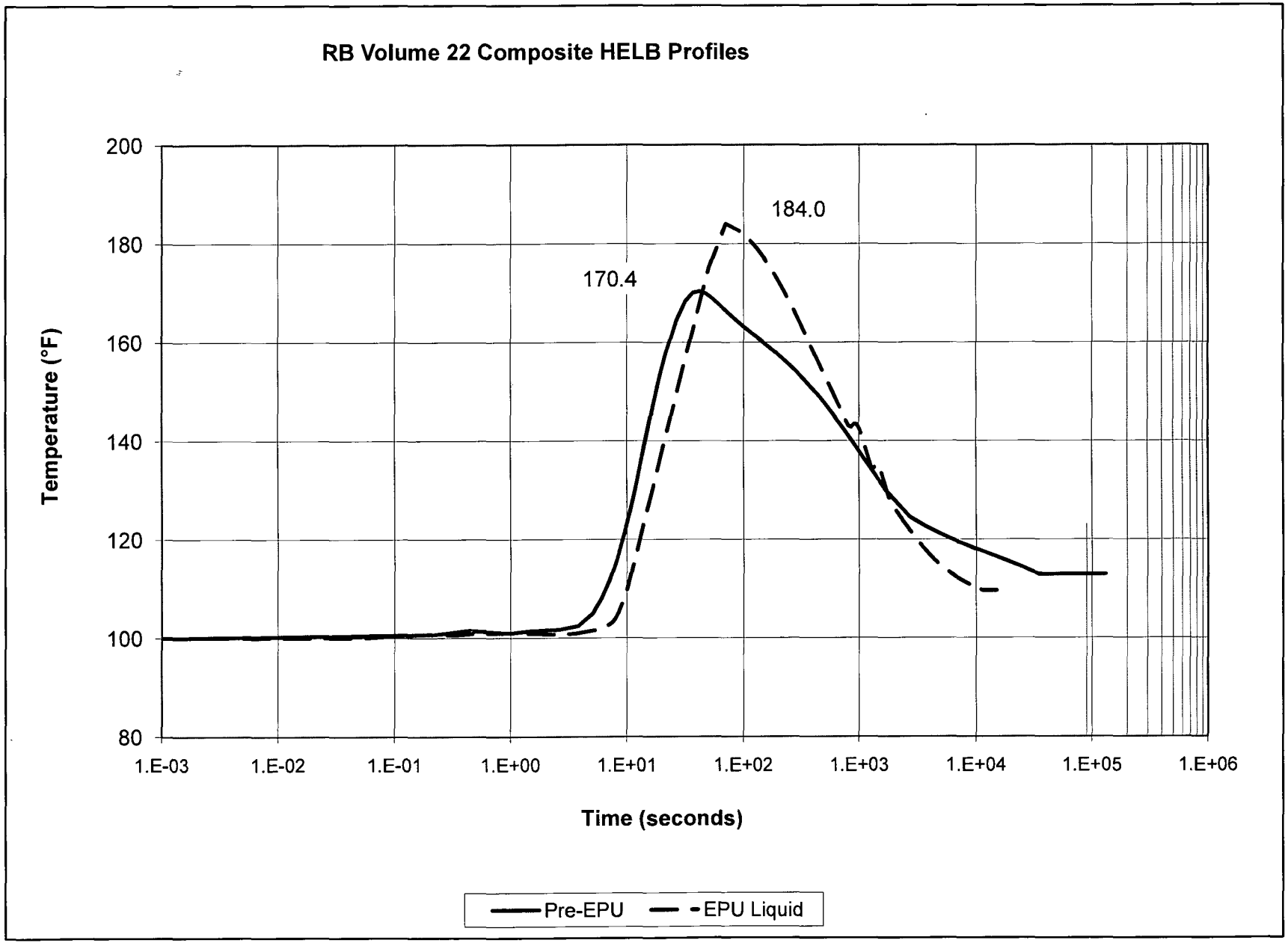

RB Volume 13 Composite HELB Profiles
Temperature (°F)
180
170
160
150
140
130
120
110
100
90
80
166.2
127.3
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid

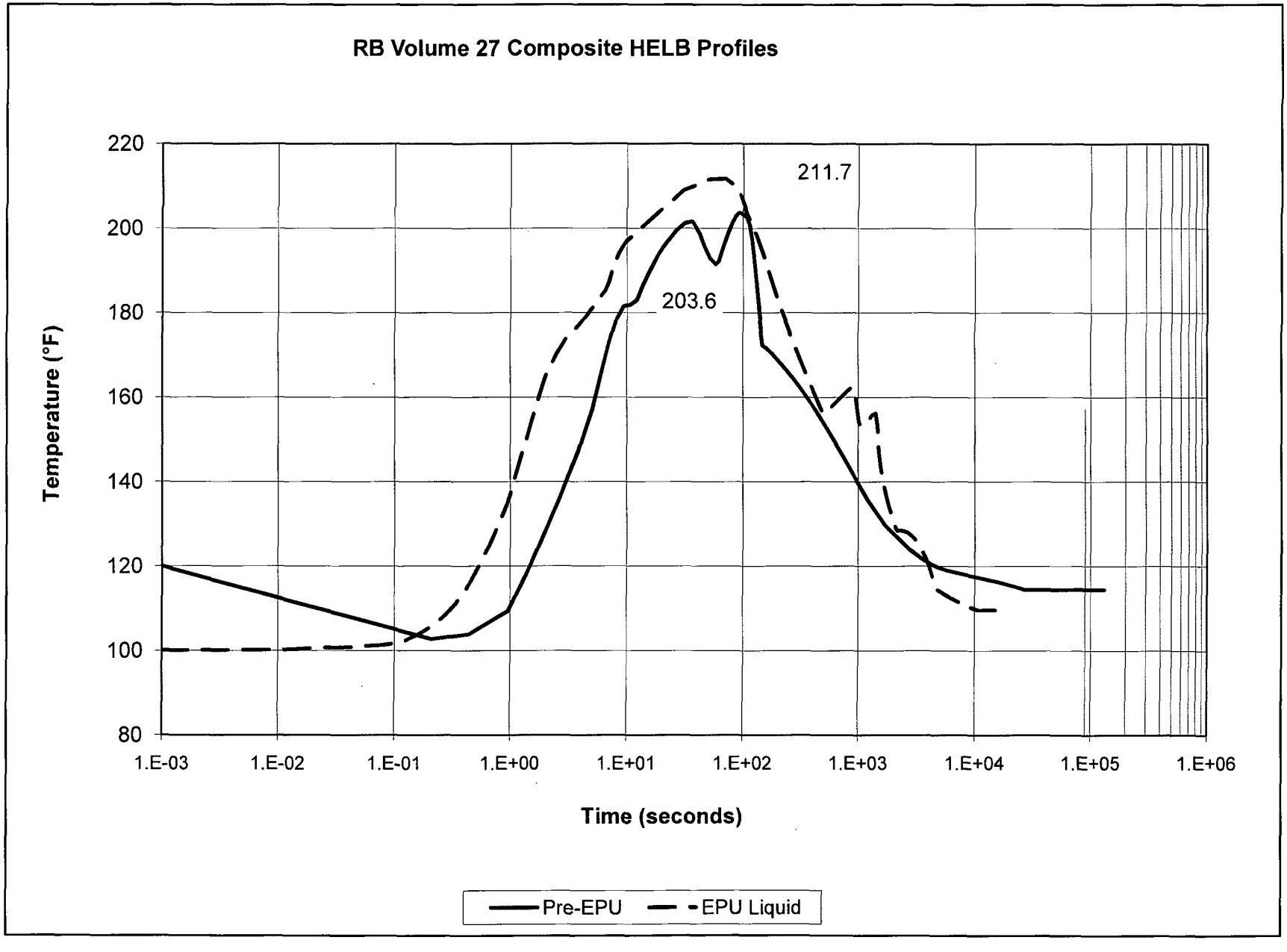

RB Volume 14 Composite HELB Profiles
Temperature (°F)
180
170
160
150
140
130
120
110
100
90
80
174.7
153.3
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid

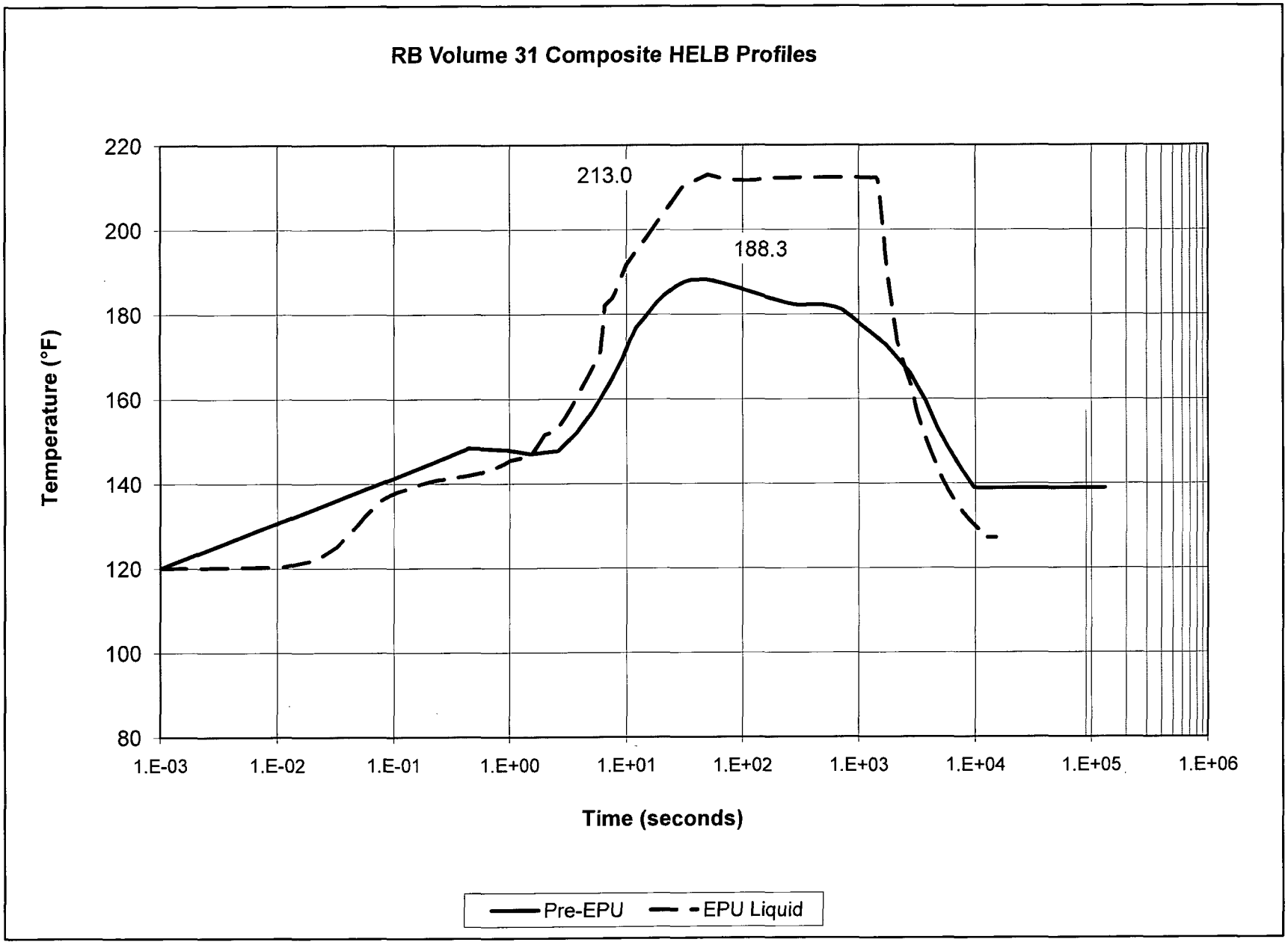

RB Volume 18 Composite HELB Profiles
Temperature (°F)
220
200
180
160
140
120
100
80
208.1
209.7
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid

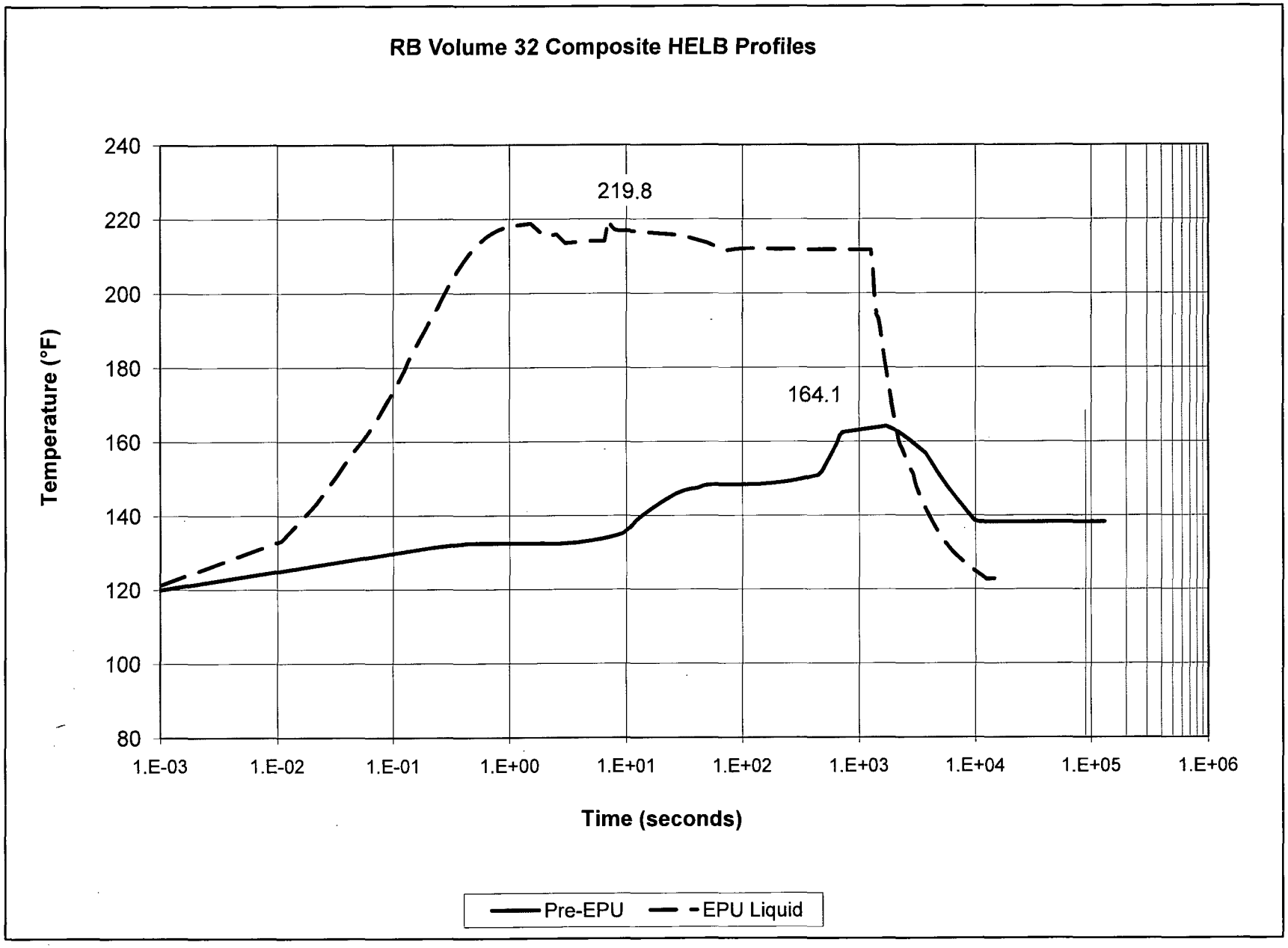

RB Volume 19 Composite HELB Profiles
Temperature (°F)
220
200
180
160
140
120
100
80
209.3
175.9
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid

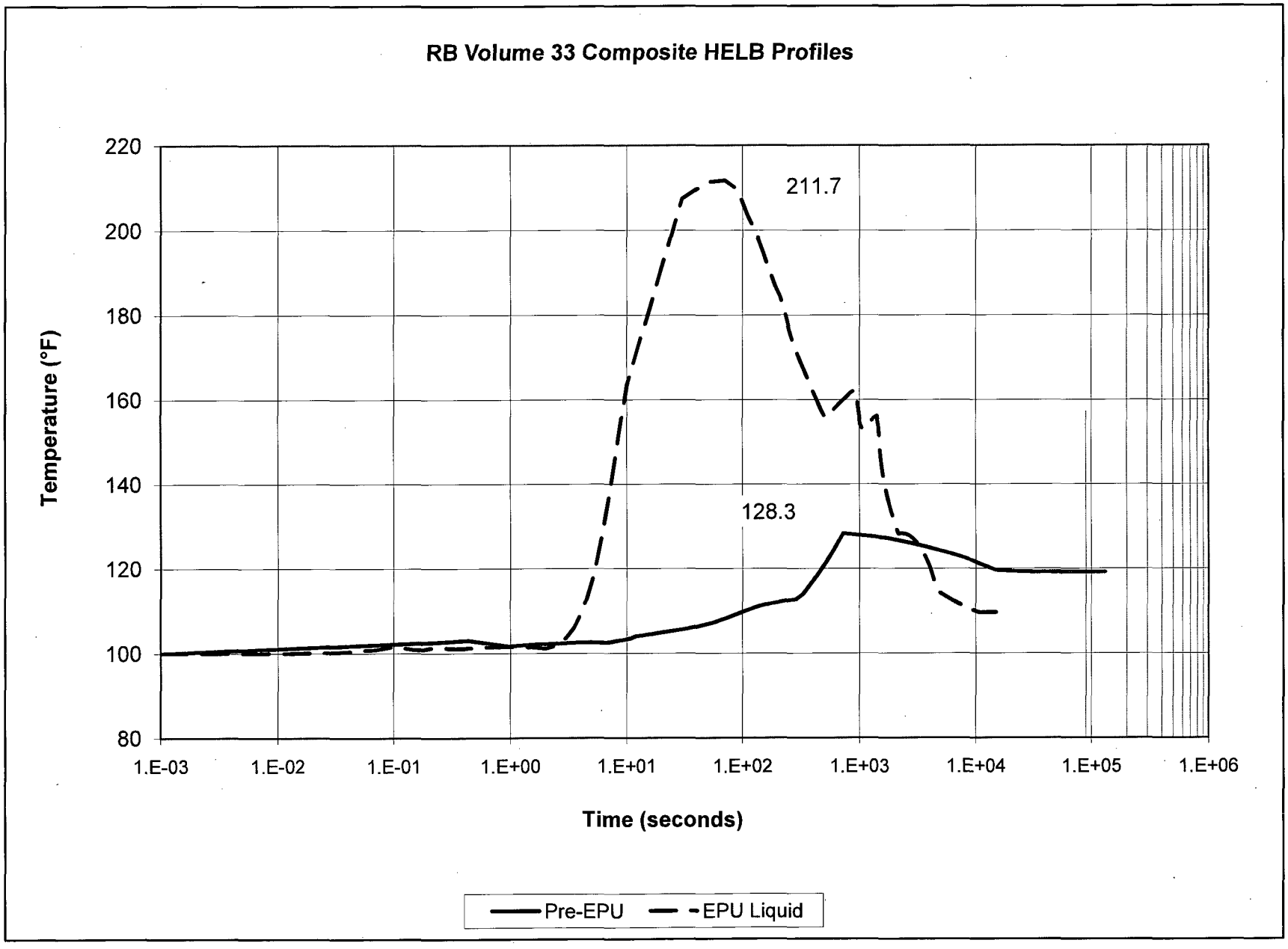

RB Volume 20 Composite HELB Profiles
Temperature (°F)
200
180
160
140
120
100
80
179.4
104.8
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid

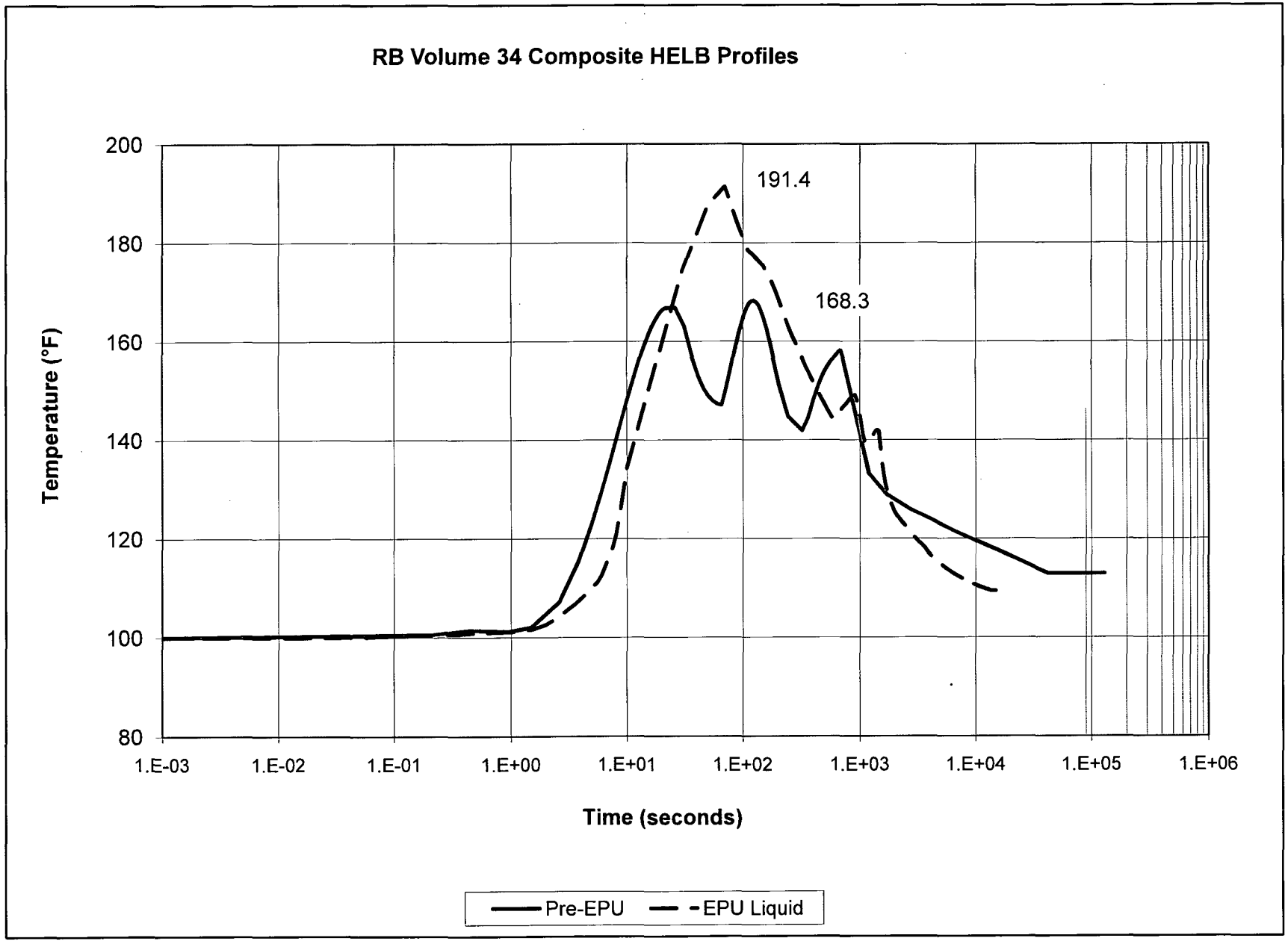

RB Volume 22 Composite HELB Profiles
Temperature (°F)
200
180
160
140
120
100
80
184.0
170.4
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid


RB Volume 27 Composite HELB Profiles
Temperature (°F)
220
200
180
160
140
120
100
80
211.7
203.6
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid


RB Volume 31 Composite HELB Profiles
Temperature (°F)
220
200
180
160
140
120
100
80
213.0
188.3
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid


RB Volume 32 Composite HELB Profiles
Temperature (°F)
240
220
200
180
160
140
120
100
80
219.8
164.1
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid


RB Volume 33 Composite HELB Profiles
Temperature (°F)
220
200
180
160
140
120
100
80
211.7
128.3
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid


RB Volume 34 Composite HELB Profiles
Temperature (°F)
200
180
160
140
120
100
80
191.4
168.3
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
1.E+03
1.E+04
1.E+05
1.E+06
Time (seconds)
Pre-EPU
EPU Liquid