ATTACHMENT 7

Braidwood and Byron Stations

Measurement Uncertainty Recapture Technical Evaluation

(NRC Regulatory Information Summary 2002-03 Requested Information)

(Non-Proprietary Version)

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ACRONYM LIST

Expression	Definition or Use
AFW	auxiliary feedwater
AMS	ATWS mitigation system
ANS	American Nuclear Society
AOR	analysis of record
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
AST	alternate source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AVB	anti-vibration bar
B&PV	Boiler and Pressure Vessel
BIT	boron injection tank
BOL	beginning of life
BOP	balance of plant
B&W	Babcock & Wilcox
CEDE	committed effective dose equivalent
CLTP	current licensed thermal power
CPT	critical power trajectory
CR	control room
CRDM	control rod drive mechanism
CREA	control rod ejection accident
CVCS	chemical and volume control system
CWO	core-wide oxidation
DBE	design basis earthquake
DCF	dose conversion factors
DE	dose equivalent
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DRLL	dropped rod limit lines
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EFPY	effective full-power year
EOL	end of license
EPRI	Electric Power Research Institute
ESF	engineered safety features
ESFAS	engineered safety feature actuation system

ACRONYM LIST (cont.)

FIV	flow-induced vibration
FNΔH	nuclear enthalpy rise hot channel factor
FON	fraction of nominal
FWCV	feedwater control valve
FWIV	feedwater isolation valve
HFP	hot full power
HGR	heat generation rate
HHSI	high-head safety injection
HLSO	hot leg switchover
HZP	hot zero power
LBA	licensing basis analysis
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LCP	Lower Core Plate
LOCA	loss-of-coolant accident
LMO	local metal oxidation
LPZ	low population zone
LOOP	loss of offsite power
LRA	locked rotor accident
LTCC	long-term core cooling
M&E	mass and energy
MSIV	main steam isolation valve
MSS	main steam system
MSSV	main stem safety valve
MTC	moderator temperature coefficient
MTO	margin to overfill
MUR-PU	measurement uncertainty recapture power uprate
NRC	Nuclear Regulatory Commission
NRS	narrow range span
NSR	non-safety related
NSSS	nuclear steam supply system
ODSCC	outside diameter stress corrosion cracking
OPΔT	overpower delta-T
OTΔT	overtemperature delta-T
P_a	peak containment internal pressure (10CFR50, Appendix J)
PCM	percent millirho
PCT	peak cladding temperature
PLHR	peak linear heat rate

ACRONYM LIST (cont.)

PMTC	positive moderator temperature coefficient
PORV	power-operated relief valve
PTLR	pressure and temperature limits report
PTS	pressurized thermal shock
PSV	pressurizer safety valve
PUGR	power uprate growth rate
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPV	reactor pressure vessel
RSAC	Reload Safety Analysis Checklist
RSG	replacement steam generator
RTDP	revised thermal design procedure
RTP	rated thermal power
RT_{PTS}	reference temperature pressurized thermal shock
RWAP	rod withdrawal at power
RWST	refueling water storage tank
RWFS	RCCA withdrawal from subcritical
SAFDL	specified acceptable fuel design limit
SAL	safety analysis limit
SAT	station auxiliary transformer
SBLOCA	small-break loss-of-coolant accident
SBO	station black out
SG	steam generator
SGBS	stream generator blowdown system
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SI	safety injection
SLB	steam line break
STDP	Standard Thermal Design Procedure
TEDE	total effective dose equivalent
TDF	thermal design flow
T_{FW}	feedwater temperature
TPR	thimble plug removal
TRM	Technical Requirements Manual
TSP	tube support plate
TTS	top of tubesheet

ACRONYM LIST (cont.)

UAT	unit auxiliary transformer
UCP	upper core plate
UET	unfavorable exposure time
UFM	ultrasonic flow meter
UFSAR	Updated Final Safety Analysis Report
UPS	uninterruptable power supply
USE	upper shelf energy

INTRODUCTION

This attachment contains the Exelon responses to the NRC Regulatory Issue Summary 2002-03, requested information for MUR power uprates. The LAR attachment sections match the NRC Regulatory Issue Summary 2002-03, sections for ease of review.

I FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

1. *A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. The description should include:*

   A. *Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique*

   B. *A reference to the NRC's approval of the proposed feedwater flow measurement technique*

   C. *A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique*

   D *The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique*

   E. *A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty*

   F. *Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:*
      i. *maintaining calibration*
      ii. *controlling software and hardware configuration*
      iii. *performing corrective actions*
      iv. *reporting deficiencies to the manufacturer*
      v. *receiving and addressing manufacturer deficiency reports*

   G. *A proposed allowed outage time for the instrument, along with the technical basis for the time selected.*

   H. *Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level*

RESPONSE TO I FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

I.1 Implementation of the Feedwater Ultrasonic Flow Meter

Byron and Braidwood Stations will be utilizing the Cameron (formerly Caldon) CheckPlus Leading Edge Flow Meter (LEFM) system ultrasonic multi-path, transit time flowmeter on their main feedwater lines. This system provides highly accurate feedwater flow and temperature measurements that will reduce the measurement uncertainty to each Unit's reactor thermal power (heat balance) calculation. This reduced measurement uncertainty supports increasing each Unit's current licensed rated thermal power (RTP) of 3586.6 MWt by approximately 1.63% to 3645 MWt.

I.1.A Cameron Topical Reports Applicable to the LEFM CheckPlus System

The referenced Topical Reports are:

- Cameron Engineering Report ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level using the LEFM Check System,* Caldon Inc., March 1997 (Reference I-1)
- Topical Report (TR) Engineering Report ER-157P, Revision 8, *Caldon Ultrasonics Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,"* dated May 11, 2009 (Reference I-2)

I.1.B NRC Approval of Cameron LEFM CheckPlus System Topical Reports

The NRC approved the Topical Reports referenced in I.1.A above in the following documents:

- Review of Caldon Engineering Topical Report ER 80P, *Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System*, March 8, 1999 (Reference I-3)
- Final Safety Evaluation by the Office of Nuclear Reactor Regulation Engineering Report ER-157P Topical Report, Revision 8, *Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System*, Cameron Measurement Systems Project NO. 1370; dated August 16, 2010 (Reference I-4)

I.1.C Implementation of Guidelines and NRC SER for the Cameron LEFM CheckPlus System

The LEFM CheckPlus system will be installed and operated in accordance with in the manufacturer's requirements as described in Topical Reports ER-80P (Reference I-1) and ER-157P (Reference I-2). The system will be used for continuous calorimetric power determination by direct links with the each Unit's plant computer. Even though the LEFM CheckPlus system is not safety-related it is designed and manufactured in accordance with Cameron's Quality Assurance Program, which conforms to 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The LEFM spool pieces will be installed in Byron and Braidwood, Units 1 and 2, 16-inch feedwater piping as shown in Attachment 11 to this License Amendment Request (LAR). The installation location is downstream of the common feedwater header where the lines split into four straight sections of piping.

The spool pieces are installed sufficiently upstream of the existing feedwater flow venturis and downstream of any piping components such that no adverse interactions are created. No flow straighteners will be installed.

Each of the Unit's LEFMs was calibrated at the Alden Research Laboratory facility using a hydraulic duplicate of the principal hydraulic features of the plant configuration. The calibration tests determined the meter factor (a.k.a. meter calibration constant) for each of the Unit's LEFMs. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests were also performed at the Alden Research Laboratory facility to determine meter factor sensitivity to upstream hydraulics. Copies of the Unit specific Meter Factor Calculation and Accuracy Assessments (References I-5a through 5d) based on the Alden Laboratory test results are provided in Appendix A.3 of Attachments 8a through d of the this LAR.

Both the transducers and the electronics cabinet will be located in the Main Steam Line Tunnel. The electronics cabinet is provided with its own cooling system, comprised of two air conditioners. The air conditioners serve to maintain suitable ambient conditions internal to the electronic cabinet. Under normal full power conditions the transducers will not be exposed to any radiation. However tests have been conducted on the PZT-5A piezoceramic material in the transducers where it has been exposed to gamma radiation on the order of 10^7 roentgens. The tests concluded that there were negligible losses of material properties even at that high exposure level. Based on the above, no damage or degradation to the instruments is anticipated due to the ambient conditions or radiation exposure in the area of installation.

In approving Cameron Engineering Report 157-P the NRC stated that licensees can reference TR ER-80P and follow the example of ER-157P, Revision 8, for their plant-specific analyses subject to meeting five qualifications (Reference I-4). The five qualifications are listed below along with a discussion of how each will be satisfied for Byron and Braidwood Stations, Units 1 and 2:

NRC Qualification 1:

> *Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.*

Response: As described in Section I.1.G operation above 3586.6 MWt will be limited to 72 hours if the LEFM CheckPlus is not operable.

NRC Qualification 2:

> *A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.*

Response Byron and Braidwood Stations will not apply a secondary condition with LEFM CheckPlus in a degraded condition with increased uncertainty. As described further in Section I.1.G, Byron and Braidwood Stations will conservatively respond to a single path or single plane failure in the LEFM CheckPlus in the same manner as a complete system failure and operation above 3586.6 MWt will be limited to 72 hours.

NRC Qualification 3:

> *An applicant with a comparable geometry can reference the [Section 3.2.1 in Reference I-4] finding to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.*

Response: The downstream piping geometry concerns described in Section 3.2.1 of the NRC SE (Reference I-4) are not applicable to Byron and Braidwood Stations. As discussed above, the LEFM spool pieces will be installed in straight sections of feedwater piping and have been tested at Alden Laboratories in hydraulically equivalent configurations. Additionally, as described in response to Qualification 2 above, Byron and Braidwood Stations do not propose to apply a secondary condition to allow use of the LEFM CheckPlus with an increased uncertainty (in a "Check" equivalent mode).

NRC Qualification 4

> *An applicant that requests a MUR with the upstream flow straightener configuration discussed in [Section 3.2.2 in Reference I-4] should provide justification for claimed CheckPlus uncertainty that extends the justification provided in [Reference 17 in Reference I-4]. Since the [Reference 17 in Reference I-4] evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.*

Response: The feedwater piping configurations at Byron and Braidwood Stations do not necessitate or use upstream flow straighteners.

NRC Qualification 5

> *An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in [Figure 1 of Reference 18 in Reference I-4].*

Response: The uncertainty associated with steam enthalpy due to moisture content for Byron Units 1 and 2 and Braidwood Units 1 and 2 respectively are ±0.0034%, ±0.0061%, ±0.0021%, and ±0.0044%. These values are based on actual in-plant moisture carryover tests. As can be seen in Table I-1 these uncertainty values are relatively small in comparison to the other uncertainties associated with the power uncertainty calculation therefore this qualification is not applicable to Byron or Braidwood Station.

I.1.D Disposition of NRC SER Criteria During Installation

In approving Cameron Topical Reports ER-80P (Reference I-3) and ER-157P (Reference I-4), and also in Reference I-6 the NRC established four criteria each licensee must address. Exelon Generation Corporation's (EGC's) response to those criteria is provided as follows:

I.1.D.1 NRC Criterion 1

Discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurement and plant operation.

Response to NRC Criterion 1

Implementation of the MUR power uprate license amendment will include developing the necessary procedures and documents required for continued calibration and maintenance of the LEFM system. Plant maintenance and calibration procedures will be revised to incorporate Cameron's maintenance and calibration requirements prior to raising power above the current licensed thermal power (CLTP) of 3586.6 MWt. The Byron and Braidwood Station Technical Requirement Manuals (TRM) will be revised as discussed in Sections I.1.G and H below, and in Attachment 1 to the LAR to address contingencies for inoperable LEFM instrumentation.

A modification package has been developed for each installation outlining the steps to install and test the LEFM CheckPlus system. When each unit is shutdown for their respective refueling outages, as delineated in the schedule provided in the License Amendment Request cover letter, the LEFM CheckPlus systems will be installed. Following installation, testing will include an in-service leak test, comparisons of feedwater flow and thermal power calculated by various methods, and final commissioning testing. The LEFM CheckPlus system installation and commissioning will be performed according to Cameron procedures. Commissioning and start-up of the LEFM CheckPlus System will be performed by qualified Cameron personnel with site personnel assistance. The commissioning process provides final positive confirmation that actual field performance meets the uncertainty bounds established for the instrumentation. Final site-specific uncertainty analyses acceptance will occur after completion of the commissioning process.

The Byron and Braidwood Stations LEFM CheckPlus system was calibrated in a site-specific model test at Alden Research Laboratory. A copy of the Alden Research Laboratory certified calibration report is contained in the Cameron Meter Factor Reports (LAR Attachments 8a through 8d, Appendix A.3). The testing at Alden Laboratory provides traceability to National Standards. The spool piece calibration factor uncertainty is based on these Cameron engineering reports. The calibration tests included a site-specific model of each of the Units hydraulic geometry. The installations at Byron and Braidwood Stations do not require and will not employ upstream flow straighteners. A discussion of the impact of plant-specific installation factors on the feedwater flow measurement uncertainty is also provided in Attachments 8a through 8d.

Preventive maintenance will be performed based on vendor recommendations. The preventive maintenance program and LEFM CheckPlus system continuous self-monitoring feature ensure that the LEFM remains bounded by the Topical Report ER-80P (Reference I-1), as supplemented by ER-157P (Reference I-2), analysis and assumptions. Establishing and continued adherence to these requirements assures that the LEFM CheckPlus system is properly maintained and calibrated. The preventive maintenance activities will be identified via the associated plant modification package. Typical activities

performed include power supply checks, pressure transmitter checks, and clock verifications. Maintenance of the LEFM system will be performed by personnel who are qualified on the LEFM.

Instrumentation, other than the LEFM system, that contributes to the power calorimetric computation will be periodically calibrated and maintained using existing site procedures. Maintenance and test equipment, tolerance settings, calibration frequencies, and instrumentation accuracy were evaluated and accounted for in the thermal power uncertainty calculation.

I.1.D.2 NRC Criterion 2

> *For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system, and bounds the analysis and assumptions set forth in Topical Report ER-80P.*

Response to NRC Criterion 2

At the time of this submittal, only Byron Station Unit 1 and Braidwood Station Unit 2 have installed the LEFM CheckPlus systems. Based on the results of the modification and commission testing the LEFM CheckPlus system as installed is in conformance with the analysis and assumptions given in Cameron's Topical Report ER-80P (Reference I-1), ER-157P (Reference I-2), and the Byron and Braidwood unit specific "Bounding Uncertainty Analysis for Thermal Power Determination Reports" (References I-7a through d), as well as the performance parameters identified in the Alden Laboratory Meter Factor Calculation and Accuracy Assessments (References I-5a through 5d). As of June 17, 2011, there have been no performance, operational, or maintenance issues that would indicate any non-conformance with the above.

I.1.D.3 NRC Criterion 3

> *Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both venturi and the LEFM for comparison.*

Response to NRC Criterion 3

Cameron has performed Unit specific bounding uncertainty analysis for Byron and Braidwood Stations, Unit 1 and 2 (References I-7a through 7d). Copies of these analyses are provided in attachments 8a through 8d of the LAR. The calculations in these analyses are consistent with Cameron's Topical Report ER-80P (Reference I-1), as supplemented by ER-157P (Reference I-2), ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation" (Reference I-8) and Exelon standard NES-EIC-20.04 (Rev. 5). This approach has been approved by the NRC in References I-3 and I-4. The core thermal power uncertainty calculation which takes into account the uncertainty associated with the feedwater flow venturis is performed in accordance with Exelon standard NES-EIC-20.04 (Rev. 5) and is consistent with ISA-RP67.04.02-2000 (Reference I-8).

The fundamental approach used is to statistically combine inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are

arithmetically combined to form statistically independent groups, which are then combined using the square root of the sum of the squares approach to determine the overall uncertainty.

I.1.D.4 NRC Criterion 4

For plants where the LEFM was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), provide additional justification for use. This justification should show either that the meter installation is independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated LEFM, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to NRC Criterion 4

Criterion 4 does not apply to Byron or Braidwood Stations, Units 1 or 2. Byron and Braidwood Stations LEFM CheckPlus systems were calibrated at Alden Research Laboratory. Cameron engineering reports for each of the Units evaluating the calibration test data from Alden Research Laboratory have been completed and are provided in LAR Attachments 8a through 8d (Appendix A.3). The calibration factors used for each Units LEFMs are based on the analysis contained in these reports.

I.1.E Total Power Measurement Uncertainty

Feedwater flow and temperature are the main inputs for determining the plant secondary calorimetric power, which is used in turn to determine the reactor thermal power. The feedwater mass flow rate and temperature are transmitted from the LEFM electronics cabinet to the Unit's plant process computer (PPC) for use in the calorimetric software application (secondary plant heat balance) which determines reactor thermal power. This improved measurement accuracy for feedwater mass flow and temperature over that currently obtainable with venturi-based instrumentation and thermocouples reduces total uncertainty in the calculation of RTP to be far less than the nominal 2% currently assumed in many accident analyses thereby allowing an increase in reactor thermal power equivalent to the decrease in uncertainty.

The uncertainty calculations for Byron and Braidwood Stations, Units 1 and 2, are documented and provided in LAR Attachments 8a through 8d. In addition to the uncertainties associated with the parameters provided by the LEFM CheckPlus system, the uncertainties associated with the other plant parameters used by the plant computer to calculate the calorimetric are combined and taken into consideration. For consistency in applying the total power uncertainty to all four Units, Exelon has conservatively used the highest thermal power uncertainty value of ± 0.345%, based on Braidwood Unit 1, when calculating the proposed uprate power level and applying operational limitations.

Multiplication factors based on the ratio of LEFM CheckPlus feedwater flow to venturi feedwater flow will be generated and used by calorimetric application, executing on the PPC, to adjust the venturi based calorimetric value in the event of an LEFM failure. The multiplication factors minimize the deviation between the calculated thermal power based on LEFM CheckPlus system measurements and the

calculated thermal power based on the venturi flow measurements. During normal LEFM CheckPlus system operation, at five second intervals, the calorimetric application calculates a new set of multipliers (LEFM flow/venturi flow) for each feedwater line. At one minute intervals, the multipliers are then worked into a set of running average multipliers. In the event of an LEFM CheckPlus system failure, the running average multipliers become fixed and are applied within the calorimetric application to the final calorimetric value based upon the venturi flows.

As long as all LEFM CheckPlus instruments remain operable, reactor power will be calculated utilizing the LEFM feedwater flow. Upon any LEFM CheckPlus system failure, reactor power will be calculated utilizing the venturi based feedwater flow corrected by the running average multipliers (correction to LEFM CheckPlus feedwater flow). If at the end of 72 hours (the proposed Allowed Outage Time in the TRM) the LEFM CheckPlus system is not returned to operable, reactor power will be calculated based on venturi feedwater flow uncertainties assuming a 2% uncertainty and the affected Units power will be reduced to pre-MUR reactor power limitations (3586.6 MWt). All actions are monitored automatically and controlled by the PPC calorimetric application. The existing uncorrected venturi-based feedwater flow will continue to be maintained and used for feedwater control and other functions.

If the PPC becomes unavailable a controlled hand calorimetric procedure is available for manually calculating the reactor power as required. The LEFM CheckPlus based feedwater values can be obtained locally from the LEFM CheckPlus system panel and, if operating, may be used to calculate reactor thermal power via the hand calorimetric procedure. In the event that both the PPC and the LEFM CheckPlus system are inoperable, the hand calorimetric procedure contains the necessary directions to ensure a venturi based calculation at pre-MUR reactor power limitations (3586.6 MWt). In addition, multiple other parameters (Nuclear Instrumentation System (NIS) Power monitors, differences between the RCS loop hot leg and cold leg temperatures, steam flow, feed flow, turbine first stage pressure, main generator output) provide indication of reactor power level.

Transducer Replacement

Uncertainty associated with transducer replacement was addressed by Cameron in References I-2, I-9 and I-10. Cameron performed numerous tests with various potential placement of the transducer element in the housing. Cameron determined that test results were bounded by predicted behavior and that the analyses predicted a larger uncertainty than was obtained during testing. The system uncertainties incorporate an additional transducer variability uncertainty in both the profile factor uncertainty and in the installation uncertainty.

Transducer replacement uncertainty has been included in the Byron and Braidwood LEFM CheckPlus system uncertainty calculations. Therefore, this issue is adequately addressed for the LEFM CheckPlus installations for Byron and Braidwood Stations, Units 1 and 2.

I.1.F Calibration and Maintenance Procedures of Instruments Affecting the Power Calorimetric

I.1.F.i Maintaining Calibration

Calibration and maintenance for the LEFM CheckPlus hardware and instrumentation will be performed using procedures based on the appropriate Cameron LEFM CheckPlus technical manuals, which ensures that the LEFM CheckPlus system remains bounded by the Topical Report ER-80P analysis and assumptions. Routine preventive maintenance activities for the LEFM will be as discussed in Section I.1.D.1. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated using approved procedures. Preventive maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. Work will be planned and executed in accordance with established Byron and Braidwood Station work control processes and procedures.

I.1.F.ii Controlling Software and Hardware Configuration

Cameron's verification and validation (V&V) program fulfills the requirements of ANSI/IEEE-ANS Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E (Reference I-11), and ASME NQA-2-1999, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference I-12). In addition, the program is consistent with guidance for software V&V in EPRI TR-103291s, "Handbook for Verification and Validation of Digital Systems," December 1994 (Reference I-17). Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM CheckPlus system are provided in Reference I-1.

After installation, the LEFM CheckPlus system software configuration will be maintained using existing procedures and processes. The plant computer software configuration is maintained in accordance with the Exelon Nuclear change control process, which includes verification and validation of changes to software configuration. Configuration of the hardware associated with the LEFM CheckPlus system and the calorimetric process instrumentation will be maintained in accordance with Exelon Nuclear configuration control processes.

I.1.F.iii Performing Corrective Actions

Plant instrumentation that affects the power calorimetric, including the LEFM inputs, will be monitored by Byron and Braidwood Stations personnel. In accordance with the Station's corrective action programs, deficiencies will be documented and necessary corrective actions will be identified and implemented.

I.1.F.iv Reporting Deficiencies to the Manufacturer

Deficiencies associated with the vendor's processes or equipment will be reported to the vendor as needed to support corrective action.

I.1.F.v Receiving and Addressing Manufacturer Deficiency Reports

Cameron has procedures to notify users of important LEFM deficiencies. Byron and Braidwood also have existing processes for addressing manufacturer's deficiency reports. Applicable deficiencies will be documented and addressed in the Byron and Braidwood corrective action program.

I.1.G Completion Time and Technical Basis

Byron and Braidwood Stations propose to continue to operate the Unit at the MUR uprated power for up to 72 hours subsequent to an LEFM system becoming inoperable. In accordance with the proposed TRM, if the LEFM system is declared inoperable (i.e., "Alert" or "Fail" condition), the Technical Limiting Condition for Operation (TLCO) will require that either the LEFM system be restored to operable within 72 hours or power is to be reduced to ≤ 3586.6 MWt (CLTP or 98.3% of MUR LTP). (Refer to proposed TRM in Attachment 1 to the LAR.)

The electronics cabinet performs continuous monitoring of LEFM CheckPlus system parameters to identify any problems with the instrumentation. The LEFM self-verification feature provides a comprehensive check of electronics, timing, signal-to-noise ratio, signal amplitude, noise levels, and average non-fluid delay. These features are described in detail in the LEFM topical reports.

An LEFM CheckPlus system "Alert" alarm indicates a loss of redundancy and the calculated power level error associated with the LEFM CheckPlus system flow measuring system in this condition is increased. An "Alert" alarm is caused by:

- Loss of a single process input,
- Loss of a single flow plane (loss of one or more flow transducers in a flow plane) on one or more feedwater lines,
- Loss of a single redundant spool piece resistance temperature detector (RTD) on either line,
- Loss of a single redundant feedwater header pressure input,
- Loss of a single electronics unit redundant component,
- Process input or output is calculated outside a pre-determined allowable range by one processing unit, or
- Internal self-check indicates system parameters that exceed pre-established limits and affect a single plane.

An LEFM CheckPlus system "Fail" alarm indicates a loss of function. A "Fail" alarm is caused by:

- Loss of both redundant process inputs,
- Loss of both flow planes any feedwater lines,
- Loss of both redundant spool piece RTDs on a single loop,
- Loss of both feedwater header pressure inputs,
- Failure of both redundant components in the electronics unit,

- A process input or output is calculated outside a pre-determined allowable range by both processing,
- Loss of the data link between the LEFM system and the process computer, or
- Internal self-check indicates system parameters that exceed pre-established limits and affect multiple planes in any loop.

In the event the LEFM CheckPlus system status changes to either “Alert” or "Fail" the Operations personnel are alerted through an annunciator in the main control room. The plant process computer will also provide a computer alarm message to the Control Room if the status of the LEFM instrumentation changes.

The basis for the proposed 72 hour Allowed Outage Time (AOT) is as follows:

- A completion time of 72 hours provides plant personnel sufficient time to plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.
- During the Allowed Outage Time (AOT), when the LEFM system is inoperable, the “normalized” feedwater flow from the venturis will be used for the calorimetric until the LEFM is returned to operable as discussed in I.1.E above. To ensure that the venturi based calorimetric is consistent with the LEFM CheckPlus system based calorimetric, the venturi-based flow rate is corrected to the most recent LEFM measurements as described in Section I.1.E.
- Regarding potential drift in the measurement of feedwater differential pressure across the feedwater flow venturis:
  - There has been no evidence of feedwater flow venturi fouling at Byron or Braidwood Stations. This is based on a review of historical work orders that document this observation as part of a procedural requirement performed every refueling outage. In addition, historical data was gathered over the last several years where feedwater venturi flow was analyzed. This data indicated that there was no divergence in feedwater flow indication that would suggest venturi fouling. Therefore, any fouling or consequently sudden de-fouling is extremely unlikely, especially within a 72 hour period.
  - If fouling of the venturis were to occur, the fouling would result in a higher than actual indication of feedwater flow. This condition results in an overestimate of the calculated calorimetric power level, which is conservative, as the reactors will actually be operating below the calculated power level.
  - Table A-1, in Reference I-1, indicates a typical power measurement uncertainty calculation for a two-loop PWR to be approximately 1.4%. The systematic error associated with feed flow nozzle differential pressure in this calculation is shown to be approximately 1.0%. Assuming this was calculated based on an 18-month cycle; this would represent a maximum potential drift in the differential pressure measurement of less than 0.002% per day. Over a 72-hour period, this would have an insignificant effect on the feedwater flow measurement. Feedwater flow differential pressure instrument drift history at Byron and Braidwood is consistent with the assumptions of this typical calculation.

- As described in Cameron report ER-157P (Reference I-2), the LEFM CheckPlus consists of two redundant planes of transducers and a single path or single plane malfunction results in a minimal increase in feedwater flow uncertainty. For Byron and Braidwood Stations, operators will conservatively respond to a single path or single plane failure in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode.
- Operators routinely monitor other indications of core thermal power, including Nuclear Instrumentation (NIS) Power Range Monitors, Loop Δ-Temperatures, steam flow, feed flow, turbine first stage pressure, and main generator output.

A control room annunciator response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine the LEFM CheckPlus system status and cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop specific procedures for operator and maintenance response actions.

The limitations discussed above regarding operation with an inoperable LEFM CheckPlus system will be included in the Technical Requirements Manual (TRM) and associated implementation procedures, which will be revised prior to implementation. Attachment 3 to this License Amendment Request provides the proposed TRM revision.

As described in Section I.1.E, the Braidwood and Byron calorimetric application on the Plant Process Computer will execute three simultaneous calculations of reactor power. As long as the LEFM system is operable, reactor power will be calculated utilizing the LEFM flow. If the LEFM system becomes inoperable reactor power will be calculated utilizing the venturi feedwater flow normalized to the LEFM feedwater flow. If at the end of the Completion Time, the LEFM system is not operable, reactor power will be calculated based on venturi feedwater flow assuming a 2% uncertainty and reactor power will be reduced to pre-MUR reactor power limitations.

If reactor power is below pre-MUR power during the time the LEFM system is inoperable, current TRM rules of usage (stated in TRM LCO 3.0.d) will not allow reactor power to be raised above pre-MUR power limitations during the Completion Time.

The NRC has previously approved MUR power uprate applications with Completion Times of 72 hours (References I-13 through I-16).

I.1.H Actions for Exceeding Completion Time and Technical Basis

As described above, these actions are covered in the proposed TRM which is provided in Attachment 3a and 3b to this License Amendment Request document. The LEFM TLCO requires that if an LEFM system is declared as inoperable and is not restored to operable status within 72 hours, then power is to be reduced to ≤ 3586.6 MWt (CLTP or 98.3% of MUR LTP).

I.1.I Refer ences

I-1 Cameron Engineering Report ER-80P, Revision 0, *Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level using the LEFM Check System,* Caldon Inc., March 1997.

I-2 Topical Report (TR) Engineering Report ER-157P, Revision 8, "Caldon Ultrasonics Engineering Report ER-157P, 'Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System'," dated May 11, 2009 (ML091340322)

I-3 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry, TU Electric, Comanche Peak Steam Electric Station, Units 1 and 2 – *Review of Caldon Engineering Topical Report ER 80P, Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System* (9903190065, ADAMS legacy library), March 8, 1999.

I-4 Letter from Thomas B. Blount Deputy Director, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation to Mr. Ernest Hauser, Director of Sales, Cameron; Final Safety Evaluation by the Office of Nuclear Reactor Regulation Engineering Report ER-157P Topical Report, Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System," Cameron Measurement Systems Project NO. 1370; dated August 16, 2010 (ML071500358 and ML071500360).

I-5a Cameron Caldon® Ultrasonics Engineering Report 829, Rev 1, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for Byron Unit 1,* July 2010.

I-5b Cameron Caldon® Ultrasonics Engineering Report 832, Rev 1, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for Byron Unit 2,* July 2010.

I-5c Cameron Caldon® Ultrasonics Engineering Report 843, Rev 0, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for Braidwood Unit 1,* July 2010.

I-5d Cameron Caldon® Ultrasonics Engineering Report 844, Rev 0, *LEFM CheckPlus Meter Factor Calculation and Accuracy Assessment for Byron Unit 2,* July 2010.

I-6 NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications, dated January 31, 2002 (ML013530183)

I-7a Cameron Caldon® Ultrasonics Engineering Report ER-800, Revision 1, *Bounding Uncertainty Analysis for Thermal Power Determination at Byron Unit 1 Nuclear Generating Station Using the LEFM CheckPlus System,* October 2010.

I-7b Cameron Caldon® Ultrasonics Engineering Report ER-801, Revision 1, *Bounding Uncertainty Analysis for Thermal Power Determination at Byron Unit 2 Nuclear Generating Station Using the LEFM CheckPlus System,* October 2010.

I-7c Cameron Caldon® Ultrasonics Engineering Report ER-802, Revision 1, *Bounding Uncertainty Analysis for Thermal Power Determination at Braidwood Unit 1 Nuclear Generating Station Using the LEFM CheckPlus System,* October 2010.

I-7d Cameron Caldon® Ultrasonics Engineering Report ER-803, Revision 1, *Bounding Uncertainty Analysis for Thermal Power Determination at Braidwood Unit 2 Nuclear Generating Station Using the LEFM CheckPlus System,* October 2010.

I-8 ISA-RP67.04.02-2000, *Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation.*

I-9 *Cameron Customer Information Bulletin CIB-125*, Revision 0, April 23, 2007

I-10 Letter from Hauser, Ernie (Cameron Measurement Systems) to U.S. Nuclear Regulatory Commission, "Caldon Ultrasonics Engineering Report ER-157, 'Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,' Revision 8," May 11, 2009 (ML091340322).

I-11 ANSI/IEEE-ANS Standard 7-4.3.2, 1993, *IEEE Standard Criteria for Digital computers in Safety System of Nuclear Power Generating Station*.

I-12 ASME NQA-2-1999, *Quality Assurance Requirement for Nuclear Facility Application*

I-13 NRC letter to Exelon Nuclear, LaSalle County Station, Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC NOS. ME3288 and ME3289), September 16, 2010 (ML 101830361)

I-14 NRC letter to Nebraska Public Power District; Cooper Nuclear Station – Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC NO. MD7385), June 30, 2008 (ML081540280)

I-15 NRC Letter to Southern Nuclear Operating Company, Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Issuance of Amendments Regarding Appendix K Measurement Uncertainty Recovery, September 23, 2003 (ML032590944)

I-16 NRC Letter to Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Re: Amendment Measurement Uncertainty Recapture Power Uprate, July 22, 2008 (ML091820366)

I-17 EPRI TR-103291s, "Handbook for Verification and Validation of Digital Systems," December 1994

II ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

A. Identify the transient or accident that is the subject of the analysis

B. Confirm and explicitly state that

i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant

ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC

iii. the analyses of record are not changed by the requested power uprate

C. Confirm that bounding event determinations continue to be valid

D. Provide a reference to the NRC's previous approvals discussed in Item B. above

RESPONSE TO II ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

II Accidents and Transients Bounded by the Analyses of Record for the Measurement Uncertainty Recapture

II.1 Intr oduction

A review of UFSAR Chapter 15 was performed to support the Byron and Braidwood Measurement Uncertainty Recapture (MUR) power uprate with respect to the accident analyses. The UFSAR review was conducted to confirm that the existing analyses of record (AOR), as currently presented in the UFSAR, were either performed conservatively and remain valid and bounding for the proposed power uprate or were explicitly reanalyzed.

The analyses generally address the core and/or NSSS thermal power in one of three ways and were correspondingly evaluated for MUR uprate conditions as follows:

(1) Analyses that apply a 2.0% increase to the initial power level to account for the power measurement uncertainty.

These analyses would normally not have to be re-performed to address MUR uprate conditions because the sum of the proposed core power level increase and the decreased power measurement uncertainty falls within the previously analyzed conditions. The existing 2.0% uncertainty is reallocated so a portion is applied to uprate power and the remainder is retained to accommodate the power measurement uncertainty.

During the evaluation process for the MUR several legacy issues associated with the AOR for the LOCA and Main Steam Line Break (MSLB) mass and energy (M&E) analyses were identified and are being tracked for completion in the Station's corrective action programs. These two analyses were re-performed to address these legacy issues and take into consideration applicable adjustments to the inputs based on MUR power uprate conditions. Additionally, as discussed in Attachment 1, the Steam Generator Tube Rupture Analysis was also re-performed. For the purposes of this submittal, these analyses have been included in Section III for accidents/transients that are not considered to be bounded; an additional level of detail has been included to summarize the salient information from the re-analysis.

(2) Analyses that are performed at 0% power conditions.

These analyses would normally not have to be re-performed to address MUR uprate conditions because they are not dependent on power; however, as discussed in Sections III.1.A.5.5 and III.1.A.5.9, the hot zero power steam line break and the uncontrolled rod withdrawal from subcritical, respectively, were reanalyzed using the VIPRE subchannel analysis code. As discussed in Section II.2.9, the rod ejection at hot zero power was not reanalyzed. For the purposes of this submittal, these analyses have been included in Section III for accidents/transients that are not considered to be bounded; an additional level of detail has been included to summarize the salient information from the analysis.

(3) Analyses that employ a nominal power level.

These analyses have been re-performed for the proposed MUR power level. As discussed in Section III.1.A.2, many of these analyses have been re-performed utilizing the VIPRE subchannel analysis code. The adoption of the VIPRE code accounts for the majority of the accidents/transient analyses that have been included in Section III. For those analyses, as discussed in Section III.1.A.2, the "nominal" core power for the transient response was conservatively assumed to be 102% (3672 MWt NSSS) and the associated DNBR analysis was performed assuming 101.7% (3662 MWt) consistent with Revised Thermal Design Procedure (RTDP).

Table II-1 below identifies the accident/transient analyses that were evaluated as part of MUR power uprate and indicates if the current AOR remains bounded. For the reasons discussed above, there are a significant number of accidents that have been re-analyzed for the MUR power uprate and as such have been included in Section III, "Accidents and Transients for which the Existing Analyses of Record Do Not Bound Plant Operations at the Proposed Uprated Power Level." Table II-1 also provides direction as

to where in this document discussion of the accident/transient analysis may be found. The table also provides summary information pertaining to NRC approval of the current AOR, the use of NRC approved methodologies, or if NRC approval is being requested for the methodologies being employed for the re-analysis.

Table II-1 UFSAR Accidents, Transients and Other Analyses

Accident/Transient	UFSAR Section	Current AOR Power Level(1)	MUR Analysis Power Level	RIS Section	NRC Approval
Internal Flooding	3.6.1	100%	102%(2)	II.2.17	UFSAR Attachment D3.6
Main Steam Line Break Mass and Energy Releases Outside Containment	3.6.1	102%	Bounded	II.2.15	License Amendment 119/113 (Reference II.2.12-1)
Safe Shutdown Fire Analysis	9.5.1	100%	102%(2)	II.2.18	Byron/Braidwood Stations Fire Protection Report (Amendment 24, December 2010)
Natural Circulation Cooldown	5.4.7.2.7	102%	Bounded	II.2.16	License Amendment 119/113 (Reference II.2.16-2).
Short-term LOCA Mass and Energy Releases	6.2.1	N/A(3)	Bounded	II.2.14	License Amendment 119/113 (Reference II.2.14-1).
Long-term LOCA Mass and Energy Releases	6.2.1.3.1	102%	Reanalyzed	III.15	NRC approved methodologies.(References III.15-3 through 8)
Main Steam Line Break Mass and Energy Releases Inside Containment	6.2.1.4	102%	Reanalyzed	III.16	NRC approved methodology (Reference III.16-1) and License Amendment 119/113 (Reference III.16-4)
Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature	15.1.1	100%	"Nominal"(5)(6)	III.2	NRC approved methodology (References III.2-1, 3 and 4) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval
Feedwater System Malfunctions Causing an Increase in Feedwater Flow	15.1.2	100%	"Nominal" [5][6]	III.2	NRC approved methodology (References III.2-1, 3 and 4) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Excessive Increase in Secondary Steam Flow	15.1.3	100%	"Nominal" [5]	III.3	NRC approved methodologies (References III.3-1 and 2).
Inadvertent Opening of a Steam Generator Relief or Safety Valve	15.1.4	N/A	N/A	II.2.1	This event is bounded by the hypothetical steamline break discussed in UFSAR Sections 15.1.5 and 15.1.6.
Steam System Piping Failure at Zero Power	15.1.5	0%	0%[7]	III.4	NRC approved methodology (References III.4-1 and III.4-3) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Radiological Consequences of a Postulated Steamline Break Using AST	15.1.5.3 15.3.3.4	100%	3672 MWt (NSSS)	III.17	NRC approved methodologies (References III.15-1) License Amendment 119/113 (Reference III.15-2)

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval
Steam System Piping Failure at Full Power	15.1.6	100%	"Nominal"[5] [6]	III.5	NRC approved methodology (References III.5-1, 3 and 4) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow	15.2.1	N/A	N/A	N/A	There are no pressure regulators whose failure or malfunction could cause a steam flow transient.
Loss of External Load/Turbine Trip/Inadvertent Closure of Main Steam Isolation Valves/Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	15.2.2 - 15.2.5	RCS Overpressure 102%	Bounded	III.6	License Amendment 138/131 (Reference III.11-4)
		Transient 100%	"Nominal"[5]		NRC approved methodologies (References III.6-1 and 2) and License Amendment 138/131 (Reference III.11-4)
		MSS Overpressure No existing AOR	"Nominal" [5]		NRC approved methodologies (References III.6-1 and 2) and License Amendment 138/131 (Reference III.11-4)
Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)	15.2.6	102%	Bounded	II.2.2	License Amendment 138/131 (Reference III.11-4)
Loss of Normal Feedwater Flow	15.2.7	102%	Bounded	II.2.3	License Amendment 138/131 (Reference III.11-4)
Feedwater System Pipe Break	15.2.8	102%	Bounded	II.2.4	License Amendment 119/113 (Reference II.2.12-1)

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval
Partial Loss of Forced Reactor Coolant Flow	15.3.1	100%	"Nominal"[5][6]	III.7	NRC approved methodology (References III.7-1 and 3) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Complete Loss of Forced Reactor Coolant Flow	15.3.2	100%	"Nominal"[5][6]	III.8	NRC approved methodology (References III.8-1 and 3) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Reactor Coolant Pump Shaft Break/Locked Rotor with Loss of Offsite Power	15.3.3 - 15.3.5	PCT / RCS Over Pressure: 102%	Bounded	III.9	License Amendment 119/113 (Reference II.2.12-1)
		100%	"Nominal"[5][6]		NRC approved methodology (References III.9-1 and 2) used for the transient analyses. License Amendment 119/113 (Reference II.2.12-1) As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	15.4.1	0%	0%[7]	Transient II.2.5	License Amendment 119/113 (Reference II.2.12-1).
				DNB III.1.A.5.9	As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	15.4.2	Overpressure 8% (Limiting Case)	Bounded	III.10	NRC approved methodologies License Amendment 119/113 (Reference II.2.12-1)
		DNB 100% 60% 10%	100%, 60%, 10% of "Nominal"[5]		
Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)	15.4.3	100%	100%[7]	Transient II.2.6	License Amendment 119/113 (Reference II.2.12-1)
				DNB III.1.A.5.8	As described in Section III.1.A, this LAR requests approval for use of the NRC approved VIPRE computer code to perform DNB analysis.
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	15.4.4	N/A	N/A	II.2.7	This event is precluded by the Technical Specifications and thus, has previously been deleted from the UFSAR.
Malfunction or Failure of the Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	15.4.5	N/A	N/A	N/A	N/A
Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant	15.4.6	Not Power Dependent	Bounded	II.2.8	License Amendment 119/113 (Reference II.2.12-1)

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	15.4.7	Cycle Specific	Cycle Specific	III.1.C	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Reference III.1-12)
Spectrum of Rod Cluster Control Assembly Ejection Accidents	15.4.8	HFP – 102%	Bounded	II.2.9	License Amendment 119/113 (Reference II.2.12-1)
		HZP – 0%			
Inadvertent Operation of Emergency Core Cooling System During Power Operation	15.5.1	Peak Pressurizer Volume 102%	Bounded	III.11	License Amendment 138/131 (Reference III.11-4)
		100%	"Nominal"[5]		NRC approved methodology (References III.11-1 and 2) used for the transient analyses. License Amendment 138/131 (Reference III.11-4)
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	15.5.2	N/A	N/A	II.2.10	This event is bounded by the CVCS Malfunction that Decreases Boron Concentration event discussed in Section II.2.8 and the inadvertent operation of the emergency core cooling system at power event discussed in Section III.1.
Number of BWR Transients	15.5.3	N/A	N/A	N/A	N/A
Inadvertent Opening of a Pressurizer Safety or Relief Valve	15.6.1	100%	"Nominal"[5]	III.12	NRC approved methodologies (References III.12-1 and 2)
Failure of Small Lines Carrying Primary Coolant Outside Containment	15.6.2	Not Power Dependent	Bounded	II.4.2	License Amendment 119/113 (Reference II.2.12-1)
Steam Generator Tube Rupture	15.6.3	102%	Reanalyzed	III.13	Revised analysis provided with this submittal (Appendix 5a).
Spectrum of BWR Steam System Piping Failures Outside of Containment	15.6.4	N/A	N/A	N/A	N/A

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval
Loss of Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (Best Estimate LOCA)	15.6.5	102%	Bounded	II.2.11	License Amendment 170/164 (Reference II.2.11-3)
Small Break LOCA Analysis	15.6.5.2.2	102%	Bounded	II.2.12	License Amendment 119/113 (Reference II.2.12-1)
Post-LOCA Long-Term Core Cooling/Subcriticality	15.6.5.2.4	102%	Bounded	II.2.13	License Amendment 119/113 (Reference II.2.13-1) and Reference II.2.13-2
BWR Transient	15.6.6	N/A	N/A	N/A	N/A
Gas Waste System Leak or Failure	15.7.1	102%	Bounded	II.4.4	License Amendment 119/113 (Reference II.2.12-1)
Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)	15.7.2	Not Power Dependent	Bounded	II.4.5	License Amendment 119/113 (Reference II.2.12-1)
Postulated Radioactive Release Due to Liquid Tank Failure (Ground Release)	15.7.3	Not Power Dependent	Bounded	II.4.6	License Amendment 119/113 (Reference II.2.12-1)
Fuel Handling Accident	15.7.4	102%	Bounded	II.4.7	License Amendment 119/113 (Reference II.2.12-1)
Spent Fuel Cask Drop Accident	15.7.5	N/A	N/A	N/A	This event is bounded by the Fuel Handling Accident
Anticipated Transients without Scram (ATWS)	15.8	100%	"Nominal"[5]	III.14	NRC approved methodology (References III.14-4)

Table II-1 UFSAR Accidents, Transients and Other Analyses *(cont.)*

Accident/Transient	UFSAR Section	Current AOR Power Level[1]	MUR Analysis Power Level	RIS Section	NRC Approval

Note:

1. Based on the current RTP of 3586.6 MWt.
2. These events were evaluated under MUR power uprate conditions and it was determined that they were not adversely affected by the MUR power uprate.
3. The short-term LOCA mass and energy releases are affected by changes in RCS temperature, which are a function of core power, operating strategy, main feedwater temperature, etc. Evaluations confirmed that the UFSAR analyses for short-term LOCA mass and energy releases used conservative RCS temperatures compared to the design RCS temperatures for the MUR power uprate.
4. Even though AOR for this accident/transient was performed at 102% of the current RTP, a reanalysis was performed to incorporate other conditions that resulted in the AOR being no longer bounded for reasons other than power level. The basis for the reanalysis is discussed in the appropriate section of the RIS as indicated.
5. For the purpose of this reanalysis a “Nominal” NSSS power of 3672 MWt was conservatively assumed for the transient analysis even though RTDP methodology was used.
6. For the purpose of this reanalysis a “Nominal” NSSS power of 3662 MWt was conservatively assumed for the DNB analysis even though RTDP methodology was used
7. The licensing basis analysis (LBA) statepoints for the DNBR analysis were evaluated with the increased nominal heat flux associated with the proposed uprate of 1.7% with 0.3% uncertainty rather than the conservative 2.0% uprate with 0% uncertainty.

II.2 Discussion of Events

UFSAR Chapter 15 accidents/transients and other UFSAR analyses were reviewed to support the Byron and Braidwood, Units 1 and 2 MUR power uprate. A brief discussion of each evaluation that has not been reanalyzed and which remains bounded is provided below.

II.2.1 Inadvertent Opening of a Steam Generator Relief or Safety Valve – UFSAR 15.1.4

The inadvertent opening of a steam generator relief or safety valve event (i.e., the credible steamline break) creates a depressurization of the secondary side with an effective opening size that is within the spectrum of break sizes analyzed by the hypothetical steamline break event. Therefore, the credible steamline break is bounded by the zero power and full power hypothetical steamline break events analyzed in Sections III.4 and III.5, respectively.

II.2.2 Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power) – UFSAR 15.2.6

The cases for the current Licensing Basis Analysis (LBA) for the loss of nonemergency AC are performed at both ends of the full power T_{avg} window including uncertainties, initial pressurizer pressures accounting for both the negative and positive pressure uncertainties, and a core power level of 102% of the nominal power (i.e., 3658 MWt). For the cases where seal injection to the reactor coolant pumps is lost the analysis is performed to ensure that the pressurizer does not become water solid. For the cases where RCP seal injection is maintained, the analysis ensures that although the pressurizer fills as a result of the event, the pressurizer safety valves will remain operable and that the Condition II event does not propagate into a more serious Condition III or IV event. In all cases, it is demonstrated that the acceptance criteria are met. Therefore, the current LBA remains bounding for the MUR power uprate, and the conclusions presented in the UFSAR remain valid.

II.2.3 Loss of Normal Feedwater Flow – UFSAR 15.2.7

Similar to the loss of non-emergency AC power event, the current LBA is performed to ensure that the pressurizer does not become water solid. The current LBA cases are performed at both ends of the full power T_{avg} window including uncertainties, initial pressurizer pressures accounting for both the negative and positive pressure uncertainties, and a core power level of 102% of the nominal power (i.e., 3658 MWt). In all cases, it is demonstrated that the acceptance criteria are met. Therefore, the current LBA remains bounding for the MUR power uprate and the conclusions presented in the UFSAR remain valid.

II.2.4 Feedwater System Pipe Break – UFSAR 15.2.8

The current LBA is performed to demonstrate that margin to the hot leg saturation temperature exists and, therefore, the core remains intact and in a coolable geometry. The current LBA is performed at 102% of the nominal core power (i.e., 3658 MWt) and demonstrates that the acceptance criterion has been met for all cases. Therefore, the current LBA remains bounding for the MUR power uprate and the conclusions presented in the UFSAR remain valid.

II.2.5 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition – UFSAR 15.4.1

This event is caused by an uncontrolled addition of reactivity to the reactor core initiated by the uncontrolled withdrawal of one or more Rod Cluster Control Assembly (RCCA) banks, resulting in a rapid power excursion. This transient is promptly terminated by a reactor trip on the power range neutron flux – low setpoint. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow. The current LBA is performed to demonstrate that the DNB design basis is met.

The initial conditions of the hot zero power (HZP) cases are not affected by the MUR because the event is analyzed at 0-percent power. Therefore, the results of the current HZP LBA cases remain valid. The LBA statepoints, which are based upon a fraction of nominal conditions, are unaffected by the increased power level since the time of reactor trip, which occurs on the power range neutron flux – low setpoint, is based on a fraction of nominal conditions (35%). Therefore, the time of trip is negligibly impacted.

To address the MUR power uprate conditions, the limiting normalized LBA statepoints were evaluated as discussed in Section III.1 A.5.9. The conclusions presented in the UFSAR remain valid.

II.2.6 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error) – UFSAR 15.4.3

The RCCA misoperation analysis includes the following events:

- One or more dropped RCCAs within the same group
- A dropped RCCA bank
- Statically misaligned RCCA
- Withdrawal of a single RCCA

The current LBA dropped RCCA transients (dropped RCCAs, dropped RCCA bank, and statistically misaligned RCCA) are evaluated to determine that the DNB design basis continues to be met. That is, the DNBR remains above the safety analysis limit value. The current LBA single RCCA withdrawal case was evaluated to verify that the number of fuel rods experiencing DNBR remains less than 5 percent of the total fuel rods in the core.

The current dropped rod LBA analysis is based on NRC approved methodology described in WCAP-11394 (Reference II.2.6-1), which involves the use of generic statepoints for the dropped rod event. Since the statepoints are presented as the relative transient response (i.e., change from initial) during the event, the statepoints are not sensitive to the initial conditions selected for the event, including the impacts of power uprates. Thus, the generic statepoints continue to be applicable to Byron and Braidwood at the MUR conditions. The increase in the nominal core heat flux is addressed with respect to the DNBR acceptance criteria. An evaluation of the DNB design basis as discussed in Section III.1.A.5.8, using the generic statepoints and the increased nominal core heat flux, confirmed that the acceptance criteria continue to be met. Therefore, the conclusions presented in the UFSAR remain valid.

References

II.2.6-1 WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.

II.2.7 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – UFSAR 15.4.4

The Technical Specifications require that all four reactor coolant pumps be operating for reactor power operation; therefore, operation with an inactive loop is precluded. This event was originally included in the UFSAR licensing basis when operation with a loop out of service was considered. Based on the Technical Specifications, which prohibit at-power operation with an inactive loop, and changes to the Technical Specifications that deleted all references to three-loop operation, this event has been previously deleted from the UFSAR.

II.2.8 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant – UFSAR 15.4.6

The current LBA boron dilution event is performed to demonstrate that the operator has at least 15 minutes in Modes 1 – 5 to terminate the RCS dilution before a complete loss of shutdown margin occurs. The critical parameters in determining the time available to terminate the dilution include the overall RCS active volume, the dilution flow rate, and the initial and critical boron concentrations. The analysis does not explicitly model or consider the initial power level. With respect to the Mode 1 analysis, the time of reactor trip, which alerts the operator that there is an event in progress, remains acceptable. The time is based on an equivalent reactivity insertion rate as determined by the uncontrolled RCCA bank withdrawal at power event, which was reanalyzed for the MUR power uprate (refer to Section III.10). The analysis confirmed that the time is acceptable. Therefore, the operator action time remains unaffected, the current LBA analysis remains bounding, and the conclusions presented in the UFSAR remain valid.

II.2.9 Spectrum of Rod Cluster Control Assembly Ejection Accidents – UFSAR 15.4.8

This event is caused by the mechanical failure of the control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft to the fully withdrawn position. The event models the power range neutron flux setpoints for primary protection. In the current LBA, the transient response for the hypothetical RCCA ejection event is analyzed with beginning-of-life (BOL) and end-of-life (EOL) reactivity feedback conditions for both the hot full power (HFP) and hot zero power (HZP) operating conditions in order to bound the entire fuel cycle and expected operating conditions. The analyses are performed to show that the fuel and cladding limits are not exceeded. The initial conditions of the HZP cases are not affected by the MUR power uprate because the event is analyzed at 0% power. Thus, the results of the current HZP LBA cases remain valid.

The full-power cases are performed at 102% of nominal core power (i.e., 3658 MWt), which bounds the MUR power uprate. Since the high neutron flux setpoint is a fraction of the nominal power level (118%), the increased nominal core power associated with the MUR power uprate would result in a peak core power that is less than 2% higher than predicted in the current analysis. However, this difference would have a negligible impact on the results because of the rapid increase in the nuclear power. Therefore, the

current HFP rod ejection analysis remains acceptable for the MUR power uprate. Therefore, the current LBA analysis remains bounding and the conclusions presented in the UFSAR remain valid.

With regards to pressure surge concerns and peak RCS pressures, the generic analysis for the pressure surge during the ejected rod event very conservatively assumed an ejected rod worth of one dollar at beginning of life and hot full power conditions (Reference II.2.9-1). The result of this generic analysis, which is applicable to the Byron and Braidwood units, indicates that the peak pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits. This conservative analysis bounds the MUR uprating condition for the program.

References

II.2.9-1 WCAP-7588-P-A, Rev. 1A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.

II.2.10 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory – UFSAR 15.5.2

This event is bounded by the evaluation of the boron dilution event in Section II.2.8 and the analysis of the inadvertent ECCS operation at power event in Section III.11. Therefore, the conclusions presented in the UFSAR remain valid.

II.2.11 Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (Best Estimate LOCA) – UFSAR 15.6.5

The Byron/Braidwood UFSAR Section 15.6.5 describes the large break LOCA ECCS analyses. The most recent large break LOCA ECCS analyses used the Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) (Reference II.2.11-1) for calculation of fuel peak clad temperature (PCT), local metal oxidation (LMO) and core-wide oxidation (CWO). The analyses have been reviewed and approved by the NRC (Reference II.2.11-2).

The realistic large break LOCA analyses were based on a core power of 102% of 3586.6 MWt (3658.3 MWt), specifically to bound full core power and uncertainty up to 102% of 3586.6 MWt. The inputs applied in the analyses were confirmed to remain applicable, bounding, or negligibly changed under MUR conditions. Therefore, the reported 10 CFR 50.46 results from the performed analyses, shown in Table II.2.11-1 and Table II.2.11-2 remain unchanged.

Table II.2.11-1 Byron/Braidwood Unit 1 Best Estimate Large Break LOCA Results
[Identical to Table 2-1 of Reference II.2.11-3]

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	1913	< 2200
95/95 LMO (%)	5.51	< 17
95/95 CWO (%)	0.25	< 1

Table II.2.11-2 Byron/Braidwood Unit 2 Best Estimate Large Break LOCA Results
[Identical to Table 2-2 of Reference II.2.11-3]

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT (°F)	2041	< 2200
95/95 LMO (%)	8.27	< 17
95/95 CWO (%)	0.33	< 1

References

II.2.11-1 WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.

II.2.11-2 Letter from N. J. DiFrancecso (USNRC) to M. J. Pacilio (Exelon), "Braidwood Station, Units 1and 2, and Byron Station, Units Nos. 1 and 2 – Issuance of Amendments RE: Large Break Loss of-Coolant Accident Analysis using the Automated Statistical Treatment of Uncertainty Method (TAC Nos. ME2941, ME 2942, ME 2943, and ME 2944), ML 103270403, December 21, 2010.

II.2.11-3 RS-09-178, "Braidwood, Units 1 & 2 and Byron, Units 1 & 2 – License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," December 16, 2009 (Accession Number ML093510099).

II.2.12 Small Break LOCA Analysis – UFSAR 15.6.5.2.2

UFSAR Section 15.6.5.2.2 describes the Byron and Braidwood Units 1 and 2 SBLOCA analyses performed for the 5% Uprate Program. Section 3.1.2 of Reference II.2.12-1 documents the NRC's approval of SBLOCA analyses for the 5% power uprate. The SBLOCA analyses have been supplemented by additional evaluations (described in UFSAR Sections 15.6.5.2.3.3.2 and 15.6.5.2.3.3.3) under the provisions of 10 CFR 50.46. The SBLOCA analyses assume a total core power of 3659 MWt, or 102% of 3587 MWt (i.e., 3586.6 MWt rounded up). Therefore, the analyzed core power is bounding for the MUR power uprate.

References

II.2.12-1 Letter from G. F. Dick (USNRC) to O. D. Kingsley (Exelon), "Issuance of Amendments: Increase in Reactor Power, Byron Station Units 1 and 2, Braidwood Station, Units 1 and 2." (TAC NOS. MA9428, MA9429, MA9426 and MA9427), ML011420274, May 4, 2001.

II.2.13 Post-LOCA and Long Term Cooling / Subcriticality – UFSAR 15.6.5.2.4

The Analyses of Record (AORs) for post-LOCA Long Term Cooling consists of three aspects: Subcriticality, Boric Acid Precipitation Control (i.e. Hot Leg Switchover (HLSO)), and Decay Heat Removal. The AORs were reviewed and evaluated as part of the Margin Uncertainty Recapture (MUR)

program. The most recent NRC approvals of the AORs are documented in Reference II.2.13-1 and Reference II.2.13-2. The following evaluations confirm that the AORs remain bounding for the proposed MUR power uprate, and that continued compliance with 10 CFR 50.46 paragraph (b) part (4) Coolable Geometry and part (5) Long Term Cooling is ensured.

- The minimum containment sump boron concentration is calculated to ensure post-LOCA subcriticality is maintained. The Subcriticality Limit is independent of power and is thus not impacted by the power uprate. The Subcriticality Limit is tracked in the Reload Safety and Analysis Checklist and is confirmed for each reload core design as part of the Westinghouse Reload Safety Evaluation (RSE) Methodology.
- The HLSO calculation uses an analyzed core power of 3658 MWt. The analyzed core power of 3658 MWt is derived from the licensed core power of 3586.6 MWt plus a calorimetric power uncertainty of 2%. This analysis remains bounding for the proposed MUR power uprate.

References

II.2.13-1 Letter from Dick, George Jr. (NRR – Project Manager) to Skolds, John L (Exelon Generation Company, LLC – President), "Hot Leg Switchover Confirmatory Analysis – Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2 (TAC NOS. MB5237, MB5238, MB5239, MB4240)," September 27, 2002. (ML022390175)

II.2.13-2 Letter from Dick, George Jr. (NRR – Project Manager) to Kingsley, O. D. (Exelon Generation Company, LLC – President), "Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC NOS. MA9428, MA9429, MA9426, and MA9427)," May 4, 2001. (ML011420274)

II.2.14 Short-Term LOCA Mass and Energy Release Analysis – UFSAR 6.2.1

The subcompartment analysis is performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) which accompanies a high energy line pipe rupture within the subcompartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the short-term LOCA blowdown mass and energy (M&E) release rates, the subcompartment volume, vent areas, and vent flow behavior. The short-term LOCA blowdown M&E release rates are affected by the initial RCS pressure and temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions associated with the MUR power uprate. Initial reactor power level is not a factor in the short-term LOCA M&E releases except for the effect of reactor power on initial RCS fluid temperatures.

The Byron and Braidwood MUR power uprate RCS initial pressure and temperatures were reviewed and confirmed to be bounded by the inputs to the existing short-term LOCA mass and energy releases (Reference II.2.14-1). Since the LOCA short-term M&E is bounded, the associated subcompartment pressurization analyses would be unaffected by the MUR power uprate.

The long term LOCA M&E and containment response analyses are not considered bounded and are discussed in Section III.15 of this attachment.

References

II.2.14-1 Letter from G. F. Dick (USNRC) to O. D. Kingsley (Exelon), "Issuance of Amendments: Increase in Reactor Power, Byron Station Units 1 and 2, Braidwood Station, Units 1 and 2." (TAC NOS. MA9428, MA9429, MA9426 and MA9427), May 4, 2001 [Accession No. ML011420274]

II.2.15 Main Steam Line Break Mass and Energy Releases Outside Containment – UFSAR 3.6.1

The main steam line break mass and energy releases used for compartment temperature response is described in Section 6.5.2 of Reference II.2.15-1. The analysis consists of 120 cases (60 per unit), addressing the effects of initial power level and break size for each Unit. The mass and energy releases and compartment response analyses assumed 102.0% of 3600.6 MWt NSSS (3586.6 MWt + 14 MWt pump heat), which bounds the MUR uprate. Using representative cases, an evaluation was done of some minor changes in other operating parameters. It was found that the peak temperatures are generally lower than those provided in Section 6.5.5 of Reference II.2.15-1. All results are below the criterion of 419°F before steamline isolation and the overall peak compartment temperature is less than the previously-reported value of 518.4°F. Reference II.2.15-1 conclusions remain valid for the steam line break event outside containment.

The main steam line break M&E and containment response analyses are not considered bounded and are discussed in Section III.16 of this attachment.

References

II.2.15-1 "Commonwealth Edison Company Byron and Braidwood Units 1 and 2 Power Uprate Project Transmittal of Licensing Report," May 18, 2000.

II.2.16 Natural Circulation Cooldown – UFSAR 5.4.7.2.7

The Byron and Braidwood Natural Circulation Cooldown analysis is documented in UFSAR Section 5.4.7.2.7 for cooldown time to RHR cut-in and cold shutdown conditions. The natural circulation cooldown analysis was performed using the NRC accepted TREAT methodology (References II.2.16-1 and 2). Because this event was analyzed at 3660 MWt (i.e., 102% of 3586.6 MWt rounded up to 3660 MWt) which is greater than the MUR reactor power of 3648 MWt and no major plant modifications that would restrict natural circulation flow have been performed, the Byron and Braidwood Natural Circulation Cooldown analysis is unaffected by the MUR power uprate.

References

II.2.16-1 NUREG-0871, Supplement 3, Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2," May 1987.

II.2.16-2 Letter from Dick, George Jr. (NRR – Project Manager) to Kingsley, O. D. (Exelon Generation Company, LLC – President), "Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 (TAC NOS. MA9428, MA9429, MA9426, and MA9427)," May 4, 2001. (ML011420274)

II.2.17 Internal Flooding – UFSAR 3.6 (Attachment D3.6)

The design bases for internal flooding outside the containment building were evaluated. The power uprate results in increased piping system flowrates (e.g., condensate, main feedwater and main steam). These changes were evaluated to determine any impact on the flooding analysis. Based on flooding analysis calculation reviews, it was determined that the current flood levels are not affected by the MUR power uprate.

II.2.18 Safe Shutdown Fire Analysis – UFSAR 9.5.1

The MUR power uprate does not change the design or function of any system or components that support safe shutdown (e.g., residual heat removal, chemical and volume control), nor does it impose any new requirements on these systems or components. Thus, sufficient equipment will continue to remain operational to achieve and maintain a safe shutdown condition in both units following a fire in any single plant fire zone.

The safe shutdown fire analysis is based, in part, on a natural circulation cooldown analysis and a single-train cooldown analysis. The natural circulation cooldown analysis remains bounding for the MUR power uprate, as noted in Section II.2.16. The single-train cooldown analysis, described in Section VI.1.C.iv, indicate that a single-train cooldown will take slightly longer under MUR power uprate conditions.

Because the power uprate slightly lengthens the time required for a cooldown, the time available to complete fire-related operator actions and maintenance activities, and still reach cold shutdown within the required time, may be reduced. Therefore, an evaluation of the time required to carry out the necessary repair activities was performed. The results of that evaluation identified the worst-case bounding fire zones (with respect to repairs) and provided a documented basis for concluding that all required repair activities could be completed within the time available under MUR power uprate conditions. The evaluation determined the maximum required time to complete cold shutdown repairs and compared that time to the time available to complete cold shutdown repairs. The time available to complete cold shutdown repairs was based on the results of the revised single-train cooldown analysis. A review of operator actions in response to a fire also confirmed that these actions are not adversely impacted by the MUR power uprate. These evaluations confirmed that the plant can, when necessary, continue to achieve cold shutdown within 72 hours, as required by regulations.

II.3 Design Transients

II.3.1 Nuclear Steam Supply System Design Transients

NSSS design transients were specified in the original design analyses of NSSS component cyclic behavior. The selected transients are conservative representations of transients that when used as a basis

for component fatigue analysis, provide confidence that the component is appropriate for its application over the 40-year plant license period. The Reactor Coolant System (RCS) and its auxiliary system components are designed to withstand the cyclic load effects from RCS temperature and pressure changes. The existing design transients were evaluated for their continued applicability at MUR power uprate conditions.

The key plant design parameters for the NSSS design transients are RCS hot and cold leg temperatures (T_{hot} and T_{cold}), steam generator secondary side steam temperature and pressure (T_{steam} and P_{steam}) and the feedwater temperature (T_{FW}), RCS thermal design flow (TDF) and the no-load temperature ($T_{no\text{-}load}$). The existing design transients for parameters except feedwater temperature bound plant operation at the uprated conditions. Those design transients with feedwater temperature variation required revision. This change was the result of the uprated full power feedwater temperature increase of 2.6°F and the introduction of the full power feedwater temperature window. The new feedwater temperature responses were developed so they would better represent uprated plant conditions. Design transients were assessed in appropriate areas.

The primary to secondary differential pressure limit was not exceeded for any normal or upset design transient. The frequencies of occurrences for the 40-year plant licensed period are unchanged for the power uprate. No new design transients are created as a result of the MUR Power Uprate Program.

II.3.2 Auxiliary Equipment Design Transients

The Byron and Braidwood Units 1 and 2 auxiliary equipment design specifications included transients that were used to design and analyze the Class 1 auxiliary nozzles connected to the RCS, and certain NSSS auxiliary systems piping, heat exchangers, pumps and tanks. The transients are sufficiently conservative, such that when used as a basis for component fatigue analysis, they provide confidence that the component will perform as intended over the current plant operating license period.

The only auxiliary equipment design transients potentially impacted by the MUR power uprate are those transients associated with full load NSSS design temperatures (T_{hot} and T_{cold}). These temperature transients are defined by the differences between RCS loop coolant temperature and the temperature of coolant in the auxiliary systems connected to the RCS loops. Since the operating coolant temperatures in the auxiliary systems are not impacted by the power uprate, the temperature difference between auxiliary systems and the RCS loops is only affected by changes in the RCS operating temperatures. The transients assume a full load NSSS T_{hot} and T_{cold} of 630°F and 560°F, respectively. These full load temperatures were selected for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters. A comparison of the approved range of T_{hot} (608.6 – 620.9°F) and T_{cold} (541.4- 555.1°F) for MUR power uprate at full load with the temperatures used to develop the current design transients indicates that the MUR-power uprate temperatures are lower. The lower MUR power uprate full load temperatures result in less severe design temperature transients. Therefore, the existing auxiliary equipment design transients are conservative and bounding for the MUR power uprate.

II.3.3 Plant Operability

The Byron/Braidwood Units 1 and 2 pressure control component sizing and plant operability transients were evaluated for the MUR power uprate program.

RCS pressure control component sizing includes the pressurizer PORV, spray, and heater capacities. These components must continue to successfully perform their intended functions. Plant operability for Condition I (normal condition) transients includes the plant response to 5%/minute loading and unloading, 10% step load increase and decrease, and large load rejection. These transients must not result in a reactor trip, ESFAS actuation, or challenge the pressurizer or main steam safety valves. Additionally, the 10% step load decrease must not lead to the actuation of the pressurizer PORVs. An evaluation was conducted to confirm the continued ability of the plant to meet these requirements at MUR power uprate conditions.

Pressure control component sizing and plant operability for the normal condition transients were each reviewed for continued applicability at MUR power uprate conditions. The reviews concluded that the MUR power uprate does not result in unacceptable plant operation for any of the transients reviewed. The existing pressure control components (pressurizer PORV, spray, and heater) meet the sizing criteria at the MUR power uprate conditions and the component capacities are adequate to mitigate the sizing basis transients without exceeding the limits. The evaluation for plant operability concludes that adequate margin will be maintained to relevant reactor trip and ESFAS setpoints and during the normal conditions transients at MUR power uprate conditions. The evaluation also concludes that the pressurizer PORVs will not be actuated for the 10% step load decrease.

The control systems remain stable and support operation at the MUR power uprate for normal condition transients. Therefore, the existing setpoints for the reactor control, pressurizer pressure control, pressurizer level control, steam generator level control, and steam dump control remain valid for MUR power uprate conditions.

II.4 Radiological Consequences

II.4.1 Spectrum of Rod Cluster Control Assembly Ejection Accidents Dose Evaluation – UFSAR 15.4.8

As discussed in UFSAR Section 15.4.8, the control rod ejection accident (CREA) analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The core inventory source term used in the current CREA analysis is a function of core power, enrichment, burn-up, gap fractions for non-LOCA events from Regulatory Guide 1.183, an assumed percent of failed fuel, an assumed percent of melted fuel, and an assumed radial peaking factor. The existing CREA dose evaluation was performed using the core inventory that assumes 3658.3 MWt, which is 102% of 3586.6 MWt. No changes to the assumed percent of failed fuel, assumed percent of melted fuel, or assumed radial peaking factor are required to support the MUR power uprate. The steam release modeled in the current CREA analysis is consistent with a core thermal power of 3658.3 MWt (102% of 3586.6 MWt). The release pathways and dose conversion factors are unchanged from the AST license amendment requests and associated safety evaluation reports (SERs). The current commitment to the NRC (RAI response letter RS-06-019) is to incorporate the updated χ/Q values when the analysis is revised; however, the dose analysis was not revised because this analysis is performed at a power level that bounds the MUR power uprate power level.

Therefore, the current CREA dose evaluation remains bounding for the MUR power uprate.

II.4.2 Failure of Small Lines Carrying Primary Coolant Outside Containment – UFSAR 15.6.2

The failure of small lines carrying primary coolant outside containment analysis was performed for radionuclide inventory based on a normalized primary coolant concentration limited to the Technical Specifications Dose Equivalent Iodine-131 limits, which removes the power dependence from the analysis. As discussed in UFSAR Section 15.6.2, the current analysis resulted in an exclusion area boundary whole body dose of 0.03 rem and thyroid dose of 1.0 rem for Byron and an exclusion area boundary whole body dose of 0.04 rem and thyroid dose of 1.4 rem for Braidwood. The doses are compared to 10% of the 10 CFR 100 criteria. The 10 CFR 100 acceptance criterion for failure of small lines carrying primary coolant outside containment exclusion area boundary whole body dose was 25 rem and exclusion area boundary thyroid dose was 300 rem. The radiological atmospheric dispersion factor (χ/Q) and dose conversion factors that were used in the analysis remain unchanged.

Therefore, the dose evaluation for the failure of small lines carrying primary coolant outside containment will not be impacted by the MUR power uprate.

II.4.3 LOCA Dose Evaluation – UFSAR 15.6.5

As discussed in UFSAR Section 15.6.5, the loss of coolant accident (LOCA) event analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The core inventory source term used in the current LOCA analysis is a function of core power, enrichment, and burn-up. The current LOCA dose analysis is based on a core inventory that assumes 3658.3 MWt, which is 102% of 3586.6 MWt. The LOCA radiological consequences result from the release of the core inventory to the RCS and then to the environment. The release pathways and dose conversion factors are unchanged from the AST license amendment requests and associated safety evaluation reports (SERs). The current commitment to the NRC (RAI response letter RS-06-019) is to incorporate the updated χ/Q values when the analysis is revised; however, the dose analysis was not revised because this analysis is performed at a power level that bounds the MUR power uprate power level.

Therefore, the existing LOCA radiological analysis remains bounding for the MUR power uprate.

II.4.4 Waste Gas Decay Tank Rupture – UFSAR 15.7.1

The Explosive Gas and Storage Tank Radioactivity Monitoring Program defined in TS 5.5.12 limits the quantity of radioactivity contained in a waste gas decay tank to less than an amount that would result in a whole body exposure of 0.5 rem to any individual in an unrestricted area in the event of an uncontrolled release of the tank's contents.

The current waste gas decay tank rupture analysis was performed with a reactor coolant inventory at 3658.3 MWt, which is 102% of 3586.6 MWt. The analysis resulted in an exclusion area boundary whole body dose of 0.54 rem for Byron and 0.73 rem for Braidwood, which is reported in UFSAR Table 15.0-12 and compared to the 10 CFR 100 acceptance criterion. The 10 CFR 100 acceptance criterion for waste gas decay tank rupture exclusion area boundary whole body dose was 25 rem.

Therefore, the MUR power uprate will have no impact on this accident.

II.4.5 Liquid Waste Tank Rupture - UFSAR 15.7.2

The current liquid waste tank rupture analysis was performed for radionuclide inventory based on a normalized primary coolant concentration limited to the Technical Specifications Dose Equivalent Iodine-131 limits, which removes the power dependence from the analysis for both the boron recycle holdup tank and spent resin tank. For the boron recycle holdup tank, the analysis resulted in an exclusion area boundary whole body dose of 0.44 rem and thyroid dose of 0.85 rem for Byron and an exclusion area boundary whole body dose of 0.6 rem and thyroid dose of 1.2 rem for Braidwood. For the spent resin storage tank, the analysis resulted in an exclusion area boundary whole body dose of 0.00016 rem and thyroid dose of 0.45 rem for Byron and an exclusion area boundary whole body dose of 0.00021 rem and thyroid dose of 0.61 rem for Braidwood. The doses are reported in UFSAR Table 15.0-12 and compared to the 10 CFR 100 acceptance criterion. The 10 CFR 100 acceptance criterion for liquid waste tank rupture exclusion area boundary whole body dose is 25 rem and exclusion area boundary thyroid dose is 300 rem. The radiological atmospheric dispersion factor (χ/Q) and dose conversion factors that were used in the analysis are unchanged.

Therefore, the current liquid waste tank rupture dose evaluation will not be impacted by the MUR power uprate.

II.4.6 Postulated Radioactive Release Due to Liquid Tank Failure (Ground Release) – UFSAR 15.7.3

The limiting postulated radioactive release due to postulated liquid tank failures is an unexpected and uncontrolled rupture of the boron recycle holdup tank in the auxiliary building. Upon failure, the only way any liquid effluents can be released to the environment is through a postulated crack in the auxiliary building, which would allow the contents of the tank to enter the groundwater.

For Byron Station, as discussed in UFSAR Section 2.4.13.3, for nuclides whose travel times are in excess of 0.5 years, the concentrations of all but three nuclides of the design basis liquid release, documented in UFSAR Table 2.4-20, “Inventory of Liquid Phase Isotopes in the Recycle Holdup Tank,” decay to values which are less than 10 CFR 20 limits. The three exceptions are Cs-134, Cs-137, and H-3. The reactor coolant activity in UFSAR Table 11.1-2, “Design Basis Reactor Coolant Fission and Corrosion Product Activity (Original Design),” (Original Licensed Thermal Power is 3411 MWt), bounds the uprated reactor coolant activity in UFSAR Table 11.1-13 “Uprated Design Basis Reactor Coolant Fission and Corrosion Product Activity,” for these three isotopes. Note that the values in Table 11.1-13 were calculated for 3658.3 MWt (i.e., 102% of the stretch power uprate power of 3586.6 MWt). Therefore, the three significant radionuclide concentrations in UFSAR Table 2.4-20 bound the projected values at the MUR power level (i.e., 3645 MWt). Conservatively assuming that only 10% of the saturated thickness of the aquifer between the plant building and the spring would contribute to the dilution of the effluents with the ambient groundwater, the available dilution factor and the large travel time of the effluents would reduce the concentrations of all the radionuclides including Cs-134, Cs-137, and H-3 to well below the 10 CFR 20 limits before their arrival at the spring.

For Braidwood Station, a cement bentonite slurry trench surrounds the perimeter of the main plant to restrict seepage out of the auxiliary building. The maximum elevation of the spilled fluid inside the cell is estimate to be 563 feet and the ambient groundwater elevation is 17 to 37 feet higher. Therefore there

would be no hydraulic gradient and the effluents will be contained and prevented from contaminating the surrounding groundwater.

Based on the above discussion, the MUR power uprate has no significant impact on this event due to the radioactive decay over the large travel time and the dilution with the three bounded radionuclide concentrations.

II.4.7 Fuel Handling Accident – UFSAR 15.7.4

The current fuel handling accident radiological analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The core inventory source term used in the current fuel handling accident analysis is a function of core power, enrichment, burn-up, and gap fractions for non-LOCA events from Regulatory Guide 1.183, the number of failed fuel rods, and the assumed radial peaking factor. The existing fuel handling accident dose evaluation was performed using a core inventory that assumes 3658.3 MWt, which is 102% of 3586.6 MWt, and a single failed fuel assembly (264 rods). No changes to the assumed number of failed fuel rods or assumed radial peaking factor are associated with the MUR power uprate. As part of the cycle reload safety evaluation process, the continued applicability of the gap fractions for non-LOCA events is verified per Regulatory Guide 1.183, Table 3, footnote 11. The release pathways and dose conversion factors are unchanged from the AST license amendment requests and associated SERs. The current commitment to the NRC (RAI response letter RS-06-019) is to incorporate the updated χ/Q values when the analysis is revised; however, the dose analysis was not revised because this analysis is performed at a power level that bounds the MUR power uprate power level.

Therefore, the current fuel handling accident dose evaluation remains bounding for the MUR power uprate.

II.5 Analyses to Determine Environmental Qualification Parameters

The parameters that are considered in the environmental qualification of safety-related equipment are temperature, pressure, humidity, caustic spray, submergence, and radiation.

<u>Non-Radiological Parameters</u>

All non-radiological environmental parameters for normal conditions (i.e., temperature, pressure, and relative humidity) remain bounding for the MUR power uprate.

The abnormal or accident values of relative humidity, caustic spray, and submergence conditions used in the current analyses remain bounding for the proposed uprate.

The post-accident pressure and temperature profile used in the current analyses of containment areas do not remain bounding for the proposed uprate. The evaluations of the post-accident containment pressure and temperature profiles are discussed in Sections III.15 and III.16.

The current limits on maximum temperatures and pressures used for the auxiliary building areas with environmentally qualified equipment remain bounding for the proposed uprate. Although the operating

conditions for turbine building high energy lines whose failure could affect certain auxiliary building areas are changing with MUR power uprate, an evaluation determined that the temperatures and pressures remain acceptable for MUR power uprate conditions.

The maximum temperatures and pressures determined in previous analyses of the main steam pipe tunnels and safety valve enclosures remain bounding. The mass and energy releases and compartment temperature response for these areas under MUR power uprate conditions are discussed in Section II.2.15 of this attachment.

Radiological Parameters

An evaluation of the normal radiation doses concluded that the conservatism in the current analyses was such that those analyses would remain bounding for the slight increase in normal radiation doses expected under the MUR power uprate conditions. Therefore, the normal dose contribution to the total integrated doses used for determining equipment qualification parameters remains bounding for the MUR power uprate.

An evaluation of the current radiological environmental parameters found that the post-accident dose contribution to the total integrated doses used for determining equipment qualification parameters had been analyzed with respect to a power level which bounds the MUR PU power level and found acceptable. Therefore, the total integrated doses used for determining equipment qualification parameters remain bounding for the MUR power uprate.

The environmental qualification of electrical equipment is discussed in Section V.1.C.

III ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

1. *This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling and flooding).*

2. *For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:*

   A. *Identify the transients/accidents that is the subject of the analysis*

   B. *Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate*

   C. *Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59*

   D. *Provide a reference to the NRC's approval of the plant's reload methodology*

3. *For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should include:*

   A. *Identify the transient or accident that is the subject of the analysis*

   B. *Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate*

   C. *Confirm that the limiting event determination is still valid for the transient or accident being analyzed*

   D. *Identify the methodologies used to perform the analyses, and describe any changes in those methodologies*

   E. *Provide references to staff approvals of the methodologies in Item D above*

   F. *Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology*

   G. *Describe the sequence of events and explicitly identify those that would change as a result of the power uprate*

H. Describe and justify the chosen single-failure assumption

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head, valve isolation capabilities) required to support the analysis

K. Discuss the results and acceptance criteria for the analysis, including any changes from previous analysis

RESPONSE TO III ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

III.1 Fuel

The fuel analyses and evaluations were performed to support the Byron Units 1 and 2 and Braidwood Units 1 and 2 Measurement Uncertainty Recapture Power Uprate (MUR-PU). The analyses assume a full core of VANTAGE+ fuel assemblies for the uprated Byron and Braidwood core designs. There is no fuel design change associated with the Byron and Braidwood MUR power uprate.

For completeness, Section III.1 includes the Thermal-Hydraulic analysis as well as the Fuel Structural, Nuclear Design and Fuel Rod evaluations.

III.1.A Core Thermal and Hydraulic Analysis

III.1.A.1 Input Parameters and Assumptions

The MUR power uprate DNB analyses assume a nominal core power level of 3648 MWt, which represents a 1.7% increase to the current nominal core power for Byron and Braidwood Units 1 and 2.

III.1.A.2 Method of Analysis

The thermal-hydraulic design methods for the MUR power uprate remain the same as currently in the Byron and Braidwood UFSAR except for two changes that were necessary to maintain acceptable DNBR margin:

- the NRC-approved W-3 alternative correlations in Reference III.1-1 (the ABB-NV and WLOP correlations) are used in place of the W-3 correlation (Reference III.1-3) as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable;
- the NRC-approved VIPRE-W (VIPRE) subchannel analysis code (Reference III.1-4) is used in place of the THINC-IV (THINC) subchannel analysis code (References III.1-5 and III.1-6) and the FACTRAN code (Reference III.1-7) for DNBR calculations.

The primary DNB correlation used in the analysis of the VANTAGE+ fuel at MUR power uprate conditions remains the WRB-2 DNB correlation (Reference III.1-8). The secondary DNB correlation, which supplements the primary DNB correlation for conditions where the primary DNB correlation is not applicable, is changed for the MUR power uprate. The W-3 correlation, which is the current secondary DNB correlation for the Byron and Braidwood Units, is inadequate to provide the DNBR margin necessary to support the MUR power uprate conditions. For the MUR power uprate DNB analyses, the NRC-approved W-3 alternative DNB correlations from Reference III.1-1 (the ABB-NV and WLOP correlations) are used as secondary DNB correlations.

The change to the VIPRE subchannel analysis code is necessary to implement the ABB-NV and WLOP DNB correlations from Reference III.1-1 for use in the MUR power uprate analyses as secondary DNB correlations. The NRC Safety Evaluation, Reference III.1-2, requires that the ABB-NV correlation for Westinghouse PWR application and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 code since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications. To support the use of the VIPRE code as the licensing basis subchannel analysis code for Byron and Braidwood Units 1 and 2, DNBR calculations have been performed with the VIPRE code for all of the DNB-limited UFSAR Chapter 15 events that are currently analyzed with the THINC subchannel analysis code. The DNBR calculations performed with the VIPRE code address the increased nominal heat flux and the change in power measurement uncertainty associated with the MUR power uprate. Consistent with the VIPRE modeling for PWR safety analyses established in Reference III.1-4, a 5% flow reduction to the hot assembly was assumed in all VIPRE DNBR calculations for the MUR power uprate.

The DNB analyses of the VANTAGE+ fuel in Byron and Braidwood Units 1 and 2 at MUR power uprate conditions continue to be based on the Revised Thermal Design Procedure (RTDP) (Reference III.1-9). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain the overall DNB uncertainty factors. For the MUR power uprate, the current plant operating parameter uncertainties remain applicable with the exception of the power measurement uncertainty. The Byron and Braidwood MUR power uprate is based on a reduced power measurement uncertainty associated with the use of the LEFM CheckPlus system to measure feedwater flow. Proprietary DNBR sensitivity factors, which are used to develop the DNB uncertainty factors, are calculated using the VIPRE code for ranges of conditions which bound the events for which RTDP methodology is applied. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined which meet the DNB acceptance criterion. In addition to the above considerations for uncertainties, DNBR margin is retained by performing the safety analyses to DNBR limits higher than the RTDP design limit DNBR values. Sufficient DNBR margin is conservatively maintained in the safety analysis DNBR limits as discussed in Section III.1.A.5.2 to offset the rod bow DNBR penalty and to provide flexibility in design and operation of the plant.

The Standard Thermal Design Procedure (STDP) methodology continues to be used for those DNB analyses where RTDP is not applicable. For the STDP, the initial condition uncertainties are accounted for deterministically by applying the uncertainties to the nominal conditions. The DNBR limit for STDP is the appropriate DNB correlation limit with consideration for applicable DNBR penalties.

III.1.A.3 Accep tance Criterion

The thermal-hydraulic design basis for the MUR power uprate remains the same as currently in the Byron and Braidwood UFSAR. The DNB design basis for the MUR power uprate DNB analysis is that there will be at least a 95-percent probability at 95-percent confidence level (95/95) that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events). Analytical assurance that the DNB criterion is met is provided by showing that the VIPRE-calculated DNBR is higher than the appropriate 95/95 DNBR limit for the DNB methodology and DNB correlation used in the analysis and that the VIPRE results are within the parameter ranges of the DNB correlation. The DNBR limits for the DNB correlations used with the VIPRE code are presented in Table III.1-1 for the MUR power uprate DNB analyses based on RTDP and in Table III.1-2 for the MUR power uprate DNB analyses based on STDP. In addition, the DNBR limits for the current DNB analyses with the THINC-IV code are also listed in the tables.

Table III.1-1 DNBR Limits for Events Analyzed with RTDP

DNB Correlation	Design Basis DNBR Limit for Current Analyses with THINC-IV (Typical Cell/Thimble Cell)	Design Basis DNBR Limit for MUR-PU Analyses with VIPRE (Typical Cell/Thimble Cell)
WRB-2	1.25/1.24	1.25/1.24
ABB-NV	Not applicable	1.19/1.19

Table III.1-2 DNBR Limits for Events Analyzed with STDP

DNB Correlation	DNBR Limit for Current Analyses with THINC-IV (Typical Cell and Thimble Cell)	DNBR Limit for MUR-PU Analyses with VIPRE (Typical Cell and Thimble Cell)
WRB-2	1.17	1.17
ABB-NV	Not applicable	1.13
WLOP	Not applicable	1.18
W-3 (pressure ≥ 1000 psia)	1.30	Not applicable
W-3 (500 ≤ pressure < 1000 psia)	1.45	Not applicable

III.1.A.4 Conditions for Implementation of VIPRE, the W-3 Alternative DNB Correlations (ABB-NV and WLOP), and the Revised Thermal Design Procedure (RTDP)

For the DNB analyses supporting the Byron Units 1 and 2 and Braidwood Units 1 and 2 MUR power uprate, the VIPRE-W (VIPRE) subchannel analysis code (Reference III.1-4) was used to verify that the DNB design criterion continues to be met for the VANTAGE+ fuel at MUR power uprate conditions. In

Reference III.1-4, the VIPRE code was approved for use with Westinghouse refueling methodology as a direct replacement for the THINC-IV and FACTRAN codes. Also for the MUR power uprate DNB analyses, the NRC-approved W-3 alternative correlations (ABB-NV and WLOP) in Reference III.1-1 were used in place of the W-3 correlation as the secondary DNB correlation for conditions where the primary DNB correlation (WRB-2) is not applicable.

For the implementation of the VIPRE code and the W-3 alternative DNB correlations in the Byron and Braidwood Units DNB-limited safety analyses, the NRC SER Conditions from Reference III.1-19 for the use of the VIPRE code and the NRC SER conditions from Reference III.1-2 for the use of the ABB-NV and WLOP correlations were reviewed. The DNB analyses supporting the Byron and Braidwood MUR power uprate continue to use the RTDP methodology (Reference III.1-9). The NRC SER conditions for the RTDP methodology (Reference III.1-20) were reviewed as well to address the change from the THINC-IV code to the VIPRE code and the use of the ABB-NV correlation with RTDP. The verification of compliance with the NRC SER conditions for these DNB-related topical reports is addressed below for the DNB analyses of the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions.

In Reference III.1-4, the NRC approved the VIPRE code for departure from nucleate boiling (DNB) analysis for the following UFSAR Chapter 15 transient and accidents:

- steam line break
- rod withdrawal from subcritical or power
- loss of forced reactor coolant flow
- locked rotor or shaft break
- dropped rod/bank
- feedwater malfunction

The VIPRE code used for the Byron and Braidwood DNB-limited safety analyses is a configured Quality Assurance (QA) version of the Westinghouse VIPRE-01 code that was approved in WCAP-14565-P-A (Reference III.1-4). The Westinghouse QA program contains provisions for code change control and testing. Every code modification for QA configuration is evaluated in accordance with the Westinghouse procedure on compliance with NRC-approved codes and methods. The configured VIPRE version for Byron and Braidwood has been evaluated to be in full compliance with the methodology in WCAP-14565-P-A. VIPRE is being used under the Westinghouse QA program that has been reviewed by the NRC to meet the requirements of 10 CFR 50, Appendix B, including proper user training and qualification procedures.

III.1.A.4.1 Compliance with NRC SER Conditions on the Use of VIPRE

The NRC Staff reviewed Westinghouse WCAP-14565, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis,” and concluded in a Staff SER (Reference III.1-19) that the generic topical report was an acceptable reference to support plant-specific applications for use of VIPRE-01, provided four Conditions identified in the SER were addressed by the licensees. These four conditions in the SER were considered in the safety analyses for Byron and

Braidwood Stations at the MUR power uprate conditions. The VIPRE application to calculate DNBR for the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions is in compliance with the four SER Conditions from Reference III.1-19, as addressed below. The original SER conditions on the VIPRE-01 code (Reference III.1-21) were addressed in Reference III.1-4.

WCAP-14565-P-A SER Condition 1:

> *Selection of the appropriate critical heat flux (CHF) correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.*

Response:

The WRB-2 correlation with a 95/95 correlation limit of 1.17 approved in Reference III.1-4 was used in the VIPRE DNBR calculations for the VANTAGE+ fuel in Byron and Braidwood Stations. The ABB-NV and WLOP DNB correlations are used for the analysis of the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions when the primary DNB correlation is not applicable. In Reference III.1-1, the ABB-NV and WLOP DNBR limits were approved for use with VIPRE. The 95/95 ABB-NV DNB correlation limit is 1.13 for Westinghouse PWR fuel design applications. The 95/95 WLOP DNB correlation limit is 1.18. The correlation limits used in the MUR power uprate VIPRE DNBR calculations for the VANTAGE+ fuel in Byron and Braidwood Stations are consistent with the approved values in Reference III.1-4 for the WRB-2 correlation and Reference III.1-1 for the ABB-NV and WLOP DNB correlations.

There is no fuel design change associated with the Byron and Braidwood Stations MUR power uprate. The plant-specific hot channel factors and other fuel-dependent parameters in the DNBR calculations for the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions are unchanged from the currently approved values.

WCAP-14565-P-A SER Condition 2:

> *Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.*

Response:

The core boundary conditions used in the VIPRE DNBR calculations for the VANTAGE+ fuel at MUR power uprate conditions are all generated from NRC-approved codes and analysis methodologies. The use of the 1.7% increase in the nominal core power is discussed in the safety evaluation for the MUR power uprate. The remaining reactor core boundary conditions are unchanged from the conservative values that were previously justified for the current operating license. Continued applicability of the core boundary conditions as VIPRE input is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in Reference III.1-12.

WCAP-14565-P-A SER Condition 3:

The NRC Staff's generic SER for VIPRE (Reference III.1-21) set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

Response:

As discussed in response to Condition 1, the WRB-2 correlation with a 95/95 correlation limit of 1.17, approved in Reference III.1-4, was used in the VIPRE DNBR calculations for the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions. The ABB-NV DNBR limit of 1.13 and the WLOP DNBR limit of 1.18 were previously approved in Reference III.1-1 for use with the VIPRE code.

WCAP-14565-P-A SER Condition 4:

Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE (Reference III.1-21) did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

Response:

For the Byron and Braidwood Stations MUR power uprate, application of the VIPRE code as a replacement for the THINC and FACTRAN codes does not include use in the post-CHF region.

III.1.A.4.2 Compliance with NRC SER Conditions on the Use of the W-3 Alternative DNB Correlations (ABB-NV and WLOP)

The NRC Staff reviewed Westinghouse WCAP-14565-P-A, Addendum 2, "Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," and concluded in a Staff SER, Reference III.1-2, that the generic topical report was acceptable for licensing applications, subject to the four limitations and conditions identified in the SER being addressed by the licensees. These four Limitations and Conditions in the SER were considered in the safety analyses for Byron and Braidwood Stations at the MUR power uprate conditions. The application of the ABB-NV and WLOP correlations to calculate DNBR for the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions is in compliance with the four Limitations and Conditions from Reference III.1-2, as addressed below.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 1:

> *The applicable range of the ABB-NV and WLOP correlations are presented in Table 1 and Table 2, respectively, of this SE.*

Response:

For the DNB analyses at MUR power uprate conditions that were based on the ABB-NV and WLOP correlations, the results were confirmed to be within the parameter ranges of the DNB correlations as specified in Table 1 and Table 2, respectively, of Reference III.1-2.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 2:

> *The ABB-NV correlation and the WLOP correlation must use the same F_c factor for power shape correction as used in the primary DNB correlation for a specific fuel design.*

Response:

For the DNB analyses at MUR power uprate conditions that were based on the ABB-NV and WLOP correlations, the F_c factor for power shape correction that was applied was the same as the power shape correction used for the WRB-2 correlation, which is the primary DNB correlation for the VANTAGE+ fuel in Byron and Braidwood Units.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 3:

> *Selection of the appropriate DNB correlation, DNBR limit, engineering hot channel factors for enthalpy rise, and other fuel-dependent parameters will be justified for each application of each correlation on a plant specific basis.*

Response:

The ABB-NV and WLOP DNB correlations are used for analysis of the VANTAGE+ fuel in Byron and Braidwood Units at MUR power uprate conditions when the primary DNB correlation is not applicable. In Reference III.1-1, the current ABB-NV and WLOP DNBR limits were approved for use with VIPRE. The 95/95 ABB-NV DNB correlation limit is 1.13 for Westinghouse PWR fuel design applications. The 95/95 WLOP DNB correlation limit is 1.18. The correlation limits used in the MUR power uprate VIPRE DNBR calculations for the VANTAGE+ fuel in Byron and Braidwood Stations are consistent with the approved values in Reference III.1-1.

There is no fuel design change associated with the Byron and Braidwood Stations MUR power uprate. The plant-specific hot channel factors and other fuel-dependent parameters in the DNBR calculations for the VANTAGE+ fuel in Byron and Braidwood Units at MUR power uprate conditions are unchanged from the currently approved values.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 4:

> *The ABB-NV correlation for Westinghouse PWR applications and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 (VIPRE) code since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications.*

Response:

The Westinghouse version of the VIPRE-01 code subchannel analysis code (Reference III.1-4), which has been qualified and approved with the ABB-NV and WLOP correlations, was implemented for all DNB analyses of the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions. See Section III.1.A.4.1 for compliance with SER conditions on the use of the VIPRE code.

III.1.A.4.3 Compliance with NRC SER Conditions on the Use of RTDP

The NRC Staff reviewed Westinghouse WCAP-11397, "Revised Thermal Design Procedure," and concluded in a Staff SER (Reference III.1-20) that the generic topical report was an acceptable reference to support plant-specific applications for use of RTDP, provided seven Conditions identified in the SER were addressed by the licensees. These seven conditions were considered for Byron and Braidwood Stations at MUR power uprate conditions. The RTDP application for the VIPRE DNB analysis of the VANTAGE+ fuel in Byron/Braidwood at MUR power uprate conditions is in compliance with the seven SER Conditions from Reference III.1-20, as addressed below.

WCAP-11397-P-A SER Condition 1:

> *Sensitivity factors used for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.*

Response:

Sensitivity factors were calculated using the WRB-2 and the ABB-NV DNB correlations and the VIPRE code for parameter values applicable to the VANTAGE+ fuel in Byron and Braidwood Stations at MUR power uprate conditions. These sensitivity factors were used to determine the RTDP design limit DNBR values for both correlations. The design limit DNBR values are included in the Byron/Braidwood UFSAR and Technical Specification updates for the MUR power uprate.

WCAP-11397-P-A SER Condition 2:

> *Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.*

Response:

Because the VIPRE code is used to replace the THINC-IV code for the Byron and Braidwood Stations MUR power uprate, sensitivity factors for the RTDP methodology were calculated using the VIPRE code for parameter values applicable to the VANTAGE+ fuel in Byron and Braidwood Units at MUR power

uprate conditions, as discussed in the response to Condition 1 above. See the Response to SER Condition 3 for a discussion of the use of Equation (2-3) of the topical report.

WCAP-11397-P-A SER Condition 3:

> *If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.*

Response:

As described in Reference III.1-4, the VIPRE code has been demonstrated to be equivalent to the THINC code. Equation (2-3) of WCAP-11397-P-A and the linearity approximation made to obtain Equation (2-17) were confirmed to be valid for the MUR power uprate for the combination of WRB-2 correlation and the VIPRE code as well as for the combination of the ABB-NV correlation and the VIPRE code.

WCAP-11397-P-A SER Condition 4:

> *Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.*

Response:

The only change to the operating parameter uncertainties for the Byron and Braidwood Stations MUR power uprate DNB analyses with RTDP is the reduced power calorimetric uncertainty associated with the use of the LEFM to measure feedwater flow. The reduced power calorimetric uncertainty used for the MUR power uprate is presented in Section I.1.E. The remaining plant operating parameter uncertainties used in the current RTDP DNB analyses are applicable to Byron and Braidwood Stations at the MUR power uprate conditions.

WCAP-11397-P-A SER Condition 5:

> *Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.*

Response:

For the Byron and Braidwood Stations MUR power uprate, nominal initial conditions were only applied to DNBR calculations that used RTDP. Other analyses, such as overpressure calculations, assumed the appropriate conservative initial condition assumptions.

WCAP-11397-P-A SER Condition 6:

> *Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.*

Response:

The Byron and Braidwood Stations MUR power uprate DNBR calculations with RTDP were based on a nominal uprated core power of 3648 MWt (1.017 * 3586.6 MWt). The remaining nominal conditions used in the Byron and Braidwood Stations MUR power uprate DNBR calculations with RTDP are unchanged from the current non-uprated values. The continued applicability of the bounding input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in Reference III.1-12.

WCAP-11397-P-A SER Condition 7:

> *The code uncertainties specified in Table 3-1 (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP.*

Response:

The code uncertainties specified in Table 3-1 of WCAP-11397-P-A remain unchanged and were included in the DNBR analyses using RTDP. The THINC-IV uncertainty was applied to VIPRE, based on the equivalence of the VIPRE model approved in WCAP-14565-P-A to THINC-IV.

III.1.A.5 Analysis Summary

The DNB analyses utilizing VIPRE which were performed to support the MUR power uprate are briefly described below. Additional discussion of these analyses is provided elsewhere in Sections II and III.

III.1.A.5.1 Core Thermal Limits

The core thermal limits are required for the generation of the Overtemperature-ΔT (OTΔT) and Overpower-ΔT (OPΔT) trip setpoints. To support operation at MUR power uprate conditions, new core thermal limits were generated for the VANTAGE+ fuel in the Byron and Braidwood units. The DNB-limited portion of the MUR power uprate core thermal limits was generated with the VIPRE code using the WRB-2 DNB correlation and the RTDP methodology.

III.1.A.5.2 Axial Offset Limits

The axial offset limits are used to reduce the core DNB limit lines to account for the effect of adverse axial power distributions that are more limiting for DNB than the axial power shape used to generate the core thermal limits. New axial offset limits were generated for the VANTAGE+ fuel in the Byron and Braidwood units to address the MUR power uprate conditions. The MUR power uprate axial offset limits were generated with the VIPRE code using the RTDP methodology. For the DNB analysis of axial power distributions that were limiting in the fuel region above the first mixing vane grid, the WRB-2 DNB correlation was used with an RTDP safety analysis limit (SAL) DNBR of [ ]^{a,c}. For the DNB analysis of axial power distributions that were limiting in the fuel region below the first mixing vane grid, the ABB-NV DNB correlation was used with an RTDP SAL DNBR of [ ]^{a,c}.

III.1.A.5.3 Partial / Complete Loss of Forced Reactor Coolant Flow – UFSAR 15.3.1 and 15.3.2

As noted in Sections III.7 and III.8, the loss of flow accident was analyzed for MUR power uprate conditions. The DNBR calculations for the loss of flow accident at MUR power uprate conditions were performed using the VIPRE code to replace THINC-IV and FACTRAN. The DNBR calculations were based on the WRB-2 DNB correlation and the RTDP methodology. The effect of fuel temperatures was included in the analysis of this event. Three cases (partial loss of flow, complete loss of flow, and frequency decay complete loss of flow) were analyzed to ensure the limiting scenario was identified. The results for the partial loss of flow event at MUR power uprate conditions are shown in Section III.7.5. The results for the complete loss of flow events at MUR power uprate conditions are shown in Section III.8.5.

III.1.A.5.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) (Rods-in-DNB) – UFASR 15.3.3 through 15.3.5

As noted in Section III.9, the locked rotor accident was analyzed for MUR power uprate conditions. The locked rotor accident is classified as a Condition IV event. DNBR calculations are performed to quantify the inventory of rods that would experience DNB and be conservatively presumed to fail. The DNBR calculations for the locked rotor rods-in-DNB event at MUR power uprate conditions were performed using the VIPRE code to replace THINC-IV and FACTRAN. The DNBR calculations were based on the WRB-2 DNB correlation and the RTDP methodology. The effect of fuel temperatures was included in the analysis of this event. A conservative fuel rod power census was used to determine the percentage of rods in DNB. The results for the locked rotor rods-in-DNB event at MUR power uprate conditions are shown in Section III.9.5.

III.1.A.5.5 Steam Supply Piping Failure at Zero Power – UFSAR 15.1

As noted in Section III.4, the Hot Zero Power Steam Line Break (HZP SLB) event was analyzed for the MUR power uprate. The NRC-approved Westinghouse analysis method in Reference III.1-10 was used for analyzing the HZP SLB accident. DNBR calculations for the HZP SLB event were performed at MUR power uprate conditions using the VIPRE code, the WLOP correlation, and the STDP methodology. The WLOP correlation was used for this application because the system pressure was less than the low pressure limit of applicability for the primary DNB correlation. The STDP methodology was used because the event is initiated from Hot Zero Power conditions. Conservative accident-specific axial and radial power distributions were applied. The results for the HZP SLB event at MUR power uprate conditions are shown in Section III.4.5.

III.1.A.5.6 Steam System Piping Failure at Full Power – UFSAR 15.1.6

As noted in Section III.5, the Hot Full Power Steam Line Break (HFP SLB) event was analyzed for MUR power uprate conditions. DNBR calculations for the HFP SLB accident at MUR power uprate conditions were performed using the VIPRE code, the WRB-2 DNB correlation, and the RTDP methodology. The results for the HFP SLB event at MUR power uprate conditions are shown in Section III.5.5.

III.1.A.5.7 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature or an Increase in Feedwater Flow– UFSAR 15.1.1 and 15.1.2

As noted in Section III.2, the Hot Full Power Feedwater Malfunction (HFP FWM) event was analyzed for MUR power uprate conditions. DNBR calculations for the HFP FWM accident at MUR power uprate conditions were performed using the VIPRE code, the WRB-2 DNB correlation, and the RTDP methodology. The results for the HFP FWM event at MUR power uprate conditions are shown in Section III.2.5.

III.1.A.5.8 Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error) – UFSAR 15.4.3

As noted in Section II.2.6, the statepoints for the rod cluster control assembly (RCCA) drop event are unaffected by the 1.7% core power uprate. The NRC-approved Westinghouse analysis methods in Reference III.1-11 continue to be used for analyzing the RCCA drop event at MUR power uprate conditions. The Dropped Rod Limit Lines (DRLL) were generated to define the loci of points that result in a VIPRE-calculated minimum DNBR equal to the WRB-2 RTDP safety analysis DNBR limit for a wide range of core conditions (inlet temperature, power, and pressure). The DRLL are used to verify that the DNB design basis is met each cycle for the RCCA drop event at MUR power uprate conditions.

The maximum allowable $F^N_{\Delta H}$ limit for RCCA misalignment was determined using the VIPRE code, the WRB-2 DNB correlation, and the RTDP methodology at 101.7% (3648 MWt). This is the value of $F^N_{\Delta H}$ at normal operating conditions that results in a minimum DNBR equal to the WRB-2 RTDP safety analysis DNBR limit.

The limits provided for the RCCA drop and RCCA misalignment events are used to confirm that the DNB design basis is met for Byron and Braidwood reload cores operating at MUR power uprate conditions.

III.1.A.5.9 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Conditions – UFSAR 15.4.1

As noted in Section II.2.5, the statepoints for this zero power event are unaffected by the 1.7% core power uprate. The limiting heat flux statepoints are defined as a fraction of the nominal heat flux. DNBR calculations for the Uncontrolled Rod Cluster Control Assembly Withdrawal from Subcritical (RWFS) event were performed at MUR power uprate conditions to incorporate the VIPRE code and the ABB-NV correlation. The DNBR calculations for this event were based on the STDP methodology, since the event is initiated from Hot Zero Power conditions. Conservative accident-specific axial and radial power distributions were used in the DNB analysis. Two DNBR calculations were required for this event. The ABB-NV correlation was applied in the fuel region below the first mixing vane grid. The WRB-2 correlation was applied in the fuel region above the first mixing vane grid. The DNB criterion continues to be met for the RWFS event.

III.1.B Fuel Structural Evaluation

Evaluations for the MUR power uprate have no impact on the fuel assembly structural integrity. The original core plate motions remain applicable for the MUR power uprate. Therefore, there is no impact

on the fuel assembly seismic/LOCA structural evaluation. The MUR power uprate has an insignificant impact on the operating and transient loads, such that there is no adverse affect on the fuel assembly functional requirements. The fuel assembly structural integrity is not affected and the seismic and LOCA evaluations of the fuel are still applicable.

III.1.C Nuclear Design Evaluation

The standard set of reload core design criteria (Reference III.1-12) have been confirmed via evaluation or explicit analysis for the transition to an uprated core power level. For all Reload Safety Analysis Checklist (RSAC) items analyzed for each cycle, adequate margins to the limits have been demonstrated for recent cycles to provide assurance that these limits will not be challenged by the increase in core power level. Cycle-specific calculations are performed for each reload cycle. These cycle-specific analyses and evaluations are performed to ensure that all core design and RSAC criteria will be satisfied for the specific operating conditions of that cycle.

III.1.D Fuel Rod Design Evaluation

Cycle-specific fuel rod design analyses are performed using the NRC-approved models (References III.1-13 and III.1-14) and NRC-approved design criteria and methods (References III.1-15, III.1-16, and III.1-17) to ensure that fuel rod design criteria are satisfied for each reload cycle. The fuel rod design criteria evaluated include: rod internal pressure (gap reopening), cladding stress and strain, cladding oxidation and hydriding, fuel temperature, cladding fatigue, cladding flattening, fuel rod axial growth, plenum cladding support, and cladding free standing. These models, methods, and criteria remain unchanged for the MUR power uprate from those currently used for Byron Units 1 and 2 and Braidwood Units 1 and 2 analyses.

The methodology for confirming that extensive DNB propagation does not occur has changed. Statistical methods were previously used, while the MUR power uprate will implement the mechanistic method previously reviewed by the NRC for W-NSSS plants in Reference III.1-18. The design criteria provided in Reference III.1-15, which states that rod internal pressure will not cause extensive DNB propagation to occur, still applies to this analysis, so this change does not represent a change to any analysis of record (AOR).

III.1.D.1 Compliance with NRC SER Conditions on the Use of the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)

The NRC Staff reviewed Westinghouse WCAP-8963-P-A, Addendum 1-A, Revision 1-A, “Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)” and concluded in a Staff SER, Reference III.1-18, that the generic topical report was acceptable for licensing applications, subject to the two Limitations and Conditions identified in the SER being addressed by the licensees. These two conditions in the SER were considered in the safety analyses for Byron and Braidwood Stations at MUR power uprate conditions. The application of Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology) for Byron and Braidwood Stations at MUR power uprate

conditions is in compliance with the two SER conditions. The two SER conditions from Reference III.1-18 are addressed below.

WCAP-8963-P-A, Addendum 1-A, Revision 1-A SER Condition 1:

> *Fuel Rods will not burst prior to reaching the critical strain during the DNB propagation.*

Response:

The evaluation of Byron and Braidwood Stations MUR power uprate postulated non-LOCA accident conditions and fuel design confirmed that the ballooning strain was well below the critical strain threshold to cause either DNB propagation or clad burst.

WCAP-8963-P-A, Addendum 1-A, Revision 1-A SER Condition 2:

> *Any deviation from the current or existing approach in WCAP-8963-P-A, Addendum 1, Revision 1 constitutes a new issue and requires review by the NRC staff.*

Response:

There is no deviation in the approach and method used in the Byron and Braidwood Stations MUR power uprate evaluation from the NRC-approved approach in Reference III.1-18.

III.1.E Conclusion

The fuel design bases are met for the Byron Units 1 and 2 and Braidwood Units 1 and 2 at MUR power uprate conditions. Cycle specific evaluations to confirm that the fuel parameters are met for each reload at MUR power uprate conditions will be performed in accordance with Reference III.1-12.

III.1.F Refer ences

III.1-1 WCAP-14565-P-A, Addendum 2-P-A, “Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications,” Leidich, A. R., et al., April 2008.

III.1-2 Letter from Ho K. Nieh (NRC) to J. A. Gresham (Westinghouse), “Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-14565-P, Addendum 2, Revision 0, 'Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP [Westinghouse Low Pressure] for PWR [Pressurized Water Reactor] Low Pressure Applications' (TAC NO. MD3184),” February 14, 2008.

III.1-3 Tong, L. S., AEC Critical Review Series, “Boiling Crisis and Critical Heat Flux,” TID-25887, August 1972.

III.1-4 WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Sung, Y., Schueren, P., and Meliksetian, A., October 1999.

III.1-5 WCAP-7956-A, "THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," Hochreiter, L. E., Chelemer, H., and Chu, P. T., February 1989.

III.1-6 WCAP-12330-A, "Improved THINC-IV Modeling for PWR Core Design," Friedland, A. J. and Ray, S., September 1991.

III.1-7 WCAP-7908-A, "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove (edited by P. W. Robertson), December 1989.

III.1-8 WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly," Davidson, S. L. and Kramer, W. R., September 1985.

III.1-9 WCAP-11397-P-A, "Revised Thermal Design Procedure," Friedland, A. J. and Ray, S., April 1989.

III.1-10 WCAP-9226-P-A, Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," Scherder, W. J. (Editor), et al., February 1998.

III.1-11 WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," Haessler, R. L., et al., January 1990.

III.1-12 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," Davidson, S. L. (Editor), et al., July 1985.

III.1-13 WCAP-15063-P-A, Revision 1 with Errata (Proprietary), "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," Slagle, W. H. (Editor), July 2000.

III.1-14 WCAP-12610-P-A (Proprietary), "VANTAGE+ Fuel Assembly Reference Core Report," Davidson, S. L. and Ryan, T. L., April 1995.

III.1-15 WCAP-10125-P-A (Proprietary), "Extended Burnup Evaluation of Westinghouse Fuel," Davidson, S. L. (Editor), December 1985.

III.1-16 WCAP-13589-A (Proprietary), "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," Kersting, P. J., et al., March 1995.

III.1-17 WCAP-12488-A, Addendum 1-A, Revision 1 (Proprietary), "Addendum 1 to WCAP-12488-A, Revision to Design Criteria," January 2002.

III.1-18 WCAP-8963-P-A, Addendum 1-A, Revision 1-A (Proprietary), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)," June 2006.

III.1-19 Letter from T. H. Essig (NRC) to H. Sepp (W), “Acceptance for Referencing of Licensing Topical Report WCAP-14565, VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis (TAC NO. M98666),” January 19, 1999.

III.1-20 Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), “Acceptance for Referencing of Licensing Topical Report WCAP-11397, ‘Revised Thermal Design Procedure’,” January 17, 1989.

III.1-21 Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), “Acceptance for Referencing of Licensing Topical Report, EPRI-NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4,” May 1, 1986.

III.2 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature or an Increase in Feedwater Flow– UFSAR 15.1.1 and 15.1.2

III.2.1 Identification of Causes and Accident Description

A change in steam generator feedwater conditions that results in an increase in feedwater flow and/or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. An accidental opening of a feedwater bypass valve, which diverts flow around a portion of the feedwater heaters, is an event that causes a reduction in feedwater inlet temperature to the steam generators. An accidental full opening of one or more feedwater control valves would cause excessive feedwater flow to one or more of the steam generators. Both reduced feedwater temperature and increased feedwater flow are feedwater system malfunctions that produce increased subcooling in the affected steam generators.

At power, this increased subcooling will create a greater load demand on the Reactor Coolant System (RCS) with a resulting decrease in RCS temperature. In the presence of a negative moderator temperature coefficient, the decrease in RCS temperature will produce a reactivity insertion. The thermal capacity of the secondary plant and of the RCS attenuates the increase in core power from these reductions in feedwater temperature. The overpower – overtemperature protection systems (neutron overpower, overtemperature and overpower ΔT trips) are designed to prevent any power increase that could lead to a Departure from Nucleate Boiling Ratio (DNBR) less than the limit value.

With the plant at no-load conditions, the addition of cold feedwater may also cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease; therefore, the no-load transient is less severe than the full power case.

The net effect on the RCS due to a reduction in feedwater temperature would be similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT. In addition to the overpower – overtemperature protection systems, the steam generator high-high level trip, which closes the feedwater valves, is a protection function credited for mitigating the consequences of a feedwater system excessive flow malfunction.

III.2.2 Method of Analysis

The feedwater system malfunction for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN (Reference III.2-1), VIPRE-W (Reference III.2-2), and ANC (Reference III.2-3) computer codes and Revised Thermal Design Procedure (RTDP) methodology (Reference III.2-4) to calculate the minimum DNBR and peak linear heat rate (PLHR) values. The current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be the nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) power uprate.

III.2.3 Analysis Inputs and Assumptions

Cases for both Byron Unit 1 and Braidwood Unit 1 with Babcock and Wilcox (B&W) International replacement steam generators and Byron Unit 2 and Braidwood Unit 2 with Westinghouse Model D5 steam generators were analyzed at conditions designed to provide the most conservative and limiting results. Specifically, the following assumptions are made:

a. Initial reactor power, pressure, and RCS temperatures (consistent with the MUR uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias (1.5°F), which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.2-2.

b. The minimum steam generator tube plugging level is assumed, and maximum feedwater temperature is analyzed for each steam generator design.

c. A full power moderator density coefficient of 0.43 Δk/gm/cc, corresponding to maximum feedback for the feedwater malfunction event, is modeled. This input is bounding for MUR power uprate conditions.

d. A least negative Doppler-only power coefficient of -11 pcm/% power is assumed such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

e. The overpower ΔT reactor trip function is credited as being available to mitigate the effects of this event. In addition, for the excessive flow feedwater malfunction event, the steam generator high-high water level function is modeled, which provides a feedwater isolation signal.

f. The most limiting single failure for a feedwater system malfunction event is the failure of a protection train. No single active failure will prevent the reactor protection system from functioning properly.

III.2.4 Analysis Acceptance Criteria

A decrease in normal feedwater temperature is classified as an ANS Condition II event, fault of moderate frequency. The analysis for the feedwater system malfunction is performed to confirm that the DNB design basis is satisfied. In addition, the analysis is performed to confirm that the PLHR (kW/ft) does not exceed the limit value that precludes fuel centerline melting.

III.2.5 Analysis Results

The most limiting case for feedwater system malfunction is a reduction in feedwater temperature event with D5 steam generators at the minimum steam generator tube plugging (SGTP) level with maximum feedwater temperature. Results from the analysis of this limiting case are shown on Figures III.2-1 through III.2-5. Figure III.2-1 illustrates the nuclear power transient following the reduction in feedwater temperature. Because of the reactivity insertion produced by the resulting RCS temperature reduction, nuclear power increases from its initial value until reactor trip occurs on overpower ΔT. The core coolant

average temperature transient, pressure decay transient, and loop delta-T transient following the accident are given in Figures III.2-2, III.2-3, and III.2-4. Loop delta-T, pressure and core heat generation are reduced via the trip. The DNBR decreases initially, but increases rapidly following the trip as shown in Figure III.2-5. The calculated minimum DNBR value for the MUR power uprate is greater than the DNBR safety analysis limit of [ ][a,c]. Also, as indicated by the results reported in Table III.2-2, the PLHR remains below that which could produce fuel centerline melting. Therefore, all applicable acceptance criteria are met for the feedwater malfunction event at MUR power uprate conditions. The calculated sequence of events for the feedwater system malfunction event is shown in Table III.2-1.

The comparison of the results from the limiting case analyzed for the MUR power uprate to those from the limiting case in the current licensing basis analysis of record (AOR) is presented in Table III.2 2. The comparisons show that, in terms of maximum relative core power, the protection system limits the peak heat flux for the MUR analysis to roughly the same value as predicted for the AOR. The MUR power uprate analysis reflects an increase in the nominal core power and the use of a more conservative moderator density coefficient than was used in the AOR. The results show that the minimum DNBR is lower for the MUR power uprate than for the AOR, and the DNB design basis continues to be met for the MUR power uprate. The primary reason for the reduction in DNBR is the application of greater conservatisms in the MUR power uprate analysis to bound cycle-to-cycle variations expected in future reload core designs.

III.2.6 References

III.2-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.2-2 WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

III.2-3 WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” September 1986.

III.2-4 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

Table III.2-1 Sequence of Events

Event	Time (Seconds)
Feedwater Heater Bypass Valve Opens Fully	0.0
OPΔT trip setpoint reached	5.9
Rods Begin to Drop	13.9
Minimum DNBR Occurs	14.5
SIS Low Pressurizer Pressure Setpoint	34.8
Feedwater Isolation Occurs	41.8

Table III.2-2 Comparison of Limiting Results

Parameter	AOR	MUR
Calculated Peak Core Heat Flux (% of nominal core power)	120.2	120.6
Calculated Minimum DNBR	[ ]a,c	[ ]a,c
Safety Analysis Limit DNBR (DNB Correlation)	[ ]a,c	[ ]a,c
Calculated Peak Linear Heat Rate (kW/ft)	21.80	21.96
Peak Linear Heat Rate Limit for Fuel Melting (kW/ft)	22.40	22.30

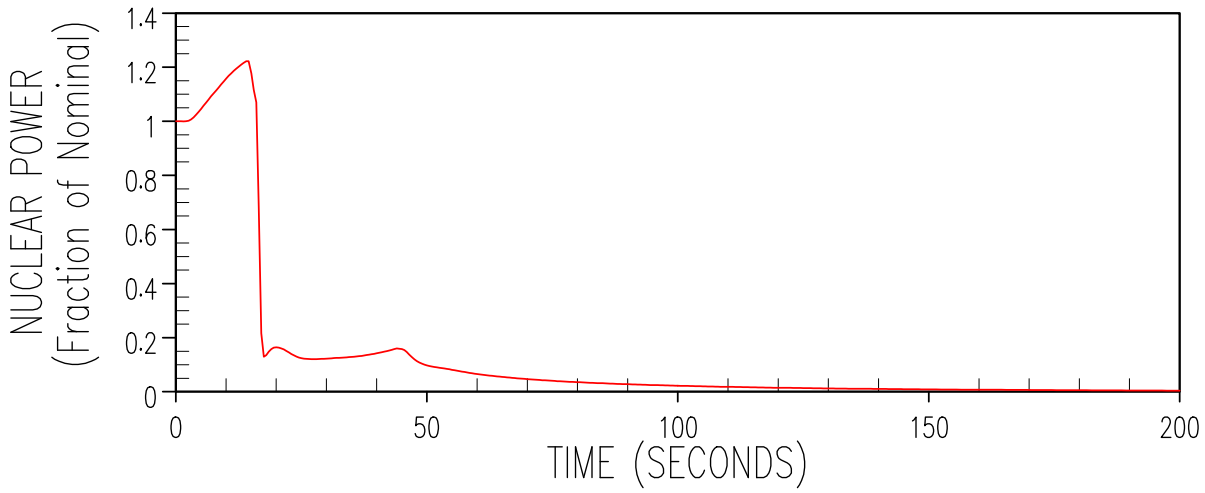

Figure III.2-1 Nuclear Power versus Time

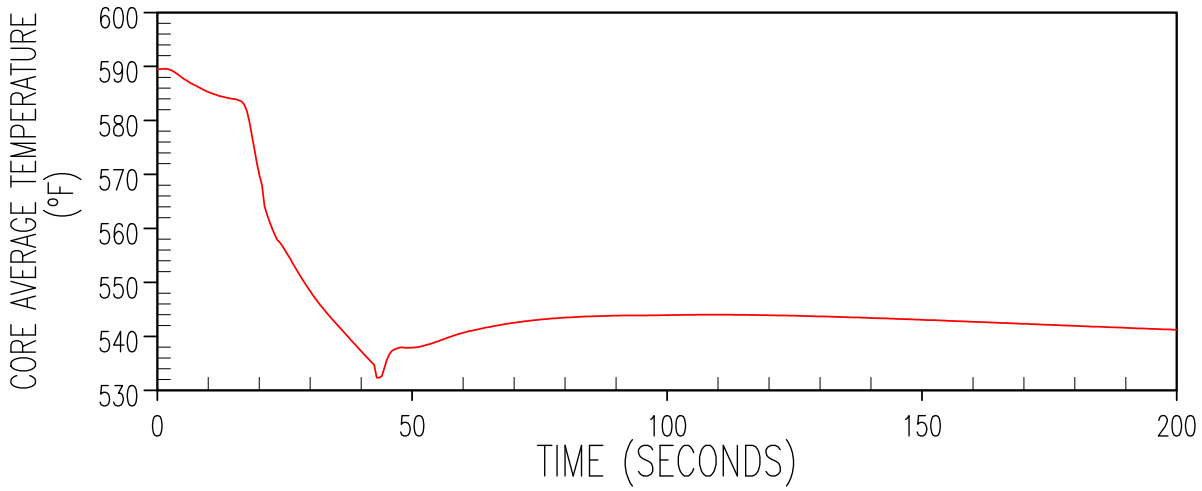

Figure III.2-2 Core Average Temperature versus Time

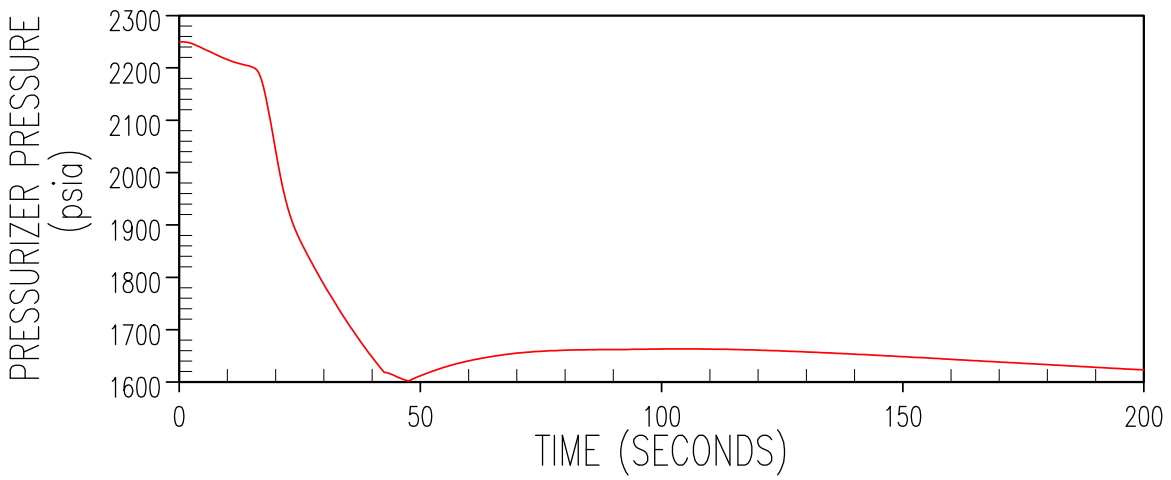

Figure III.2-3 Pressurizer Pressure versus Time

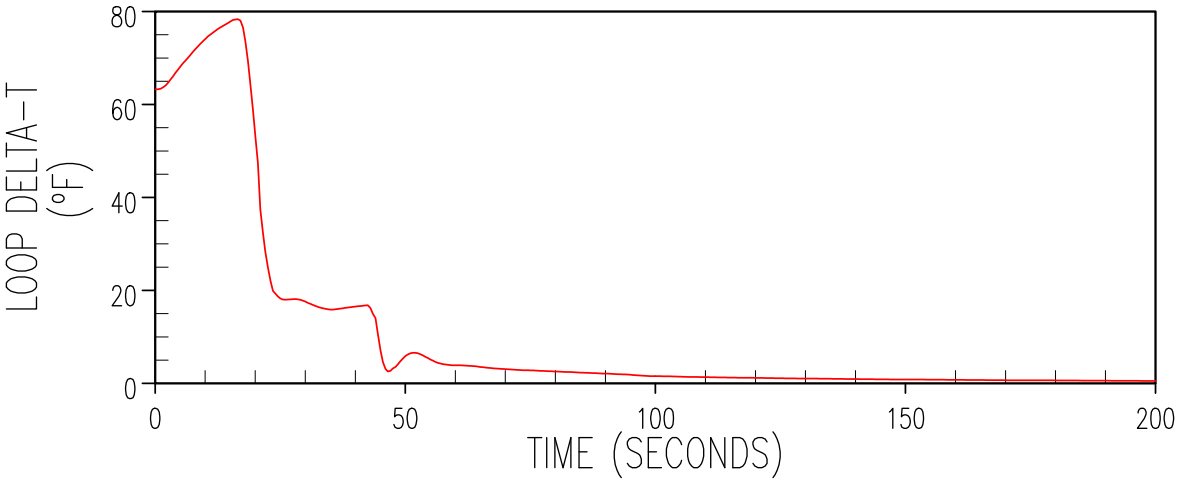

Figure III.2-4 Loop Delta-T versus Time

a,c

Figure III.2-5 DNBR versus Time

III.3 Excessive Increase in Secondary Steam Flow – UFSAR 15.1.3

III.3.1 Identification of Causes and Accident Description

An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase or a 5% per minute ramp load increase between 15% power and 100% power. Any loading rate in excess of these values could cause a reactor trip via the reactor protection system. Steam flow increases greater than 10% are discussed in Sections III.4 (Zero Power SLB) and III.5 (Full Power SLB) of this report.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control systems.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals (e.g., high reactor coolant temperature is an indication that steam dump is needed). A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system (RPS) signals:

- Low pressurizer pressure
- Overtemperature ΔT, and
- Power range high neutron flux

III.3.2 Method of Analysis

The excessive load increase accident for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN computer code (Reference III.3-1) and Revised Thermal Design Procedure (RTDP) methodology (Reference III.3-2) to calculate a minimum departure from nucleate boiling ratio (DNBR). The current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) power uprate.

III.3.3 Analysis Inputs and Assumptions

Cases for both Byron Unit 1 and Braidwood Unit 1 with Babcock and Wilcox (B&W) International steam generators and Byron Unit 2 and Braidwood Unit 2 with Westinghouse Model D5 steam generators were analyzed at conditions chosen to provide the most conservative and limiting results. Specifically, the following assumptions are made:

a. Initial reactor power, pressure, and RCS temperatures (consistent with the MUR uprated power conditions) are modeled to be at their nominal values. With the exception of the

RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.3-2.

b. Cases are run both with minimum (0%) steam generator tube plugging and maximum (5% – BWI generators and 10% – Model D5 generators) steam generator tube plugging. The maximum feedwater temperature yields more limiting results so only the maximum feedwater temperature (449.2°F) is modeled in the analyses for each steam generator design.

c. Cases are run with both minimum (beginning-of-life (BOL)) and maximum (end-of-life (EOL)) reactivity feedback assumptions. The minimum feedback cases model a full power moderator temperature coefficient of 0 pcm/°F, a least negative Doppler temperature coefficient, a least negative Doppler power coefficient and a maximum delayed neutron importance (β_{eff}). The maximum feedback cases model a very large (absolute value) negative full power moderator temperature coefficient, most negative Doppler temperature and power coefficients and a minimum delayed neutron importance (β_{eff}). Note that use of a zero moderator temperature coefficient (MTC) at full power for the minimum reactivity feedback cases is appropriate even for a plant employing a positive moderator temperature coefficient (PMTC). Typically, a PMTC is allowed by the plant Technical Specifications from 0% to some part power level (in this case 70%), at which point the PMTC linearly ramps to zero (see Technical Specifications Figure 3.1.3-1). The reactivity feedback assumptions are consistent with the current licensing basis analyses. A least negative Doppler-only power coefficient is assumed such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback. Note that all reactivity feedback coefficients are revalidated for each reload core during the Westinghouse reload process.

d. Separate cases are run assuming that the plant is in both manual and automatic rod control.

e. The most limiting single failure for an excessive load increase incident is the failure of a protection train. No reactor trip is anticipated and no single active failure will prevent the reactor protection system from functioning properly or yield more limiting analysis results.

III.3.4 Analysis Acceptance Criteria

An excessive load increase event is classified as an ANS Condition II event, a fault of moderate frequency. The criterion of interest for the excessive load increase event is that the DNB design basis is satisfied.

III.3.5 Analysis Results

The most limiting case assumes minimum reactivity feedback, automatic rod control and the BWI steam generators with zero steam generator tube plugging. The worst case minimum DNBR is [ ]^{a,c} compared to a DNBR limit of [ ]^{a,c} for over 20% safety analysis margin. The peak heat flux for the limiting case is 112.4% compared to a limit of 119%. Note that the same case for the Westinghouse Model D5 steam generators yields essentially the same results ([ ]^{a,c} minimum DNBR and a peak heat flux of 112.3%). Transient excessive load increase plots from the limiting case are shown on Figures III.3-1 through III.3-5. Figure III.3-1 illustrates the nuclear power transient during an excessive load

increase event. The pressurizer pressure and pressurizer water volume during the transient are given in Figures III.3-2 and III.3-3. The core average temperature transient is shown in Figure III.3-4 and the DNBR transient is shown in Figure III.3-5. As is seen in Figure III.3-1, nuclear power increases to about 110% with short term increases to about 112% due to the overly conservative differential rod worth assumed in the analysis. Pressurizer pressure, pressurizer water volume and core average temperature change very little during the transient. The transient DNBR drops slightly and equilibrates between [ ]^{a,c} and [ ]^{a,c} – well above the DNBR limit value of [ ]^{a,c}.

A comparison of the results from the limiting case analyzed for the MUR power uprate to those from the limiting case in the current licensing basis analysis is presented in Table III.3-2. From this comparison it can be seen that, despite the increase in power associated with the MUR power uprate, the decrease in the overall minimum DNBR calculated in the MUR power uprate analysis compared to the current licensing basis is small.

III.3.6 References

III.3-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.3-2 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

Table III.3-1 Sequence of Events – BWI SGs, 0% SGTP, Minimum Reactivity Feedback, Automatic Rod Control

Event	Time (Seconds)
Steam Flow Increases by 10%	0
Minimum DNBR Occurs ([ ]^{a,c})	530
Peak Heat Flux Occurs (112.4%)	536
Transient Terminated	600

Table III.3-2 Comparison of Results to the Current Licensing Basis

Limiting Case Description	DNBR Safety Analysis Limit	Calculated Value
Limiting Licensing Basis Case – BWI SGs, 0% SGTP, minimum reactivity feedback, automatic rod control	[ ]^{a,c}	[ ]^{a,c}
Limiting MUR Case – BWI SGs, 0% SGTP, minimum reactivity feedback, automatic rod control	[ ]^{a,c}	[ ]^{a,c}

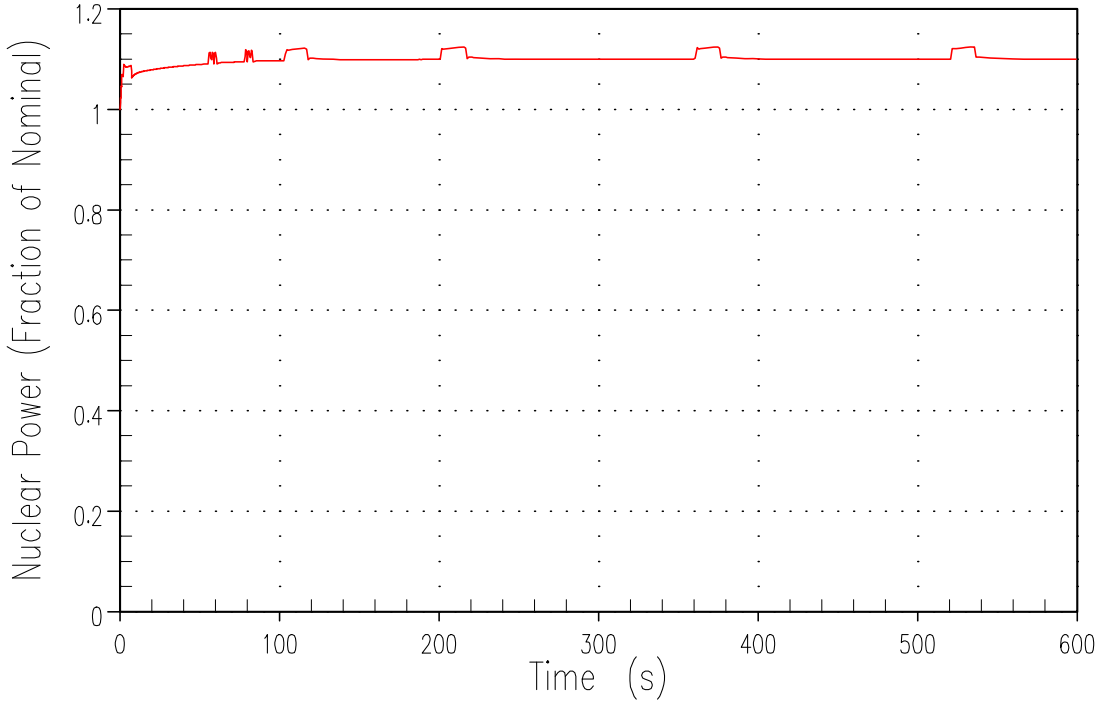

Figure III.3-1 Nuclear Power versus Time

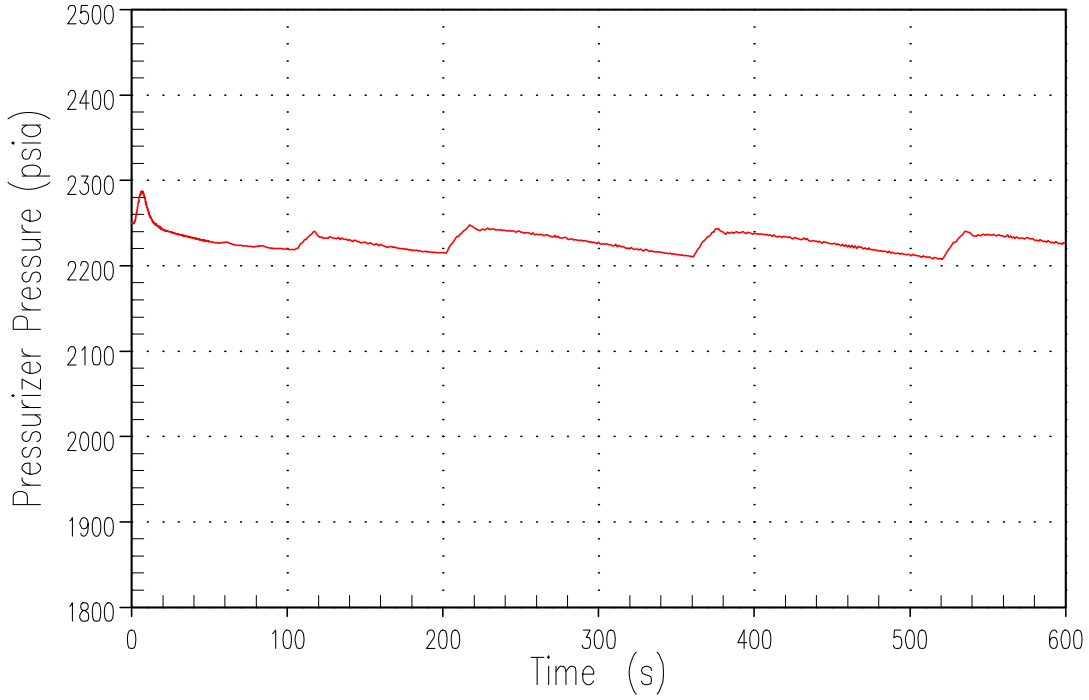

Figure III.3-2 Pressurizer Pressure versus Time

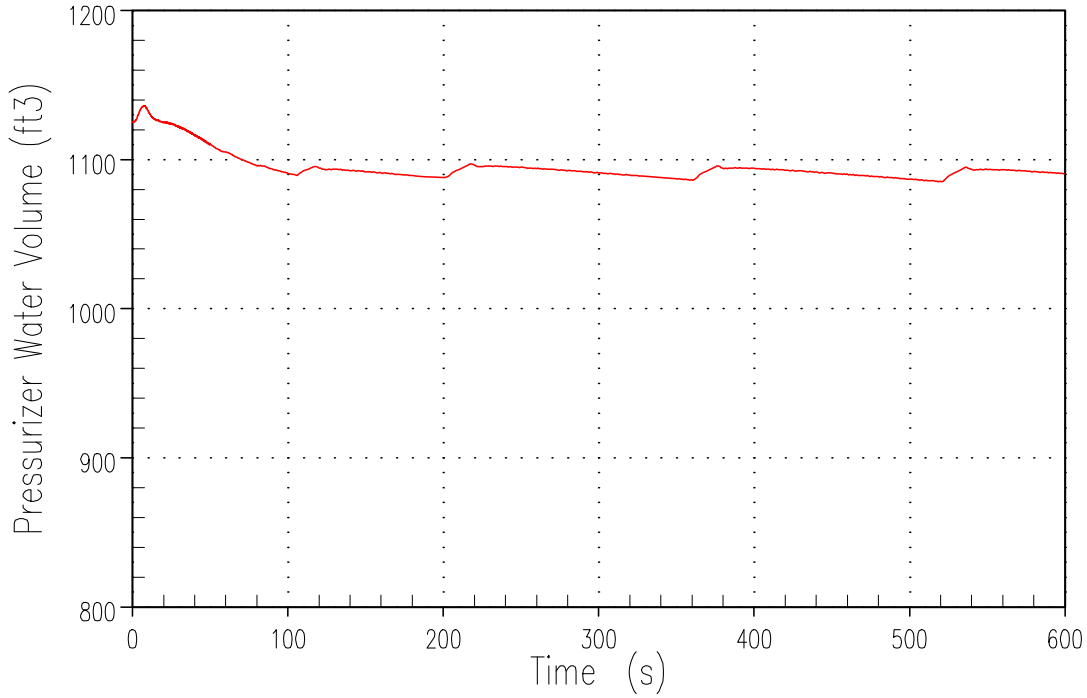

Figure III.3-3 Pressurizer Water Volume versus Time

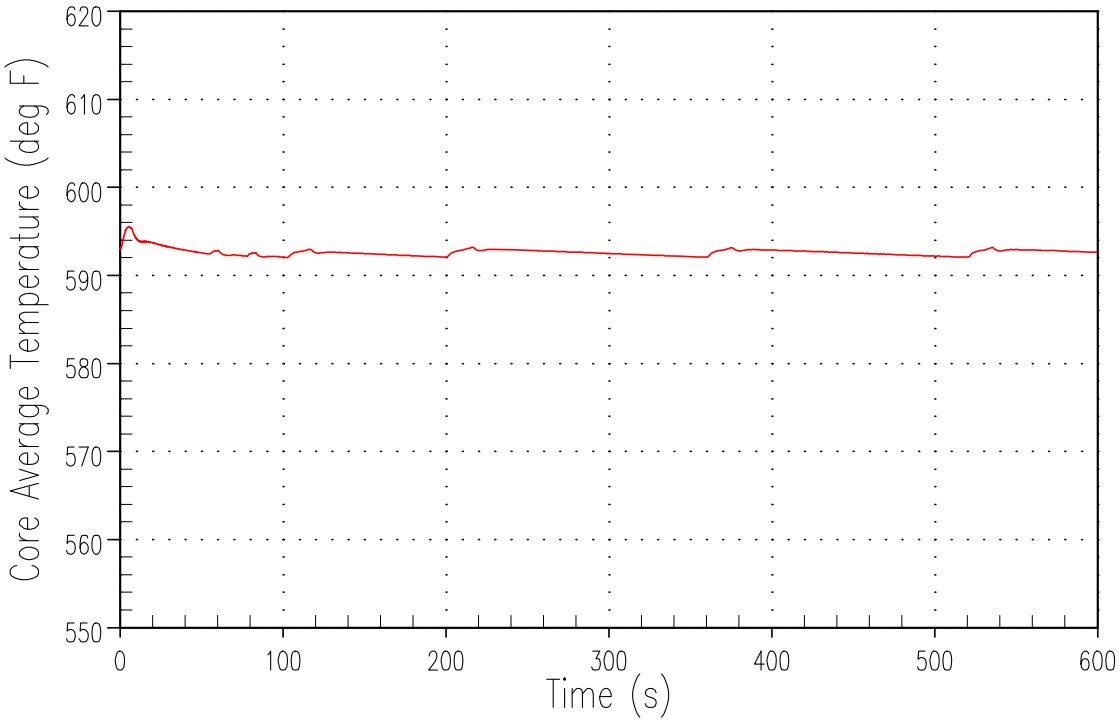

Figure III.3-4 Core Average Temperature versus Time

a,c

Figure III.3-5 DNBR versus Time

III.4 Steam Supply Piping Failure at Zero Power – UFSAR 15.1.5

III.4.1 Identification of Causes and Accident Description

The steam release arising from a break of a main steamline would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system (RCS) causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steamline break is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the high head safety injection system.

The major break of a steamline is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Cases are run for operation with and without offsite power available.

For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.1 ft^2 for Units 1 and 1.4 ft^2 for Units 2, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

III.4.2 Method of Analysis

The analysis of a ruptured steamline at zero power for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN (Reference III.4-1), VIPRE (Reference III.4-2), and ANC (Reference III.4-3) computer codes along with the non-statistical Standard Thermal Design Procedure (STDP) methodology to calculate the minimum departure from nucleate boiling ratio (DNBR) and peak linear heat rate (PLHR) values. The event was reanalyzed to address the revised reactivity feedback coefficients associated with the MUR power level increase.

III.4.3 Analysis Inputs and Assumptions

The following conditions were assumed to exist at the time of a main steam break accident:

a. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position are modeled. Operation of the control rod banks during core burn-up is restricted in such a way that the addition of positive

reactivity in a steamline break accident will not lead to a more adverse condition than the case analyzed.

b. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position is modeled.

c. The minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single failure in the emergency core cooling system (ECCS) is modeled. The ECCS consists of:

1) the passive accumulators,
2) the residual heat removal (low head safety injection) system,
3) the safety injection (intermediate head) system, and
4) the centrifugal charging (high head safety injection) system.

Only the high head safety injection system is modeled for the steamline break accident analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downstream of the refueling water storage tank prior to the delivery of concentrated boric acid to the reactor coolant loops.

d. The design value of the steam generator heat transfer coefficient including allowance for fouling is modeled.

e. Only one break size is examined for each steam generator type corresponding to the cross-sectional area of the integral flow restrictors (i.e., 1.1 ft^2 for Units 1 and 1.4 ft^2 for Units 2). Any break with a break area greater than the area of the flow restrictor, regardless of location, would have the same effect on the NSSS as the break equal to the area of the flow restrictor. The following cases have been considered in determining the core power and RCS transients:

Case 1: Complete severance of a pipe, with the plant initially at no-load conditions, with offsite power available. Since offsite power is available, the pumps are not tripped and thus maintain full RCS flow.

Case 2: Case 1 with loss of offsite power coincident with the steamline break. Loss of offsite power results in reactor coolant pump coastdown, which is assumed to begin 3 seconds after the break occurs.

f. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steamline break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power

peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied. The core parameters used for both with and without offsite power cases correspond to values determined from the respective transient analysis. Both cases assume initial hot shutdown conditions at time zero since this represents the most adverse initial condition. The hot shutdown initial conditions were considered for cases assuming initial pressurizer water volumes for both the high (588.0°F) and low (575.0°F) T_{avg} programs.

g. In computing the steam flow during a steamline break, the Moody Curve (Reference III.4-4) for f(L/D) = 0 is used.

h. Perfect moisture separation in the steam generator is assumed.

The following functions provide the protection for a steamline break:

i. Safety injection system actuation from any of the following:

1) Two-out-of-three low steamline pressure signals in any one loop,

2) Two-out-of-four low pressurizer pressure signals, or

3) Two-out-of-three high-1 containment pressure signals.

j. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

k. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves and backup feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

l. Trip of the fast acting steamline stop valves on:

1) Two-out-of-three low steamline pressure signals in any one loop.

2) Two-out-of-three high-2 containment pressure signals.

3) Two-out-of-three high negative steamline pressure rate signals in any one loop (used only during cooldown and heatup operations).

III.4.4 Analysis Acceptance Criteria

A main steamline rupture is classified as an ANS Condition IV event, a limiting fault. For this event, the main criterion is that any consequential damage to the core must not preclude long-term core cooling and that any offsite dose consequence must be within the acceptable limits of the dose methodology used by the utility. Westinghouse conservatively applies the Condition II acceptance criteria to the event; specifically that the DNBR and PLHR values are met such that damage to the fuel rods is precluded.

III.4.5 Analysis Results

Four cases are run for each generator type, considering initial pressurizer water volumes corresponding to both T_{avg} programs as well as operation with and without offsite power available. Based upon the peak heat flux calculated for each case, the limiting zero power steamline rupture case corresponds to Units 2 (D5 generator) with a break size of 1.4 ft^2, AC power available, and a low T_{avg}. The calculated minimum DNBR value ([ ]^{a,c})is above the limit value (1.18 for the WLOP DNBR correlation) and that the maximum PLHR (18.98 kW/ft) is below the limit value (22.3 kW/ft). The fuel rod end plug weld criteria are also met for the event.

Figures III.4-1 through III.4-7 present the transient responses for the limiting zero power steamline rupture case. As shown in Figure III.4-7, the core attains criticality with the RCCAs inserted (assuming one stuck RCCA) before the boric acid solution enters the RCS. The event is terminated when the boron reaches the reactor core which mitigates the return to power. It should be noted that following the steamline break, only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. The K_{eff} versus temperature corresponding to the negative temperature coefficient used is shown in Figure III.4-8. The effect of power generation in the core on the overall reactivity is shown in Figure III.4-9. The calculated sequence of events is shown in Table III.4-1.

A comparison of the results from the MUR power uprate analysis to those from the current licensing basis analysis is presented in Table III.4-2. Although the transient progression remains essentially the same, the MUR power uprate reactivity feedback coefficients created a more severe return-to-power resulting in a higher peak heat flux. Furthermore, the nuclear analyses are performed in a more conservative manner in order to bound cycle to cycle variations expected in future reloads with a conservative reference analysis. Thus, due to the increased heat flux and the conservatisms in the nuclear analyses, the reduction in DNBR and PLHR margin for this analysis is consistent with expectations.

III.4.6 References

III.4-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.4-2 WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

III.4-3 WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” September 1986.

III.4-4 Journal of Heat Transfer, “Transactions of the ASME,” Figure 3, page 134, February 1965.

Table III.4-1 Sequence of Events

Event	Time (seconds)
Steamline breaks	0.0
Low Steam Pressure Safety Injection setpoint reached	0.7
Feedwater isolation occurs	7.7
Steamline isolation occurs	8.7
Pressurizer empties	~24.8
Criticality attained	26.8
Boron reaches core	~132.8
Time of PLHR	144.6
Time of minimum DNBR	144.6

Table III.4-2 Comparison of Results to the Current Licensing Basis

Case	Peak Heat Flux (FON)	Minimum DNBR	DNBR Limit	PLHR (kW/ft)	PLHR Limit (kW/ft)
AOR - D5 SGs, Low T_{avg}, Offsite Power Available	12.0	[ ]^{a,c}	1.45	15.7	22.4
MUR - D5 SGs, Low T_{avg}, Offsite Power Available	12.9	[ ]^{a,c}	1.18	18.98	22.3

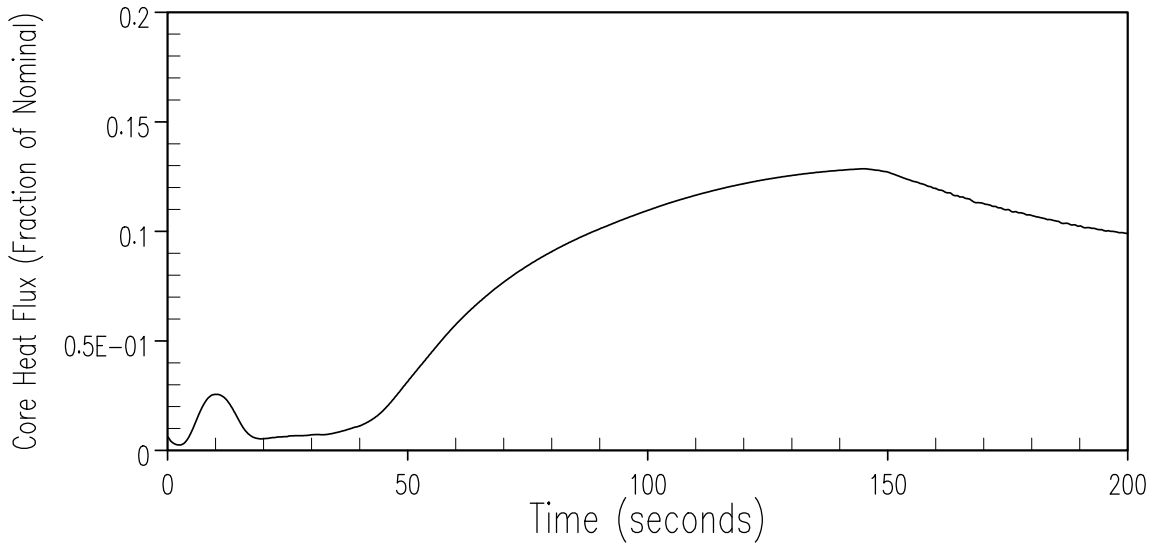

Figure III.4-1 Hot Zero Power Steamline Break – Heat Flux vs. Time

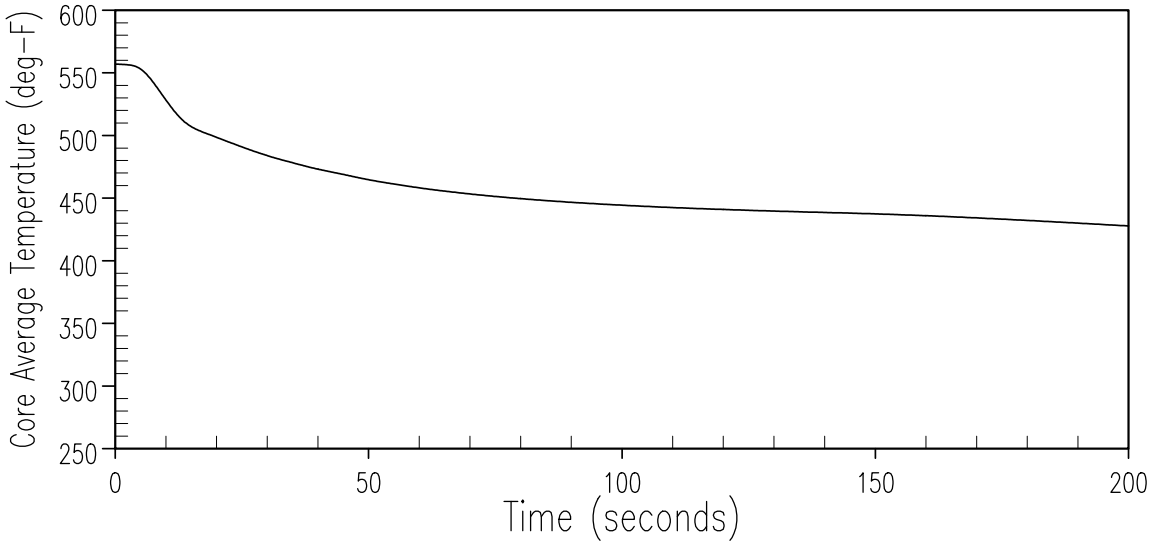

Figure III.4-2 Hot Zero Power Steamline Break – Average Temperature vs. Time

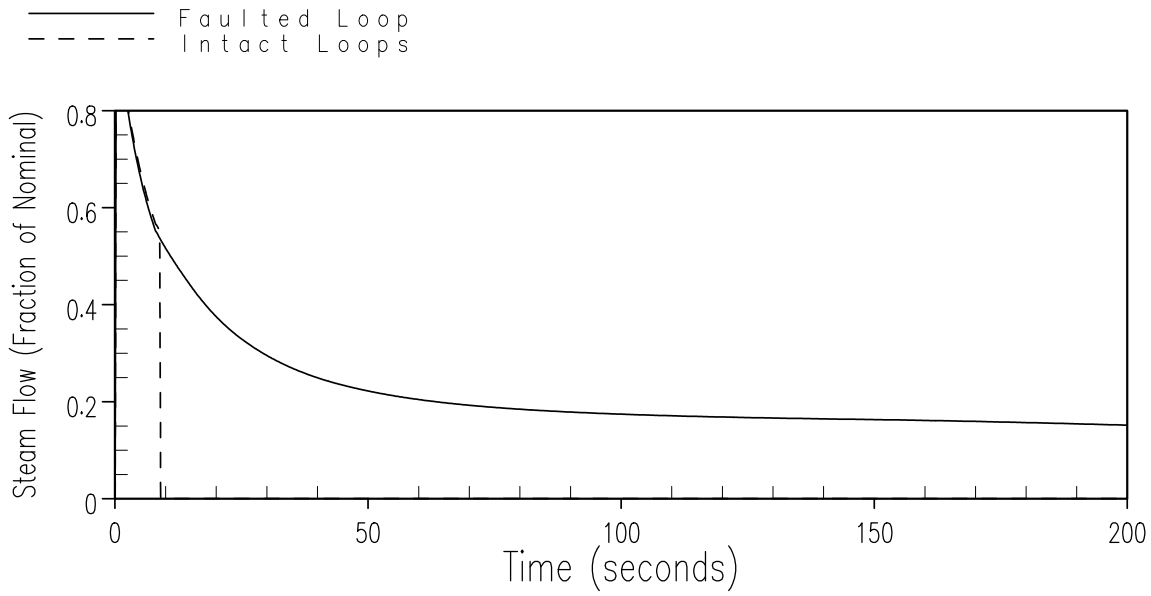

Figure III.4-3 Hot Zero Power Steamline Break – Steam Flow vs. Time

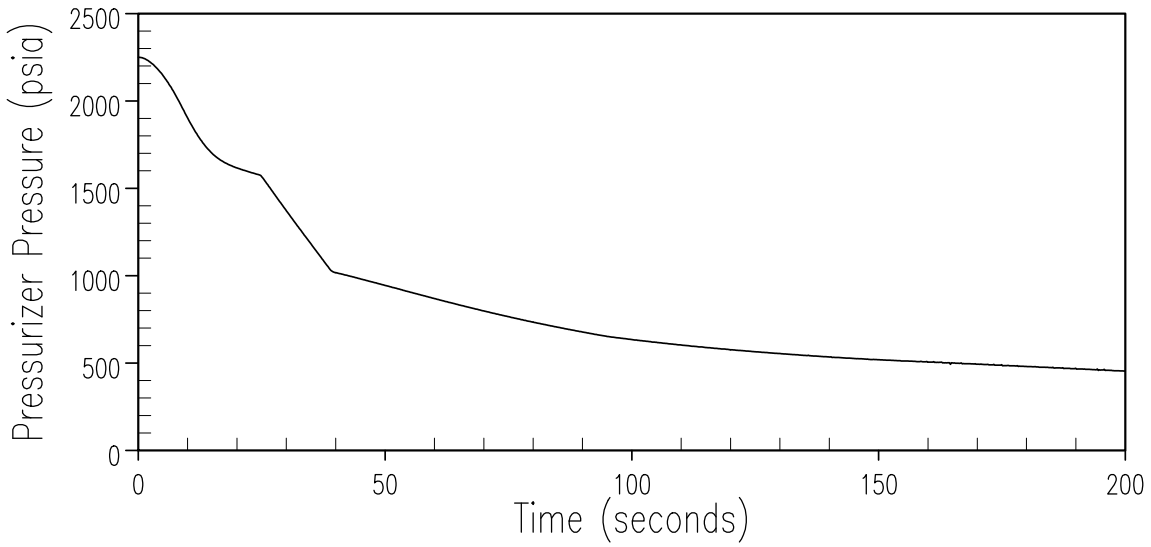

Figure III.4-4 Hot Zero Power Steamline Break – Pressurizer Pressure vs. Time

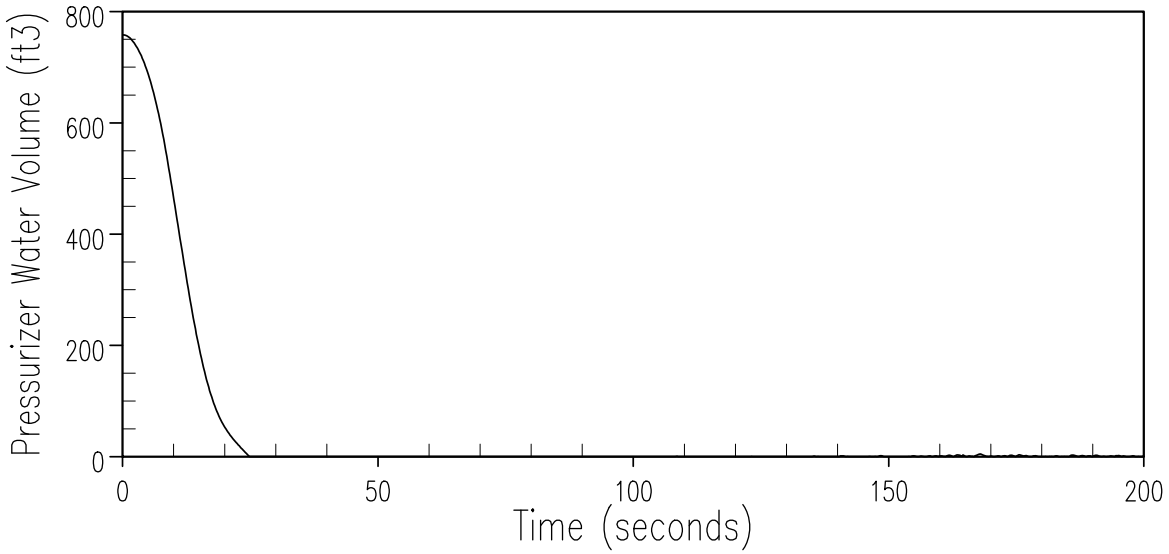

Figure III.4-5 Hot Zero Power Steamline Break – Pressurizer Water Volume vs. Time

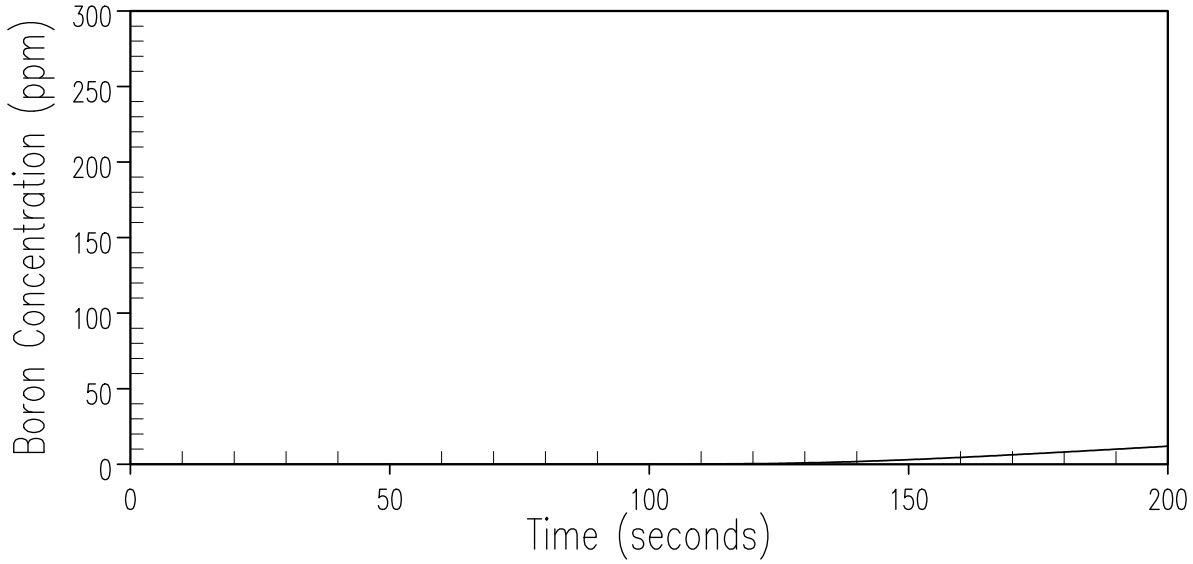

Figure III.4-6 Hot Zero Power Steamline Break – Boron Concentration vs. Time

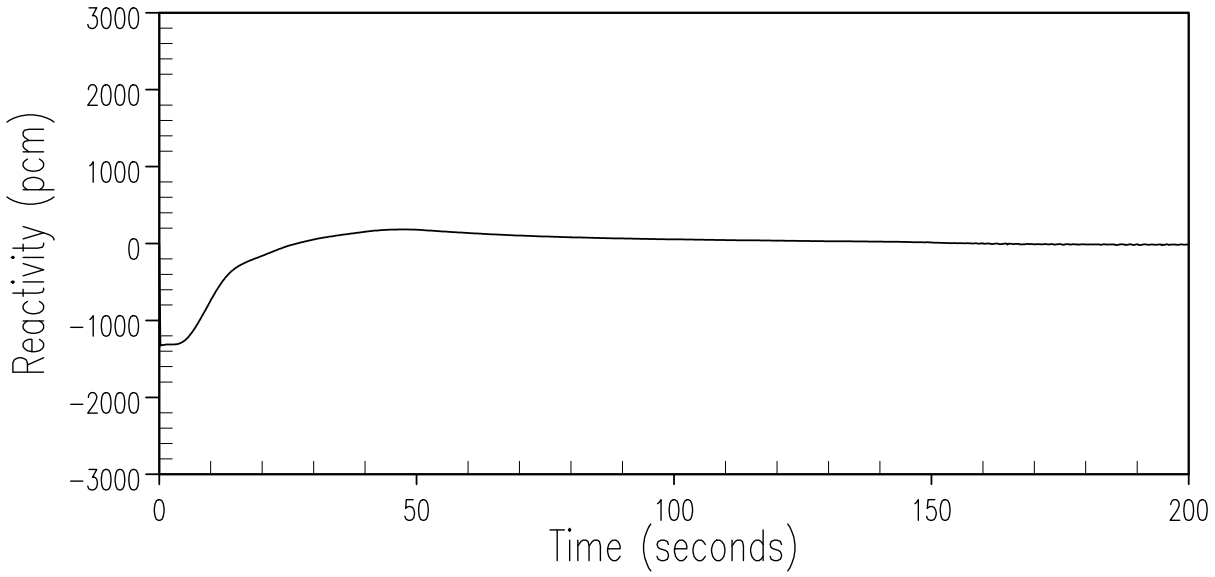

Figure III.4-7 Hot Zero Power Steamline Break – Reactivity vs. Time

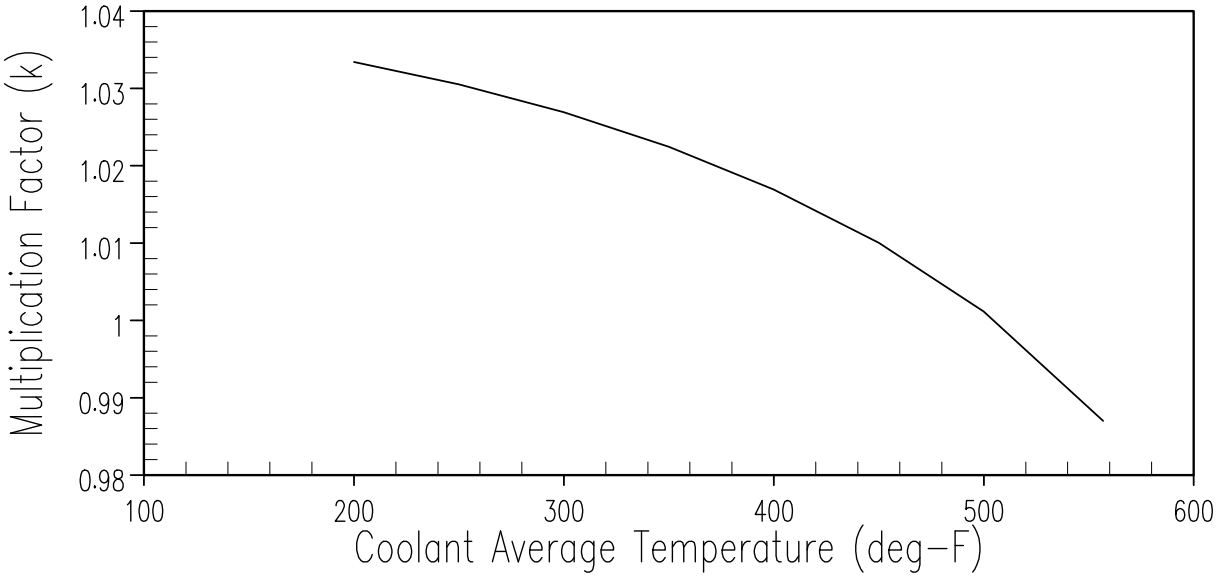

Figure III.4-8 Hot Zero Power Steamline Break – K_{eff} vs. Coolant Average Temperature

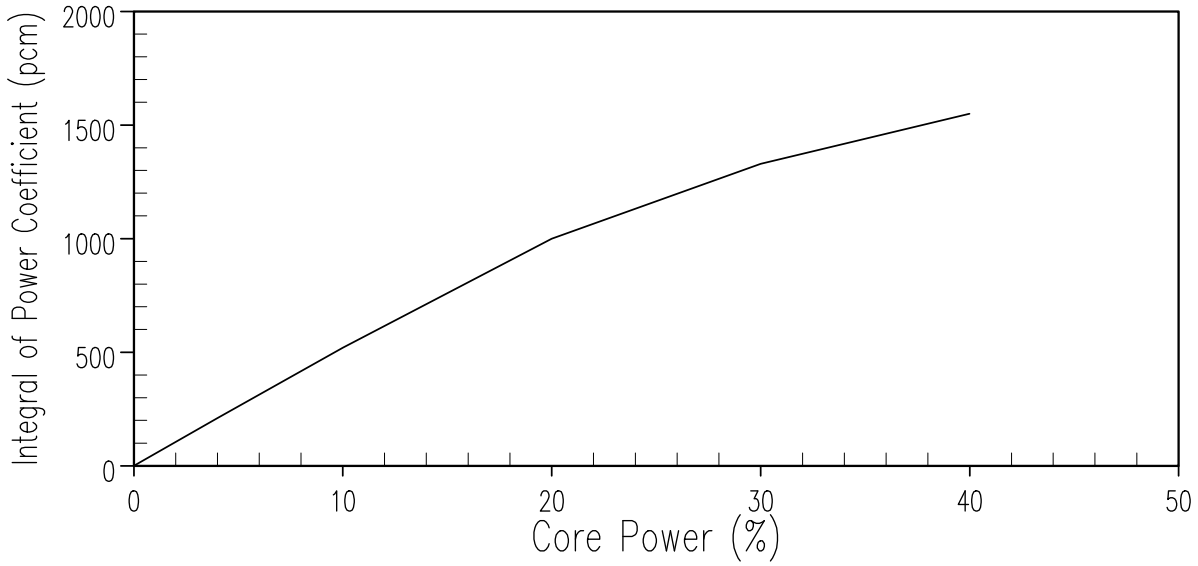

Figure III.4-9 Hot Zero Power Steamline Break – Doppler Power Feedback

III.5 Steam System Piping Failure at Full Power – UFSAR 15.1.6

III.5.1 Identification of Causes and Accident Description

The steam release arising from a break of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity and thus a power excursion is plausible. Analysis of a steam system piping failure occurring from at-power initial conditions is performed to demonstrate that core protection is maintained prior to and immediately following reactor trip. The post-trip concerns of a steam system piping failure are described in Section III.4.1 (HZP-SLB).

III.5.2 Method of Analysis

The full power steam line break analysis for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN (Reference III.5-1), VIPRE (Reference III.5-2), and ANC (Reference III.5-3) computer codes along with the Revised Thermal Design Procedure (RTDP) methodology (Reference III.5-4) to calculate the minimum departure from nucleate boiling ratio (DNBR) and peak linear heat rate (PLHR) values. The transient response of the current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) Power Uprate.

III.5.3 Analysis Inputs and Assumptions

Cases for both Byron Unit 1 and Braidwood Unit 1 with Babcock and Wilcox International (BWI) steam generators (SGs) and Byron Unit 2 and Braidwood Unit 2 with Westinghouse Model D5 SGs were analyzed at conditions designed to provide the most conservative and limiting results. Specifically, the following assumptions are made:

a. The initial core power, reactor coolant temperature, and reactor coolant system pressure were assumed to be at their nominal full-power values at uprated power conditions. Cases assuming full power operation at the high (588°F) hot full power (HFP) T_{avg} condition are considered. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.5-4.

b. The limiting break size was calculated to be 0.95 ft^2 for Units 1. The results for this case bound all other break sizes for both Units 1 and Units 2.

c. In computing the steam flow during a steam line break, the Moody curve for f(L/D) = 0 is used.

d. The analysis assumed maximum moderator reactivity feedback and least negative Doppler power feedback to maximize the power increase following the break.

e. This analysis only considers the initial phase of the transient from at-power conditions. Protection in this phase of the transient is provided by reactor trip, if necessary. The power range high neutron flux, safety injection, low pressurizer pressure and overpower ΔT reactor trip functions are credited as being available to mitigate the effects of this event.

III.5.4 Analysis Acceptance Criteria

Depending on the size of the break, a rupture to the main steam line is classified as either a Condition III (infrequent fault) or Condition IV (limiting fault) event. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable for a Condition III or Condition IV event, Westinghouse conservatively applies the Condition II acceptance criteria to the event. Specifically, this analysis shows that the DNBR and PLHR values are met such that damage to the fuel rods is precluded.

III.5.5 Analysis Results

The most limiting case (BWI SGs with a break size of 0.95 ft^2) for a rupture in a main steam line is shown on Figures III.5-1 through III.5-6. Figure III.5-1 illustrates the nuclear power transient following a steam line break. Nuclear power increases due to the presence of a negative moderator temperature coefficient until the reactor trips on overpower ΔT. Figure III.5-2 gives the heat flux, which follows a similar trend as nuclear power since the two are directly related. The core average temperature transient and pressurizer water volume transient following the accident are given in Figures III.5-3 and III.5-4. Core average temperature drops slowly due to the increased heat transfer to the secondary side and then drops rapidly after reactor trip. The pressurizer water volume follows a similar trend as the average core temperature since the two are related through density. Figure III.5-5 displays the pressurizer pressure which slowly decreases at first due to a gradual decrease in pressurizer water volume and then drops rapidly after reactor trip. The small increase in pressure at roughly 18 seconds is due to the closure of the turbine stop valves. Figure III.5-6 shows the steam pressure for both the intact and the faulted SGs. The steam pressure increases after reactor trip due to the closure of the turbine stop valves.

The calculated minimum DNBR value for the MUR power uprate is [ ]^{a,c} compared to a DNBR safety analysis limit of [ ]^{a,c}. The calculated maximum PLHR value for the MUR power uprate is 22.23 kW/ft compared to a PLHR safety analysis limit of 22.3 kW/ft. Therefore, all applicable acceptance criteria are met with respect to the pre-trip and immediately following reactor trip concerns of the rupture of a main steam line event at MUR power uprate conditions. The sequence of events for the accident is shown in Table III.5-1.

A comparison of the results from the limiting MUR Uprate case to the limiting current licensing basis case is presented in Table III.5-2. From Table III.5-2, it can be seen that, the minimum DNBR has decreased and the PLHR has increased. These results are expected for a power uprate.

III.5.6 References

III.5-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.5-2 WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

III.5-3 WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” September 1986.

III.5-4 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

Table III.5-1 Sequence of Events

Event	Time (seconds)
Steam Line Rupture	0.00
OPΔT Reactor Trip Setpoint Reached	8.46
Rods Begin to Drop	16.46
Minimum DNBR Occurs	17.00
Maximum Core Heat Flux Occurs	17.10

Table III.5-2 Comparison of Analysis Results to the Current Licensing Basis

Limiting Case Description	DNBR Safety Analysis Limit	Calculated Minimum DNBR	PLHR Safety Analysis Limit (kW/ft)	Calculated Maximum PLHR (kW/ft)
Limiting Licensing Basis Case: HFP, BWI SGs, Hi-T_{avg}, Uniform Flow, 0.968 ft^2 Break	[ ]a,c	[ ]a,c	22.4	22.16
Limiting MUR Case: HFP, BWI SGs, Hi-T_{avg}, Uniform Flow, 0.95 ft^2 Break	[ ]a,c	[ ]a,c	22.3	22.23

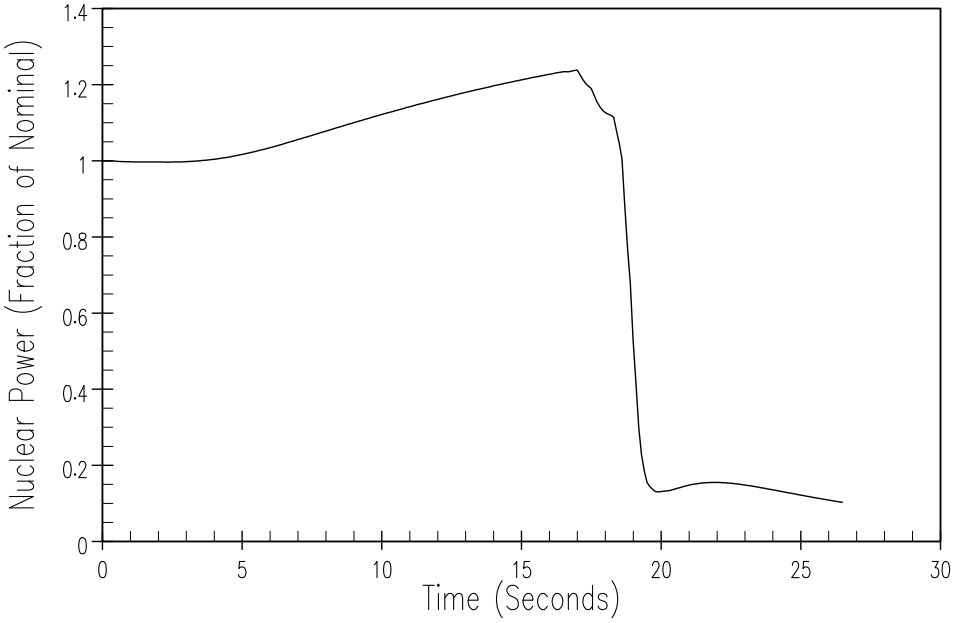

Figure III.5-1 Nuclear Power versus Time

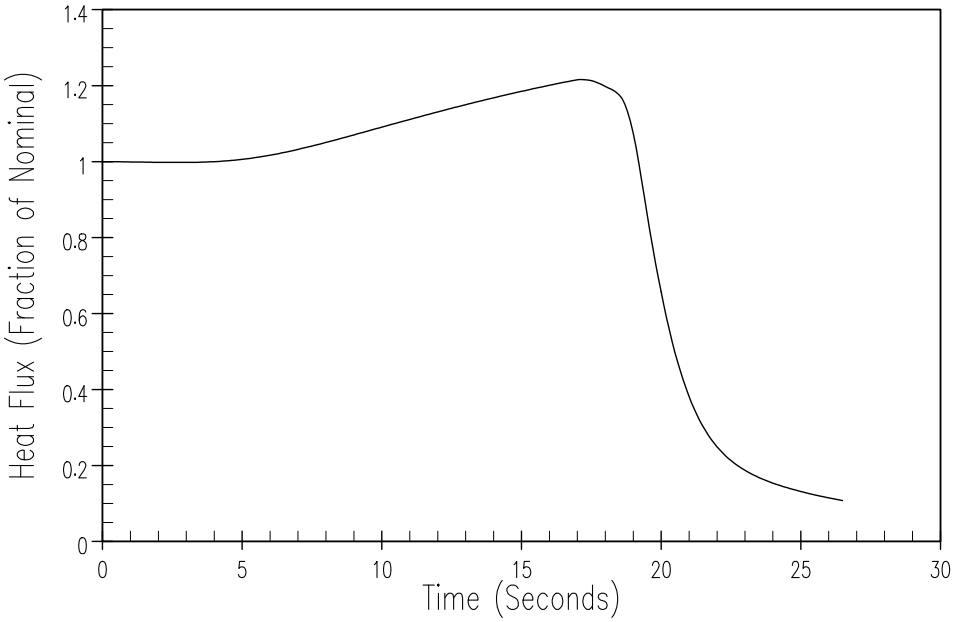

Figure III.5-2 Heat Flux versus Time

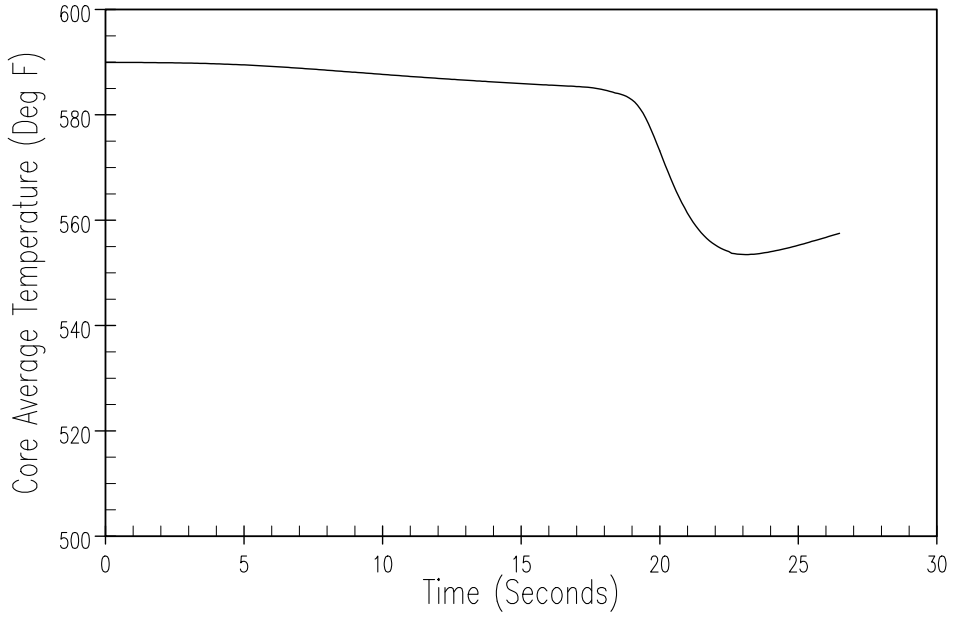

Figure III.5-3 Core Average Temperature versus Time

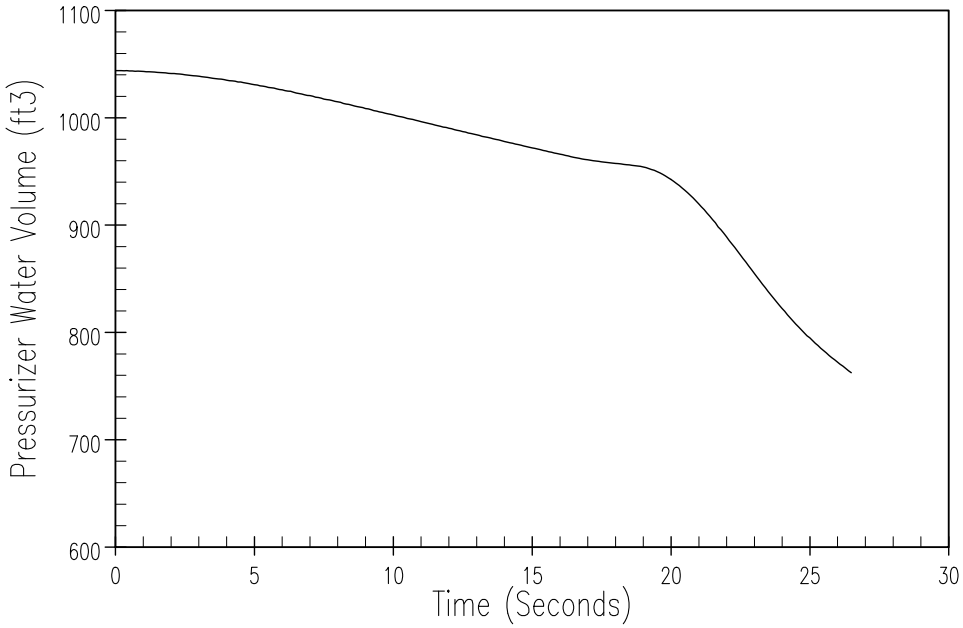

Figure III.5-4 Pressurizer Water Volume versus Time

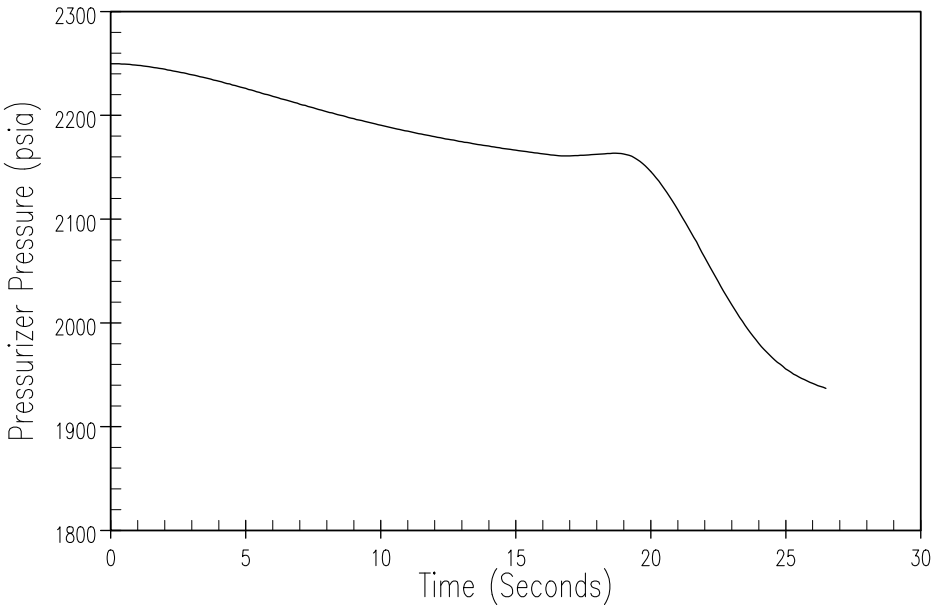

Figure III.5-5 Pressurizer Pressure versus Time

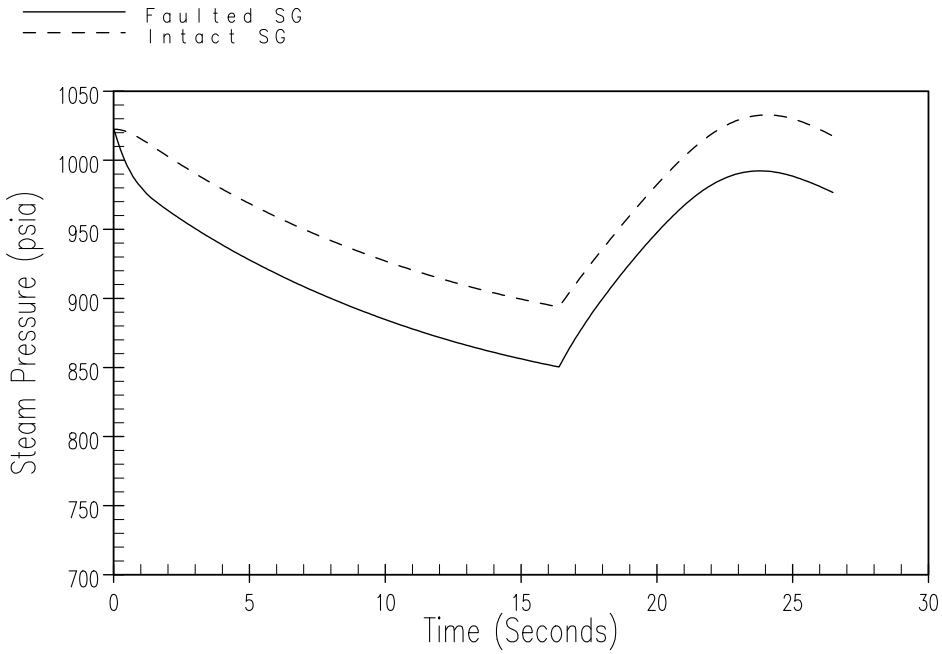

Figure III.5-6 Steam Pressure versus Time

III.6 Loss of External Load / Turbine Trip / Inadvertent Closure of Main Steam Isolation Valves / Loss of Condenser Vacuum and Other Events Causing a Turbine Trip – UFSAR 15.2.2 through 15.2.5

III.6.1 Identification of Causes and Accident Description

A turbine trip event is bounding for loss of external load, loss of condenser vacuum, inadvertent closure of main steam isolation valves and other turbine trip events. As such, this event is analyzed in detail. For a turbine trip event, the turbine stop valves close rapidly (typically 0.1 sec.) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Turbine trip initiation signals include:

a. low condenser vacuum,

b. low bearing oil pressure,

c. turbine thrust bearing failure,

d. turbine overspeed,

e. manual trip,

f. low emergency trip header pressure, and

g. loss of both redundant controllers.

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate steam dump. The loss of steam flow results in an almost immediate rise in secondary system temperature. For a turbine trip, the reactor would be tripped directly (unless below approximately 30% (P-8) power) on a signal from the turbine stop valves.

The automatic steam dump system would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser was not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the auxiliary feedwater system to ensure adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the steam generator safety valves may lift to provide pressure control.

III.6.2 Method of Analysis

Multiple cases are analyzed to address specific acceptance criteria; specifically, minimum departure from nucleate boiling ratio (DNBR) and maximum reactor coolant system (RCS) and main steam system (MSS) pressure.

For overpressure concerns, the Standard Thermal Design Procedure (STDP) methodology is used, where the uncertainties on the initial conditions (i.e., power, temperature, pressure, and flow) are explicitly modeled. Therefore, the current licensing basis RCS overpressure analysis is unaffected by the tradeoff between the increased power level and decreased uncertainty, and thus, is not impacted by the MUR power uprate. An explicit main steam system overpressure analysis designed to maximize the steam generator pressure was performed as part of the MUR power uprate. The analysis is based on the RCS

overpressure case except that automatic pressure control is assumed operable and minimum steam generator tube plugging is modeled. The analysis is performed with the NRC-approved LOFTRAN computer code (Reference III.6-1).

The DNB case is also reanalyzed for the MUR power uprate. The analysis uses the NRC-approved LOFTRAN computer code (Reference III.6-1) and Revised Thermal Design Procedure (RTDP) methodology (Reference III.6-2) to calculate a minimum DNBR. The current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the MUR power uprate.

In order to bound all of the turbine trip transients, the behavior of the unit is evaluated for a complete loss of steam load from full power primarily to show the adequacy of the pressure relieving devices and also to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip (i.e., no reactor trip on turbine trip). No credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

III.6.3 Analysis Inputs and Assumptions

Cases for both Byron Unit 1 and Braidwood Unit 1 with Babcock and Wilcox International (BWI) steam generators and Byron Unit 2 and Braidwood Unit 2 with Westinghouse Model D5 steam generators were analyzed at conditions designed to provide the most conservative and limiting results. Specifically, the following assumptions are made:

a. Minimum reactivity feedback is modeled. The analysis is performed at full power conditions with a moderator temperature coefficient of 0 pcm/^{o}F and the least negative Doppler-only power and Doppler temperature coefficients. These conditions are bounding for all operating conditions anticipated throughout each cycle.

b. Manual rod control is conservatively modeled for the event. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

c. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

d. No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. When the steam generator pressure rises to the safety valve setpoint, the steam release through the safety valves limits secondary steam pressure.

e. Main feedwater flow to the steam generators is lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation would normally occur. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

f. Reactor trip is actuated by the first reactor protection system trip setpoint reached. Trip signals are expected due to high pressurizer pressure and overtemperature ΔT (OT ΔT).

g. The analysis models the operability of all main steam safety valves (MSSVs) with setpoint tolerance greater than or equal to the Technical Specification limit of 3%.

Additional major assumptions for the main steam system overpressure case include the following:

- Initial reactor power, pressure, and RCS temperature (consistent with the MUR uprated power conditions) include uncertainties, as applicable. The uncertainty on initial reactor power is included in the nominal power analyzed. The nominal full power RCS temperature plus uncertainties, including the RCS average temperature bias, is modeled. The initial RCS pressure is assumed to be at its nominal value minus uncertainties. The RCS flow rate corresponding to thermal design flow is also modeled. Minimum steam generator tube plugging is assumed.

Additional major assumptions for the DNB case include the following:

- Initial reactor power, pressure, and RCS temperature (consistent with the MUR uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.6-2. The RCS flow rate corresponding to minimum measured flow is also modeled. Minimum steam generator tube plugging is assumed.

III.6.4 Analysis Acceptance Criteria

A loss of external load/turbine trip is classified as an ANS Condition II event, a fault of moderate frequency. The criteria of interest for the LOL/TT transient are as follows:

1. The pressure in the reactor coolant system and main steam system shall be maintained below 110% of the design value.
2. The fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs.
3. An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently. This criterion is met by ensuring that the pressurizer does not reach a water solid condition.
4. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding. This criterion is met by demonstrating that the DNB design basis is satisfied.

III.6.5 Analysis Results

The results for the loss of load/turbine trip (LOL/TT) DNB analysis performed to support the Byron and Braidwood MUR power uprate are provided in Table III.6-5. The sequence of events for the Units 1 DNB and main steam system overpressure analyses are provided in Tables III.6-1 and III.6-2, respectively. The sequence of events for the Units 2 DNB and MSS overpressure analyses are provided in Tables III.6-3 and III.6-4, respectively. In all cases analyzed, the acceptance criteria for this event have been met.

The LOL/TT DNB case results for Byron and Braidwood Units 1 are shown in Figures III.6-1 through III.6-5. Figure III.6-5 illustrates that the DNB remains above the limit of [ ]^{a,c} throughout the transient. Nuclear power is maintained at the initial value until reactor trip occurs on OTΔT, shown in Figure III.6-1. The Units 1 MSS overpressure results are shown in Figures III.6-6 through III.6-10. Figure III.6-10 confirms that the MSS pressure remains below the overpressure limit value of 1318.5 psia.

The LOL/TT trip DNB case results for Byron and Braidwood Units 2 are shown in Figures III.6-11 through III.6-15. Figure III.6-15 illustrates that the DNB remains above the limit of [ ]^{a,c} throughout the transient. Nuclear power is maintained at the initial value until reactor trip occurs on OTΔT, shown in Figure III.6-11. The Units 2 MSS overpressure results are shown in Figures III.6-16 through III.6-20. Figure III.6.20 confirms that the MSS pressure remains below the overpressure limit value of 1318.5 psia.

As discussed previously, the RCS overpressure cases are not impacted by the MUR power uprate. Therefore, the pressurizer safety valves and main steam safety valves continue to maintain RCS and MSS pressure below 110% of the respective design pressure limits.

A comparison of the results from the cases analyzed for the MUR power uprate to those from the current licensing basis analysis is presented in Table III.6-5. From this comparison it can be seen that the increase in power associated with the MUR power uprate causes the minimum DNBR to decrease, as expected. However, the minimum DNBR for both units remains above the Safety Analysis Limit. The comparison also confirms that the MSS overpressure limit continues to be met assuming an MSSV setpoint tolerance greater than or equal to the Technical Specification limit of 3%.

III.6.6 References

III.6-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.6-2 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

Table III.6-1 LOL/TT Sequence of Events for Unit 1 DNB Case

Event	Time (Seconds)
Turbine trip/Loss of main feedwater flow	0.0
Overtemperature ΔT reactor trip setpoint reached	5.4
Initiation of steam release from SG safety valves	6.0
Rods begin to drop	13.4
Minimum DNBR Occurs	14.1

Table III.6-2 LOL/TT Sequence of Events for Unit 1 MSS Overpressure Case

Event	Time (Seconds)
Turbine trip/ Loss of main feedwater flow	0.0
Initiation of steam release from SG safety valves	3.4
Overtemperature ΔT reactor trip setpoint reached	3.7
Rods begin to drop	11.7
Maximum SG Pressure occurs	15.6

Table III.6-3 LOL/TT Sequence of Events for Unit 2 DNB Case

Event	Time (Seconds)
Turbine trip/Loss of main feedwater flow	0.0
Overtemperature ΔT reactor trip setpoint reached	3.5
Initiation of steam release from SG safety valves	7.9
Rods begin to drop	11.5
Minimum DNBR Occurs	12.6

Table III.6-4 LOL/TT Sequence of Events for Unit 2 MSS Overpressure Case

Event	Time (Seconds)
Turbine trip/ Loss of main feedwater flow	0.0
Overtemperature ΔT reactor trip setpoint reached	1.8
Initiation of steam release from SG safety valves	4.8
Rods begin to drop	9.8
Maximum SG Pressure occurs	15.1

Table III.6-5 Comparison of Results to the Current Licensing Basis

Unit/SG Model	DNBR Safety Analysis Limit	Minimum DNBR	SG Pressure Limit (psia)	Peak SG Pressure (psia)
MUR 1/BWI SG	[ ]a,c	[ ]a,c	1318.5	1313.5
MUR 2/D5 SG	[ ]a,c	[ ]a,c	1318.5	1310.6
Licensing Basis 1/BWI SG	[ ]a,c	[ ]a,c	1318.5	1317.7
Licensing Basis 2/D5 SG	[ ]a,c	[ ]a,c	1318.5	1310.0

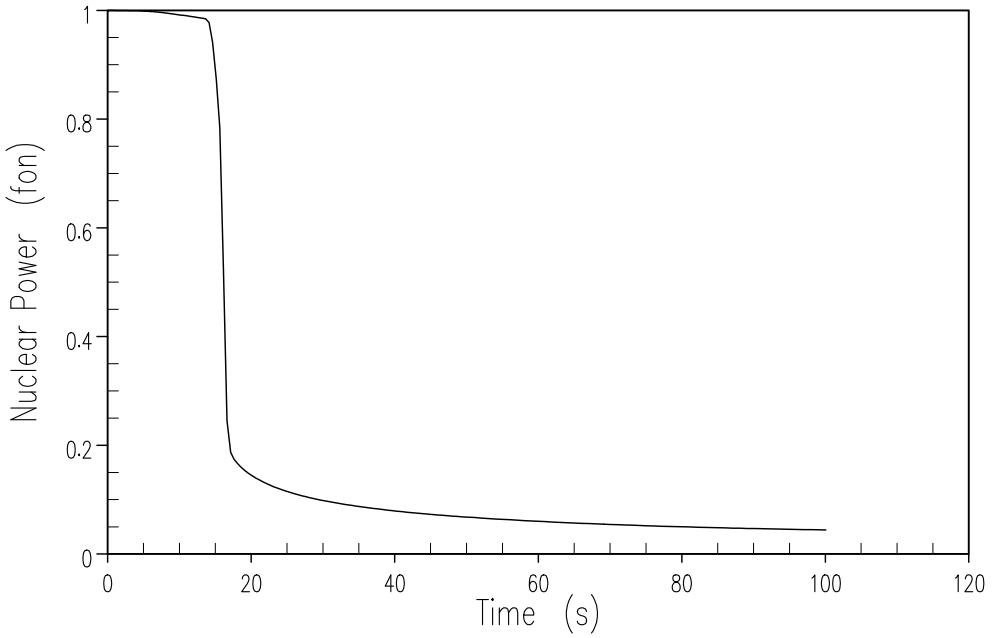

Figure III.6-1 Unit 1, Loss of Load/Turbine Trip, DNB Case, Nuclear Power

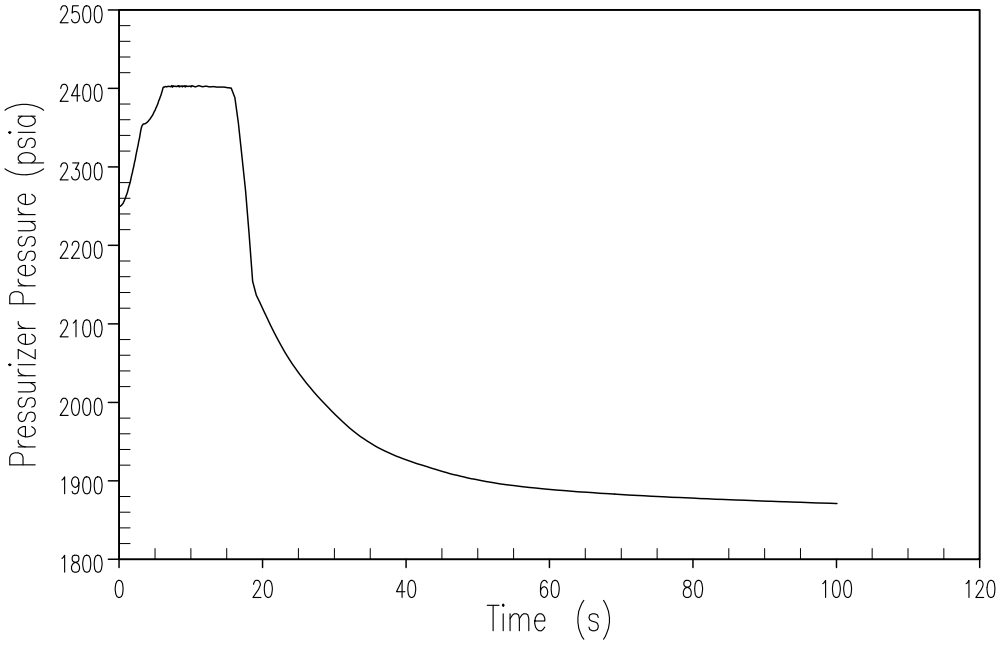

Figure III.6-2 Unit 1, Loss of Load/Turbine Trip, DNB Case, Pressurizer Pressure

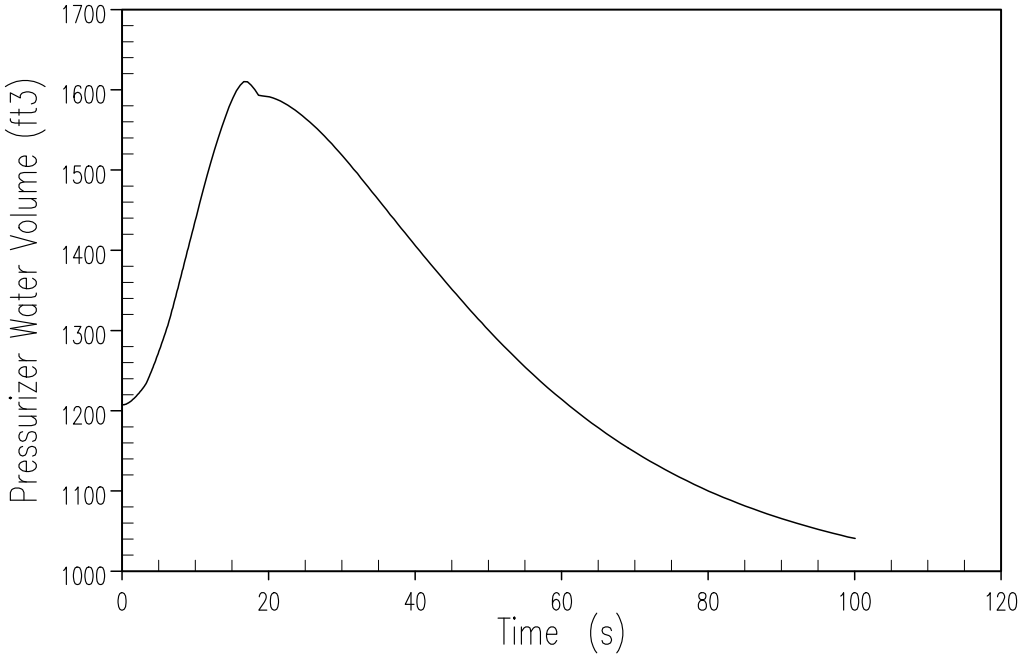

Figure III.6-3 Unit 1, Loss of Load/Turbine Trip, DNB Case, Pressurizer Water Volume

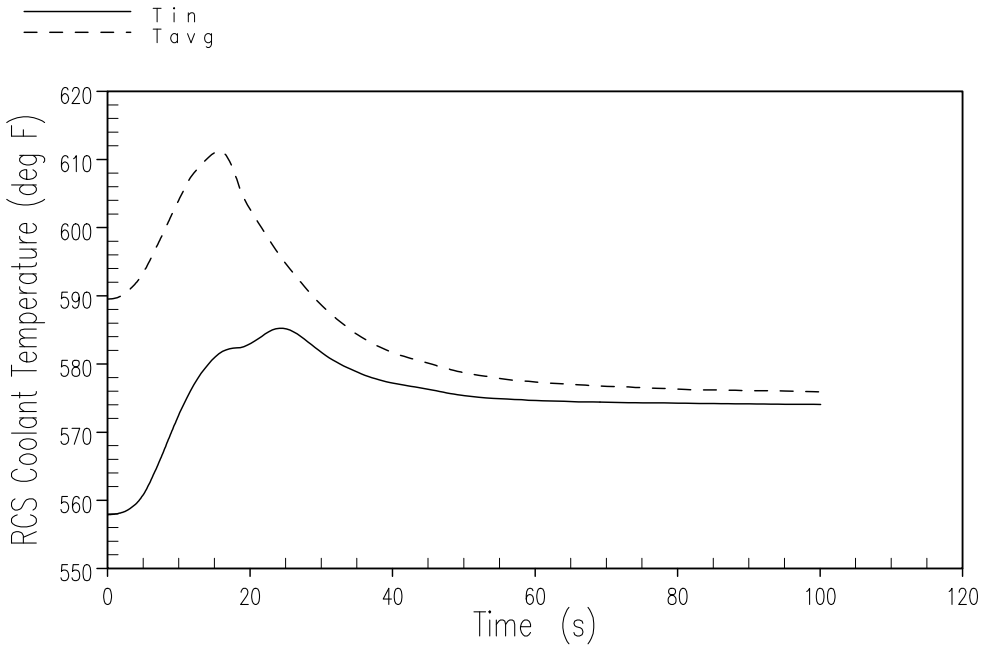

Figure III.6-4 Unit 1, Loss of Load/Turbine Trip, DNB Case, RCS Coolant Temperature

a,c

Figure III.6-5 Unit 1, Loss of Load/Turbine Trip, DNB Case, DNBR

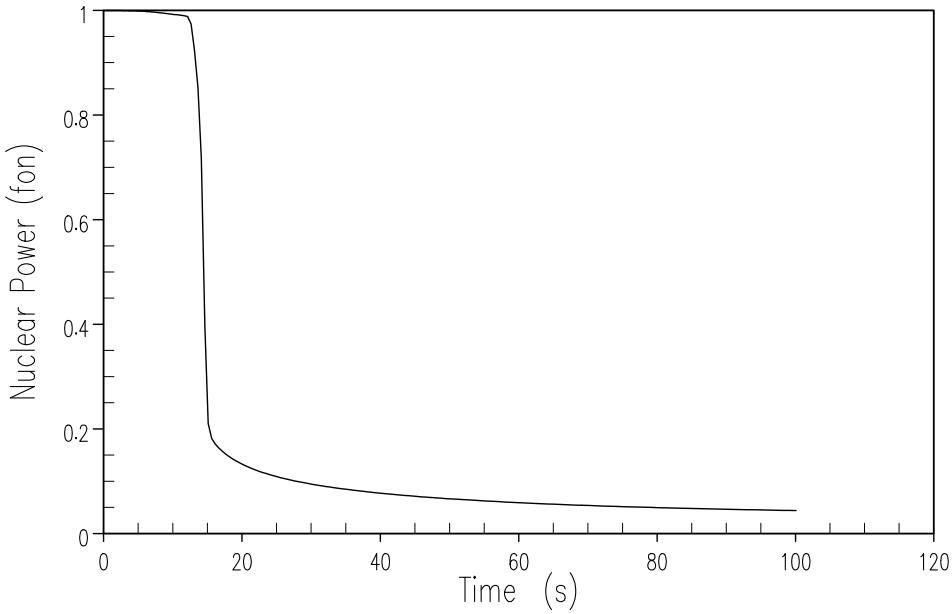

Figure III.6-6 Unit 1, Loss of Load/Turbine Trip, MSS Overpressure Case, Nuclear Power

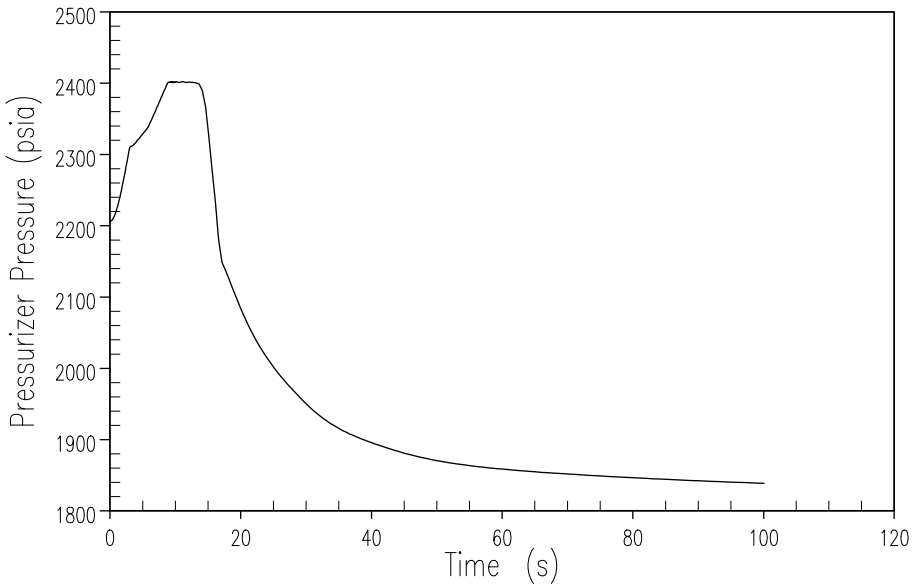

Figure III.6-7 Unit 1, Loss of Load/Turbine Trip, MSS Overpressure Case, Pressurizer Pressure

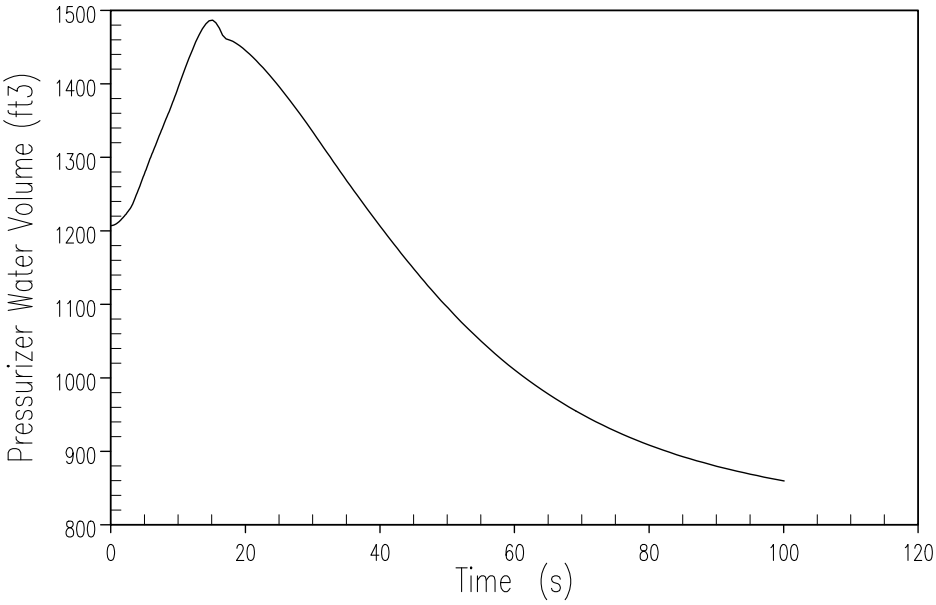

Figure III.6-8 Unit 1, Loss of Load/Turbine Trip, MSS Overpressure Case, Pressurizer Water Volume

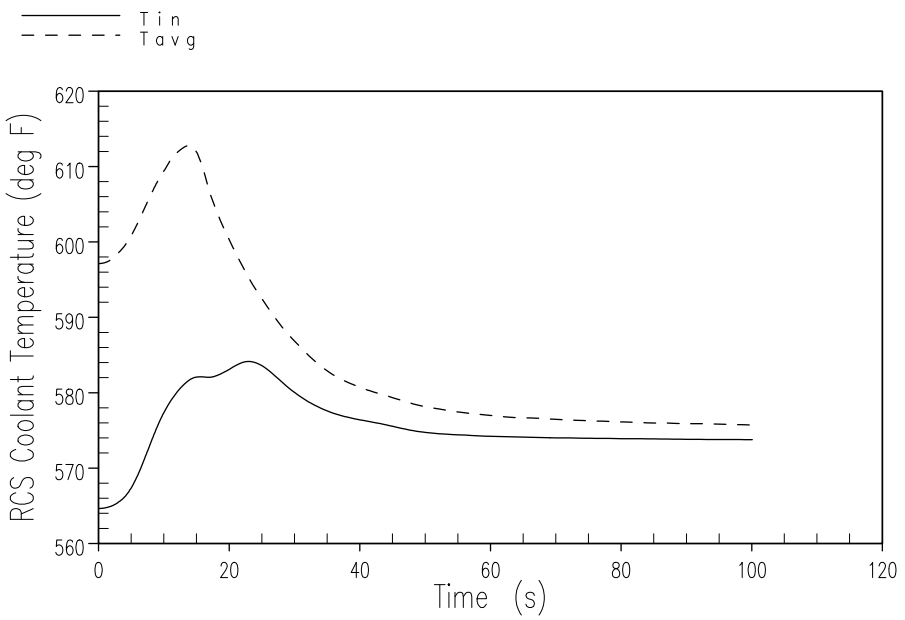

Figure III.6-9 Unit 1, Loss of Load/Turbine Trip, MSS Overpressure Case, RCS Coolant Temperature

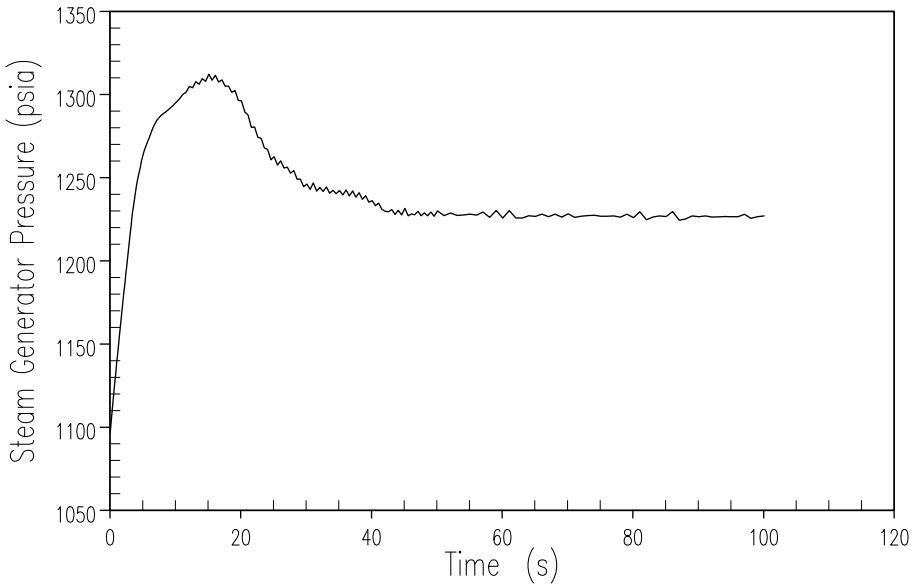

Figure III.6-10 Unit 1, Loss of Load/Turbine Trip, MSS Overpressure Case, Steam Generator Pressure

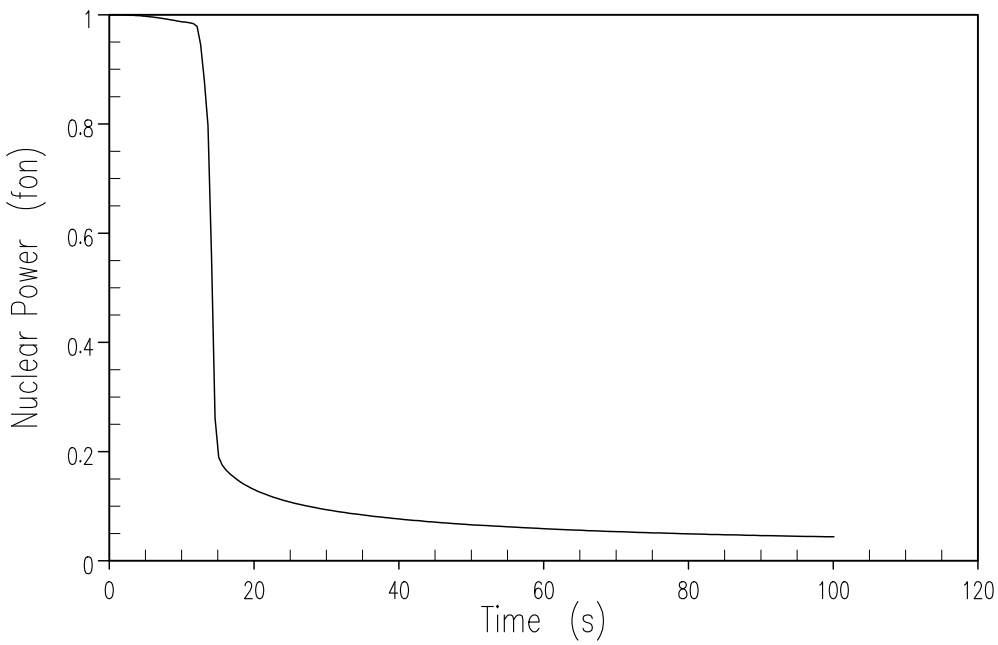

Figure III.6-11 Unit 2, Loss of Load/Turbine Trip, DNB Case, Nuclear Power

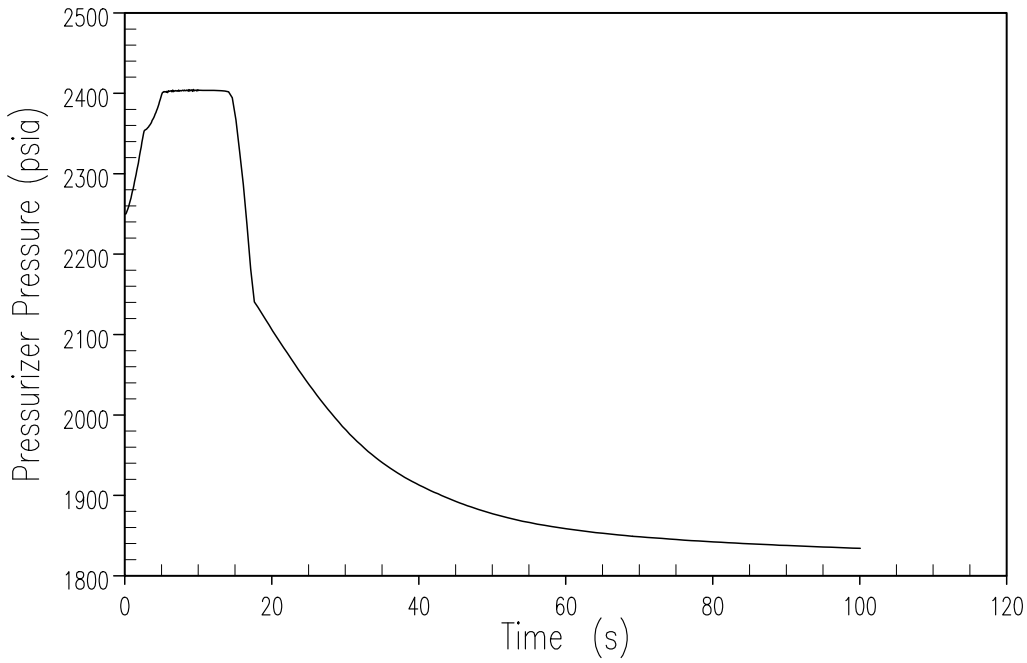

Figure III.6-12 Unit 2, Loss of Load/Turbine Trip, DNB Case, Pressurizer Pressure

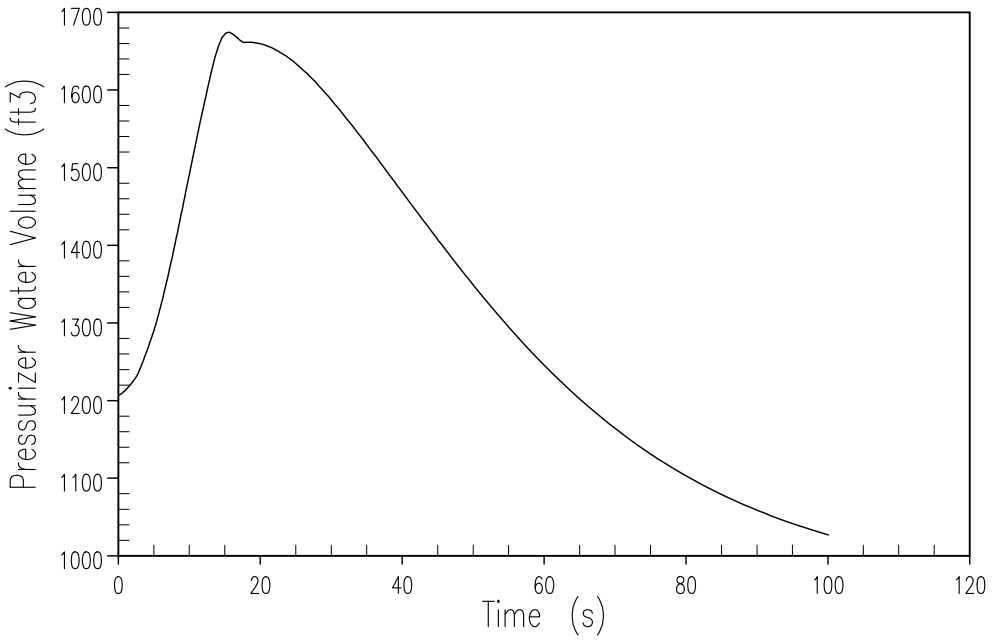

Figure III.6-13 Unit 2, Loss of Load/Turbine Trip, DNB Case, Pressurizer Water Volume

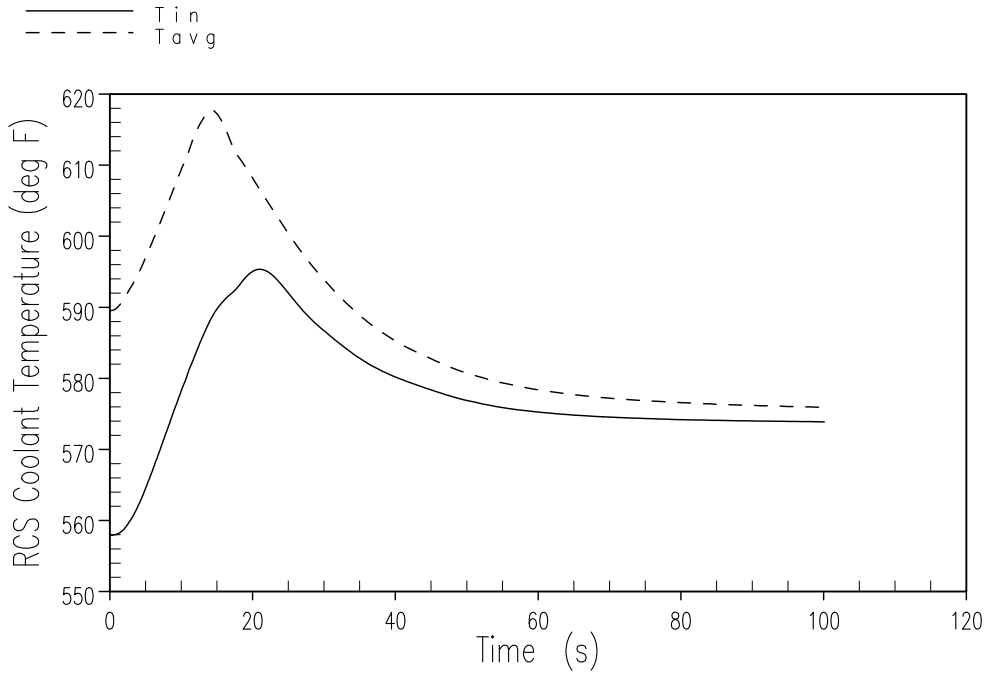

Figure III.6-14 Unit 2, Loss of Load/Turbine Trip, DNB Case, RCS Coolant Temperature

a,c

Figure III.6-15 Unit 2, Loss of Load/Turbine Trip, DNB Case, DNBR

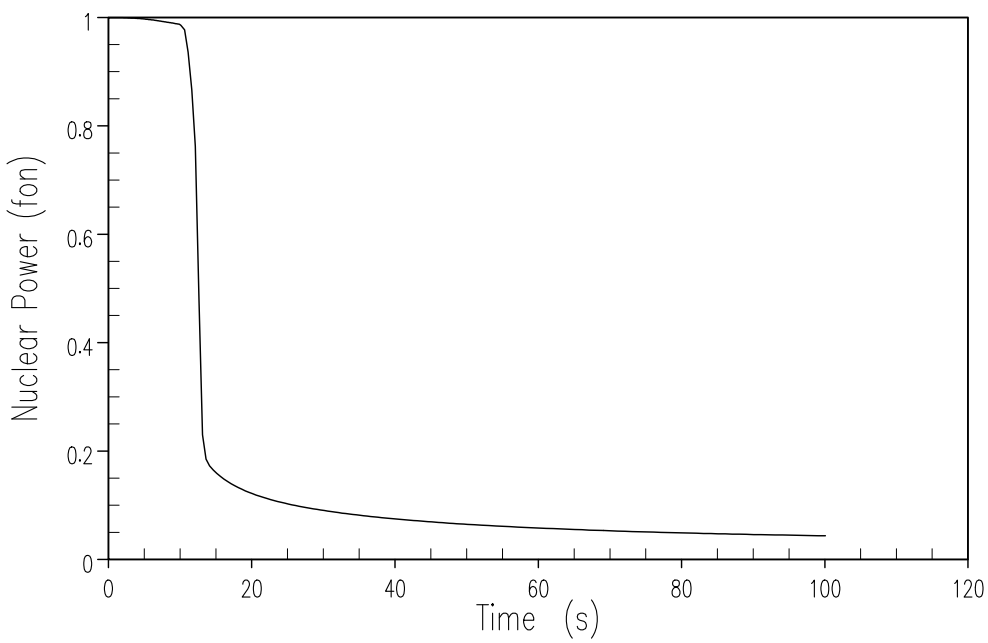

Figure III.6-16 Unit 2, Loss of Load/Turbine Trip, MSS Overpressure Case, Nuclear Power

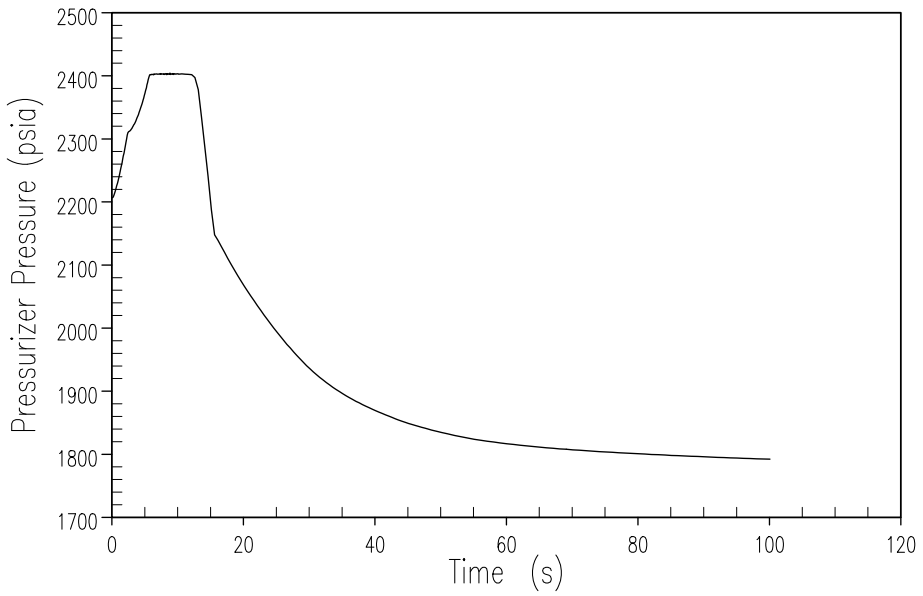

Figure III.6-17 Unit 2, Loss of Load/Turbine Trip, MSS Overpressure Case, Pressurizer Pressure

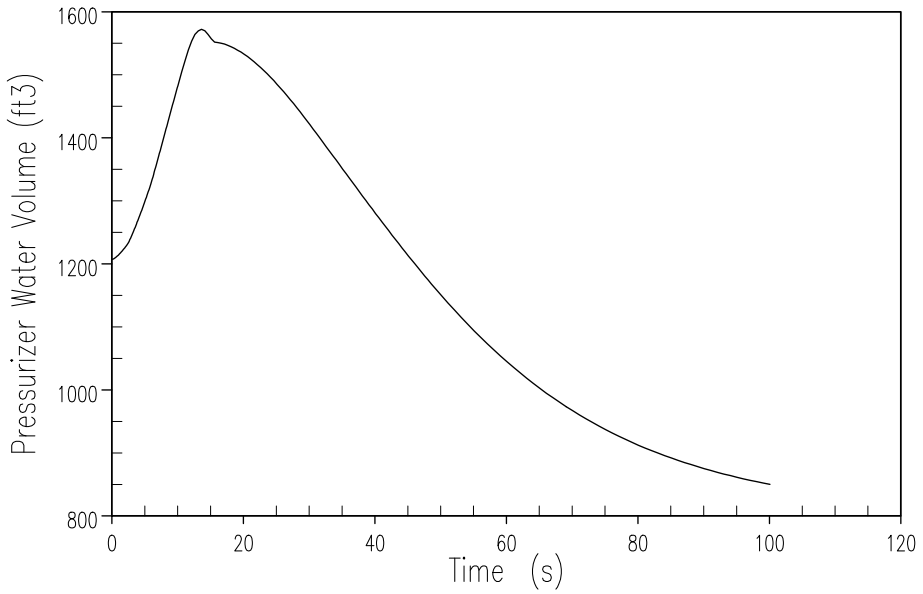

Figure III.6-18 Unit 2, Loss of Load/Turbine Trip, MSS Overpressure Case, Pressurizer Water Volume

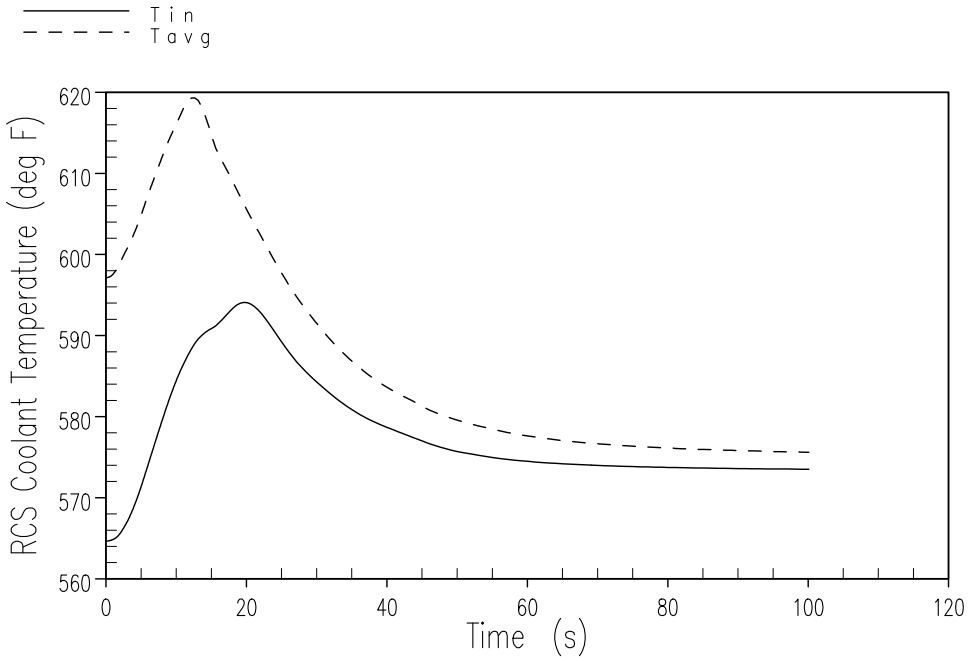

Figure III.6-19 Unit 2, Loss of Load/Turbine Trip, MSS Overpressure Case, RCS Coolant Temperature

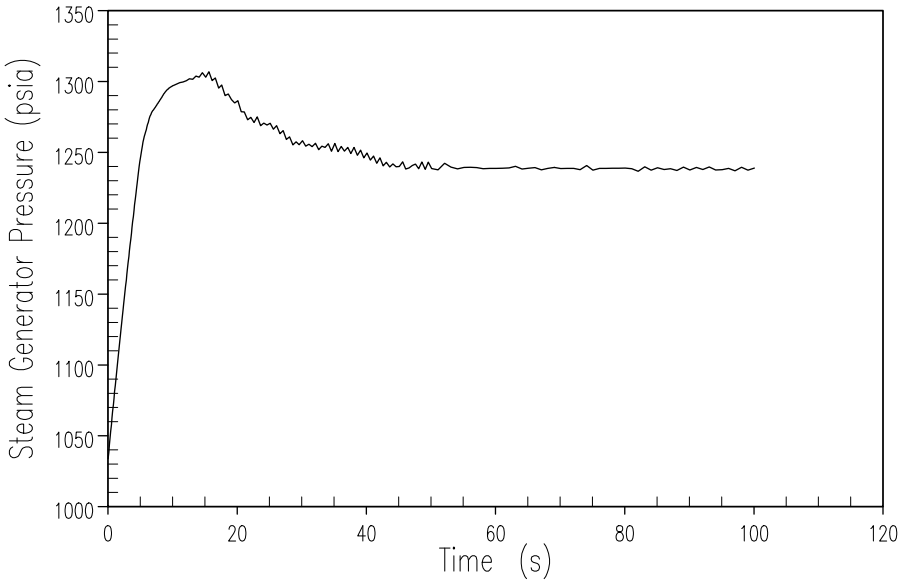

Figure III.6-20 Unit 2, Loss of Load/Turbine Trip, MSS Overpressure Case, Steam Generator Pressure

III.7 Partial Loss of Forced Reactor Coolant Flow – UFSAR 15.3.1

III.7.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump (RCP), or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

III.7.2 Method of Analysis

The partial loss of flow analysis for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN computer code (Reference III.7-1) and VIPRE computer code (Reference III.7-2) along with the Revised Thermal Design Procedure (RTDP) methodology (Reference III.7-3) to calculate a minimum departure from nucleate boiling ratio (DNBR). The partial loss of flow is analyzed as a loss of two reactor coolant pumps with four loops in operation. The transient response of the current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) power uprate.

III.7.3 Analysis Inputs and Assumptions

a. Initial reactor power, pressure, and RCS temperatures (consistent with the MUR uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.7-3.

b. The flow coastdown is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics and is based on high estimates of system pressure losses.

c. The most-negative Doppler-only power coefficient is modeled since it maximizes the positive reactivity addition during the trip (which acts to retard the power decrease).

d. A full power moderator temperature coefficient of 0 pcm/°F corresponding to minimum feedback is assumed. This assumption is valid and appropriate even for a plant employing a positive moderator temperature coefficient (PMTC). Typically, a PMTC is allowed by the plant Technical Specifications from 0% to some part power level (in this case 70%), at which point the PMTC linearly ramps to zero (see Technical Specifications Figure 3.1.3-1). The full power analysis results using an MTC of 0 pcm/°F bound those for part-power initial conditions with a PMTC at the licensed allowable MTC limit.

e. The Low Reactor Coolant System Flow reactor trip function is credited as being available to mitigate the effects of this event.

f. The most limiting single failure for a partial loss of flow event is the failure of a protection train. No single active failure will prevent the reactor protection system from functioning properly.

III.7.4 Analysis Acceptance Criteria

The partial loss of flow event is classified as an ANS Condition II event, a fault of moderate frequency. The criterion of interest for the partial loss of flow analysis is that the DNB design basis is satisfied.

III.7.5 Analysis Results

Figures III.7-1 through III.7-4 shows the transient responses for the partial loss of flow event. The reactor is tripped on a low flow signal. The calculated minimum DNBR value for the MUR power uprate is [ ][a,c] compared to a DNBR safety analysis limit of [ ][a,c]. Therefore, all applicable acceptance criteria are met for the partial loss of flow event at MUR power uprate conditions and the conclusions presented in the UFSAR remain valid.

The calculated sequence of events is shown in Table III.7-1. The affected reactor coolant pumps will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained, at which point, normal shutdown may proceed.

A comparison of the results from the MUR power uprate analysis to those from the current licensing basis analysis is presented in Table III.7-2. From this comparison it can be seen that the MUR analysis yields slightly less limiting results than the current licensing basis analysis. This is due to the increased DNB margin created by the revised core thermal limits and through the use of the VIPRE code to calculate DNB. Thus, the decrease in margin caused by the uprated power level is offset such that the overall margin for this event is maintained.

III.7.6 Refer ences

III.7-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.7-2 WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

III.7-3 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

Table III.7-1 Sequence of Events

Event	Time (Seconds)
Coastdown Begins	0.0
Low Flow Reactor Trip	1.7
Rods Begin to Drop	2.7
Minimum DNBR Occurs	3.8

Table III.7-2 Comparison of Results to the Current Licensing Basis

Case Description	DNBR Safety Analysis Limit	Calculated Value
Licensing Basis – Loss of Two RCPs with Four Loops in Operation	[ ][a,c]	[ ][a,c]
MUR – Loss of Two RCPs with Four Loops in Operation	[ ][a,c]	[ ][a,c]

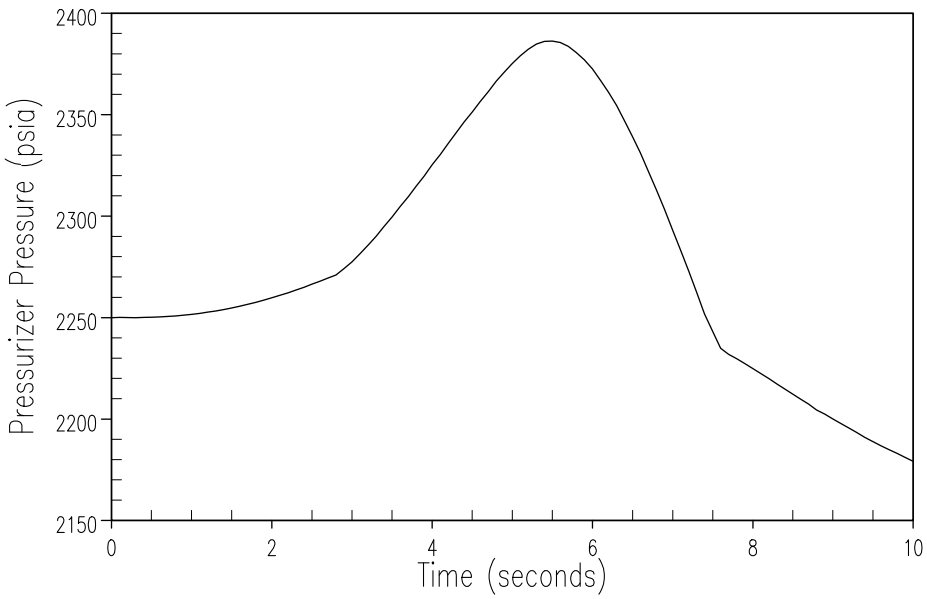

Figure III.7-1 Partial Loss of Flow – Pressurizer Pressure vs. Time

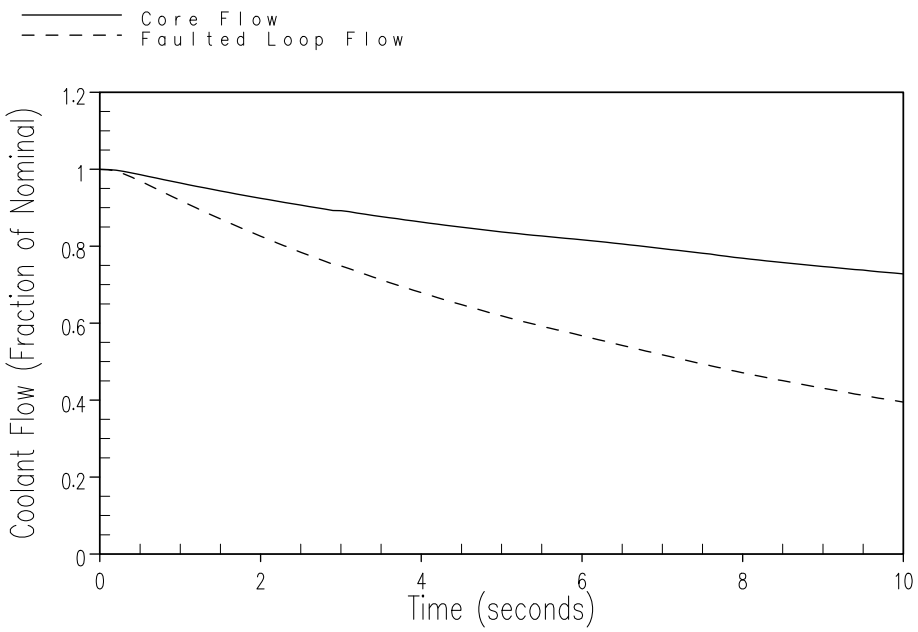

Figure III.7-2 Partial Loss of Flow – Coolant Flow vs. Time

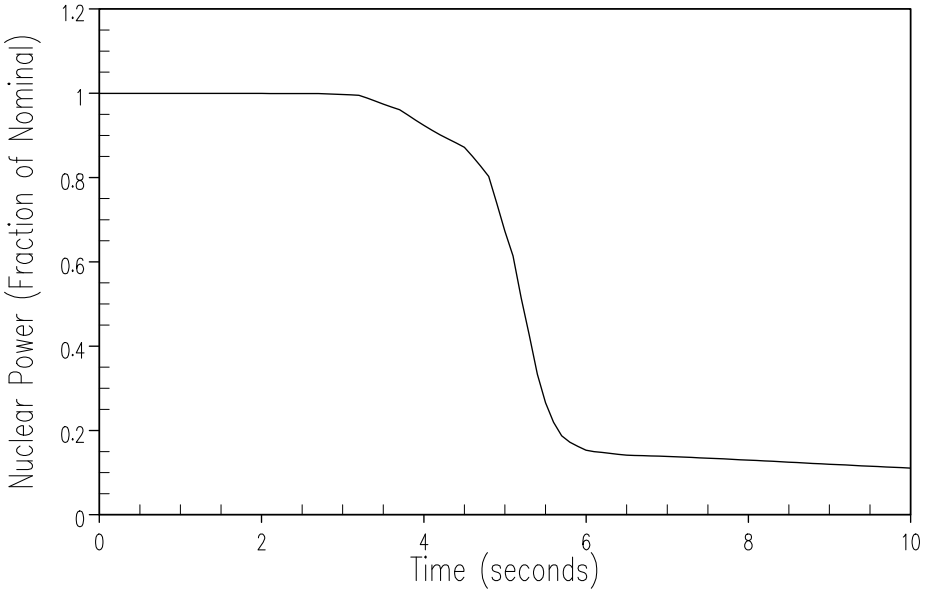

Figure III.7-3 Partial Loss of Flow – Nuclear Power vs. Time

a,c

Figure III.7-4 Partial Loss of Flow – DNBR vs. Time

III.8 Complete Loss of Forced Reactor Coolant Flow – UFSAR 15.3.2

III.8.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps (RCPs). If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

III.8.2 Method of Analysis

The complete loss of flow analysis for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN computer code (Reference III.8-1) and VIPRE computer code (Reference III.8-2) along with the Revised Thermal Design Procedure (RTDP) methodology (Reference III.8-3) to calculate a minimum departure from nucleate boiling ratio (DNBR). Two complete loss of flow scenarios are analyzed:

1. Complete loss of all four RCPs with four loops in operation
2. Frequency decay event resulting in a complete loss of forced reactor coolant flow.

The transient response of the current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) power uprate.

III.8.3 Analysis Inputs and Assumptions

a. Initial reactor power, pressure, and RCS temperatures (consistent with the MUR uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.8-3.

b. The flow coastdown is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics and is based on high estimates of system pressure losses.

c. The most-negative Doppler-only power coefficient is modeled since it maximizes the positive reactivity addition during the trip (which acts to retard the power decrease).

d. A full power moderator temperature coefficient of 0 pcm/°F corresponding to minimum feedback is assumed. This assumption is valid and appropriate even for a plant employing a positive moderator temperature coefficient (PMTC). Typically, a PMTC is allowed by the plant Technical Specifications from 0% to some part power level (in this case 70%), at which point the PMTC linearly ramps to zero (see Technical Specifications Figure 3.1.3-1). The full power analysis results using an MTC of 0 pcm/°F bound those for part-power initial conditions with a PMTC at the licensed allowable MTC limit.

e. The RCP Power Supply Undervoltage or Underfrequency reactor trip functions or the Low Reactor Coolant System Flow reactor trip function are credited as being available to mitigate the effects of this event.

f. The RCPs begin to coastdown upon reaching the undervoltage trip setpoint (modeled to occur at 0 second) for the case 1 complete loss of flow scenario. Rod motion following the undervoltage trip is modeled at 1.5 seconds (reflects the undervoltage trip time delay of 1.5 seconds).

g. The case 2 complete loss of flow scenario models a frequency decay of 5 Hz/sec at 0 second. At 1.2 seconds, the underfrequency trip setpoint of 54.0 Hz is reached. Rod motion occurs at 1.8 seconds, following a 0.6 second underfrequency trip time delay.

h. The most limiting single failure for a complete loss of flow event is the failure of a protection train. No single active failure will prevent the reactor protection system from functioning properly.

III.8.4 Analysis Acceptance Criteria

The complete loss of flow event is classified as an ANS Condition III event, an infrequent event. The criterion of interest for the complete loss of flow analysis is that the DNB design basis is satisfied.

III.8.5 Analysis Results

Figures III.8-1 through III.8-4 show the transient responses for the limiting case, which is the frequency decay complete loss of flow event (case 2). The reactor is tripped on an underfrequency signal. The calculated minimum DNBR value for the MUR power uprate is [ ]^{a,c} compared to a DNBR safety analysis limit of [ ]^{a,c}. Therefore, all applicable acceptance criteria are met for the complete loss of flow event at MUR power uprate conditions and the conclusions presented in the UFSAR remain valid.

The calculated sequence of events is shown in Table III.8-1. The speed of the reactor coolant pumps will decrease from the 5 Hz/sec frequency decay until a pump trip occurs on the underfrequency condition. Following pump trip, the reactor coolant pumps continue to coast down and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition will be attained, at which point, normal shutdown may proceed.

A comparison of the results from the MUR power uprate analysis to those from the current licensing basis analysis is presented in Table III.8-2. From this comparison it can be seen that the MUR analysis yields slightly less limiting results than the current licensing basis analysis. This is due to the increased DNB margin created through the use of the VIPRE code to calculate DNB. Thus, the decrease in margin caused by the uprated power level is offset such that the overall margin for this event is maintained.

III.8.6 References

III.8-1 WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

III.8-2 WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

III.8-3 WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

Table III.8-1 Sequence of Events

Event	Time (Seconds)
Frequency Decay to All Four RCPs Begins	0.0
Underfrequency Trip Setpoint is Reached	1.2
Rods Begin to Drop	1.8
Minimum DNBR Occurs	3.9

Table III.8-2 Comparison of Results to the Current Licensing Basis

Case Description	DNBR Safety Analysis Limit	Calculated Value
Licensing Basis – Frequency decay event resulting in a complete loss of flow	[ ][a,c]	[ ][a,c]
MUR – Frequency decay event resulting in a complete loss of flow	[ ][a,c]	[ ][a,c]

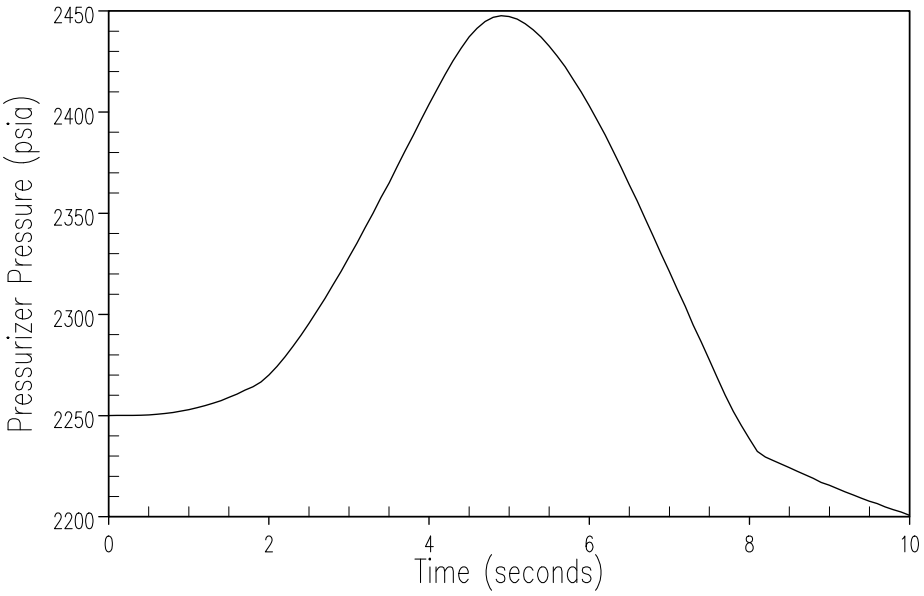

Figure III.8-1 Complete Loss of Flow – Pressurizer Pressure vs. Time

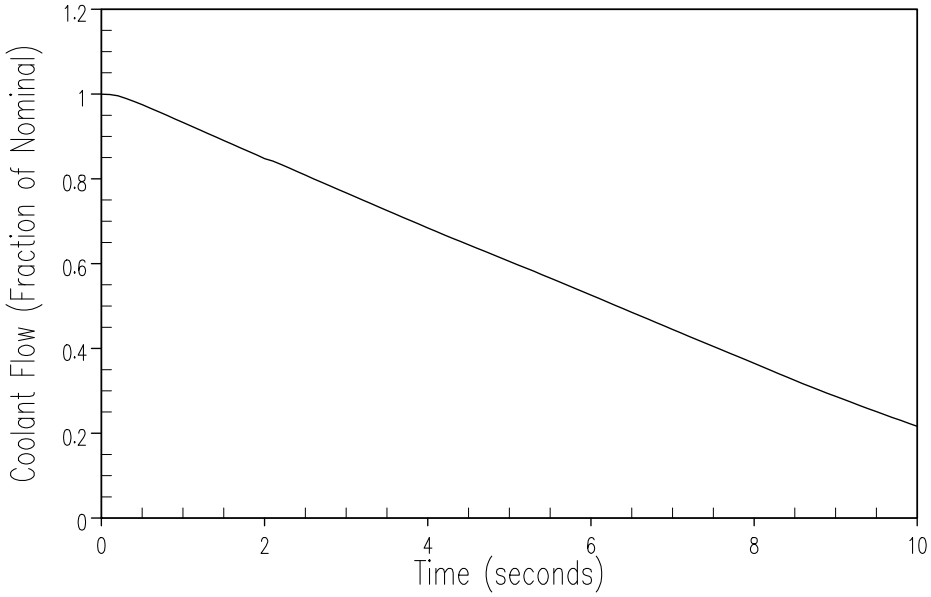

Figure III.8-2 Complete Loss of Flow – Coolant Flow vs. Time

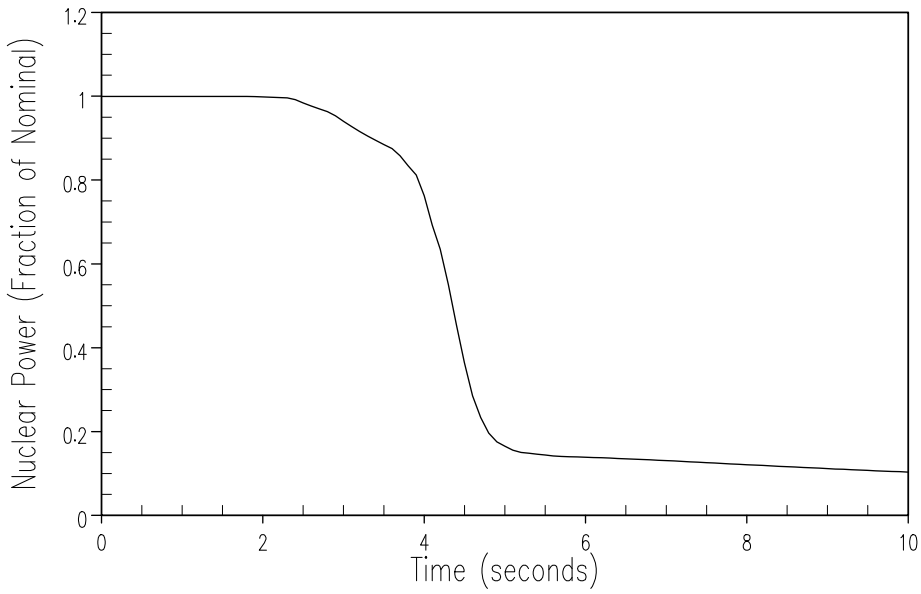

Figure III.8-3 Complete Loss of Flow – Nuclear Power vs. Time

a,c

Figure III.8-4 Complete Loss of Flow – DNBR vs. Time

III.9 Reactor Coolant Pump Shaft Seizure (Locked Rotor) / Reactor Coolant Pump Shaft Break / Locked Rotor with Loss of Offsite Power– UFSAR 15.3.3 through 15.3.5

III.9.1 Identification of Causes and Accident Description

A transient analysis is performed for the instantaneous seizure of a reactor coolant pump (RCP) rotor (locked rotor). Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator is reduced, first because the reduced flow results in a decreased tube side film coefficient, and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the RCS. The pressure increase actuates the automatic spray system, opens the power operated relief valves (PORVs), and opens the pressurizer safety valves (PSVs). The sequence of events initiated by the insurge depends on the rate of insurge and pressure increase. The PORVs are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in this analysis.

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. Only one analysis, which permits reverse spinning but no forward flow, has been performed and represents the most limiting condition for the locked rotor and pump shaft break accidents. This analysis also models a loss of offsite power concurrent with the time of trip.

III.9.2 Method of Analysis

Two cases are typically examined for the locked rotor event. The first case focuses on maximizing the primary system pressure, fuel clad temperature, and zirc-water reaction. This case is referred to as the peak pressure/peak clad temperature (PCT) case. The second case determines the percentage of fuel rods that experience a departure from nucleate boiling ratio (DNBR) less than the limit value. This case is referred to as the rods-in-DNB case.

Peak Pressure/Peak Clad Temperature Case

The peak pressure/PCT case of the current licensing basis is performed to confirm that the reactor coolant system pressure, peak clad average temperature, and hot spot zirc-water reaction limits are not exceeded. This case is analyzed using Standard Thermal Design Procedure (STDP) methodology, where the uncertainties are explicitly modeled. Therefore, the peak pressure/PCT case is unaffected by the tradeoff between increased power level and decreased uncertainty; thus, the current licensing basis analysis remains applicable to the Measurement Uncertainty Recapture (MUR) power uprate and the case is not explicitly reanalyzed.

Rods-in-DNB Case

The locked rotor rods-in-DNB case is reanalyzed for the MUR power uprate. The rods-in-DNB case is analyzed with the Revised Thermal Design Procedure (RTDP) methodology (Reference III.9-1). The transient response of the current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the MUR power uprate. The locked rotor rods-in-DNB analysis for Byron and Braidwood Units 1 and 2 is analyzed with the NRC-approved LOFTRAN computer code (Reference III.9-2) and VIPRE computer code (Reference III.9-3) to determine the percentage of fuel rods experiencing a DNBR less than the safety analysis limit. The locked rotor is analyzed as a single locked rotor/shaft break with four loops in operation, concurrent with a loss of offsite power at the time of reactor trip.

III.9.3 Analysis Inputs and Assumptions

a. Initial reactor power, pressure, and RCS temperatures (consistent with the MUR uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.9-1.

b. The flow coastdown is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics and is based on high estimates of system pressure losses.

c. The most-negative Doppler-only power coefficient is modeled since it maximizes the positive reactivity addition during the trip (which acts to retard the power decrease).

d. A full power moderator temperature coefficient of 0 pcm/°F corresponding to minimum feedback is assumed. This assumption is valid and appropriate even for a plant employing a positive moderator temperature coefficient (PMTC). Typically, a PMTC is allowed by the plant Technical Specifications from 0% to some part power level (in this case 70%), at which point the PMTC linearly ramps to zero (see Technical Specifications Figure 3.1.3-1). The full power analysis results using an MTC of 0 pcm/°F bound those for part-power initial conditions with a PMTC at the licensed allowable MTC limit.

e. The Low Reactor Coolant System Flow reactor trip function is credited as being available to mitigate the effects of this event.

f. The most limiting single failure for a locked rotor event is the failure of a protection train. No single active failure will prevent the reactor protection system from functioning properly.

g. The case analyzed models one locked rotor / shaft break with four loops in operation, concurrent with a loss of offsite power at the time of trip.

III.9.4 Analysis Acceptance Criteria

The locked rotor event is classified as an ANS Condition IV event, a limiting fault. As discussed above, only the rods-in-DNB case is impacted by the MUR power uprate. The criterion of interest for the rods-in-DNB case is to demonstrate that the percentage of fuel rods which undergo DNB is less than the percentage of failed fuel rods assumed in the radiological dose calculations.

III.9.5 Analysis Results

Figures III.9-1 through III.9-3 show the transient responses for the locked rotor rods-in-DNB event. The reactor is tripped on a low flow signal. The calculated percentage of fuel rods exceeding the DNBR limit of [ ][a,c] is less than the 2% fuel rod failures assumed in the radiological dose calculations. Since the peak pressure/PCT case is not impacted by the MUR, the results of the current licensing basis analysis continue to demonstrate that the applicable acceptance criteria for the peak pressure/PCT case are met for the MUR power uprate program. Therefore, all applicable acceptance criteria are met for the locked rotor event at MUR power uprate conditions and the conclusions presented in the UFSAR remain valid.

The calculated sequence of events is shown in Table III.9-1. The core flow rapidly coasts down to a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained, at which point, normal shutdown may proceed.

A comparison of the results from the MUR power uprate analysis to those from the current licensing basis analysis is presented in Table III.9-2. From this comparison it can be seen that the MUR analysis is more limiting than the current licensing basis analysis. This is due to how the rods-in-DNB value is calculated. In particular, the MUR DNB analysis uses more conservative inputs than the AOR to create a reference analysis that is expected to bound cycle to cycle variations in future reloads. Thus, the apparent decrease in margin is acceptable.

III.9.6 Refer ences

III.9-1 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

III.9-2 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.9-3 WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

Table III.9-1 Sequence of Events

Event	Time (Seconds)
Rotor on One Pump Locks	0.0
Low Flow Trip Setpoint Reached	0.03
Rods Begin to Drop	1.03
Maximum Rods-in-DNB Occurs	3.0

Table III.9-2 Comparison of Results to the Current Licensing Basis

Case Description	Rods-in-DNB Limit	Calculated Value
Licensing Basis – Locked Rotor/Sheared Shaft	2.0%	0.10%
MUR – Locked Rotor/Sheared Shaft	2.0%	< 2.0%*

* The MUR analysis uses more conservative inputs (specifically, the use of the generic locked rotor census adjusted to the maximum $F_{\Delta H}$ for the Byron and Braidwood units) than the AOR. This was done to create a reference analysis that is expected to bound cycle-to-cycle variations in future reloads. Thus, the apparent decrease in margin is acceptable.

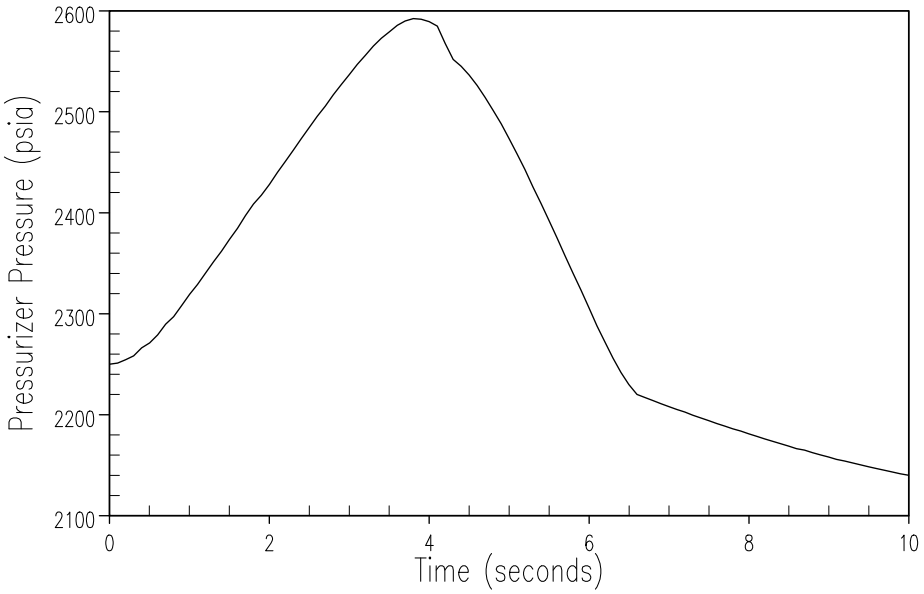

Figure III.9-1 Locked Rotor/Sheared Shaft Rods-in-DNB – Pressurizer Pressure vs. Time

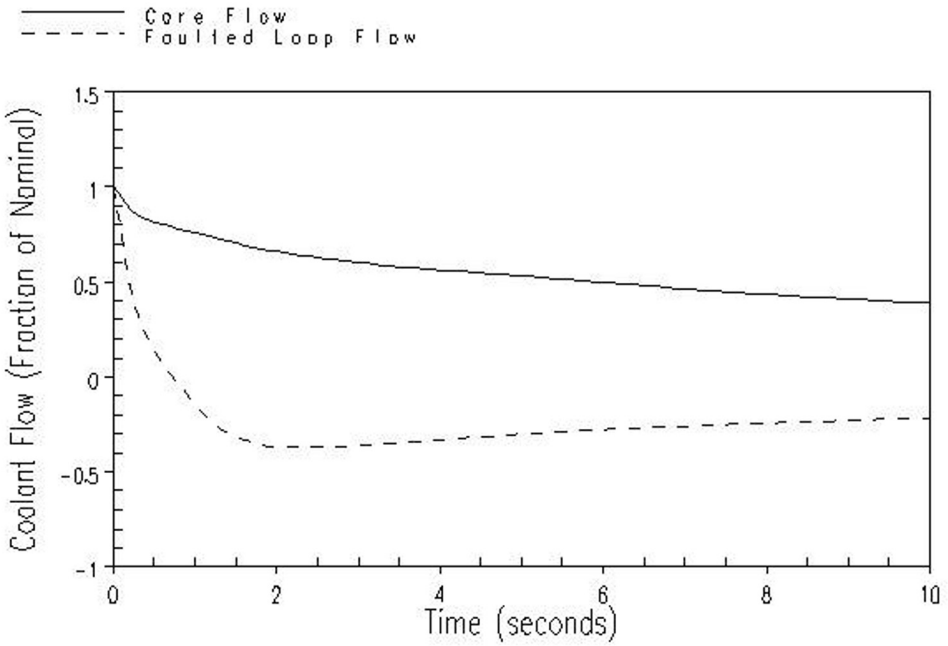

Figure III.9-2 Locked Rotor/Sheared Shaft Rods-in-DNB – Coolant Flow vs. Time

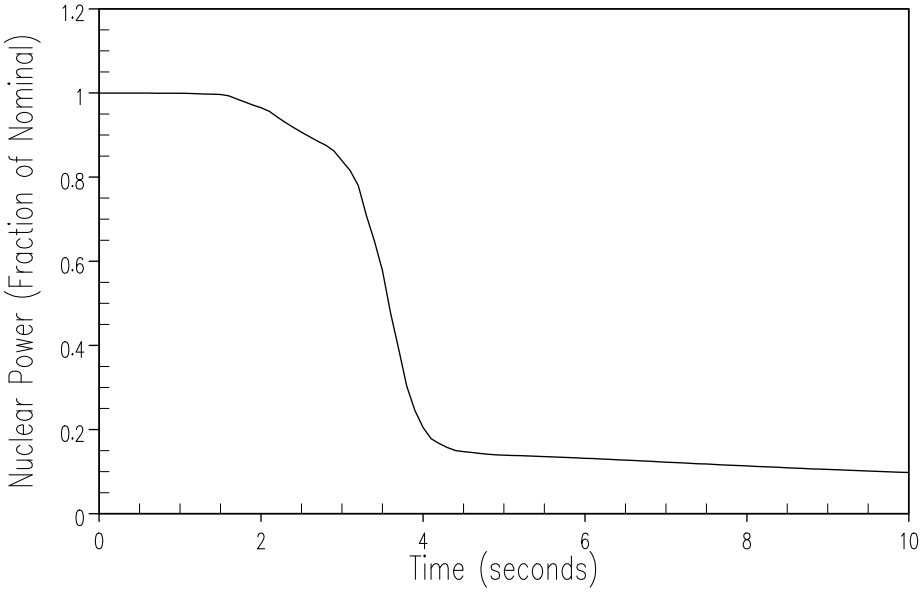

Figure III.9-3 Locked Rotor/Sheared Shaft Rods-in-DNB – Nuclear Power vs. Time

III.10 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power – UFSAR 15.4.2

III.10.1 Identification of Causes and Accident Description

An uncontrolled Rod Cluster Control Assembly (RCCA) withdrawal at power causes an increase in the core heat flux and may result from faulty operator action or a malfunction in the rod control system. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a violation of the DNBR design basis. Therefore, in order to avert damage to the fuel clad, the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit value or the fuel rod linear heat generation rate limit is exceeded.

The automatic features of the reactor protection system which prevent core damage following the postulated accident include the following:

a. Reactor trip on power range neutron flux if two-of-four channels exceed an overpower setpoint.

b. Reactor trip on positive neutron flux rate if two-out-of-four channels exceed a positive setpoint.

c. Reactor trip on overtemperature ΔT if two-out-of-four ΔT channels exceed a setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against departure from nucleate boiling (DNB).

d. Reactor trip on overpower ΔT if two-out-of-four ΔT channels exceed a setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable heat generation rate (kW/ft) is not exceeded.

e. Reactor trip on high pressurizer pressure if two-out-of four pressure channels exceed a fixed setpoint.

f. Reactor trip on high pressurizer water level if two-out-of-three water level channels exceed the setpoint when the reactor power is above approximately 10% (Permissive P-7).

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks which are not credited in the accident analysis but would serve to limit the severity of the event. These are:

a. High neutron flux (one-out-of-four power range),

b. Overpower ΔT (two-out-of-four), and

c. Overtemperature ΔT (two-out-of-four).

III.10.2 Method of Analysis

Multiple cases are analyzed assuming a range of reactivity insertion rates for both minimum and maximum reactivity feedback conditions at various power levels. Separate cases are analyzed to address specific acceptance criteria; specifically, minimum departure from nucleate boiling ratio (DNBR) and maximum reactor coolant system (RCS) pressure. For overpressure concerns, the Standard Thermal

Design Procedure (STDP) methodology is used, where the uncertainties on the initial conditions (i.e., power, temperature, pressure, and flow) are explicitly modeled. Therefore, the overpressure analyses are unaffected by the tradeoff between the increased power level and decreased uncertainty; thus, the overpressure analyses are not impacted by the Measurement Uncertainty Recapture (MUR). For DNB cases, the Revised Thermal Design Procedure (Reference III.10-1) is used. The cases presented in Section III.10.5 are representative for this event.

For the DNB cases a reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the MUR power uprate. The transients are analyzed using the LOFTRAN Code (Reference III.10-2). This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperature, pressures, and power level.

III.10.3 Analysis Inputs and Assumptions

Since Unit 1 and 2 at Byron and Braidwood Nuclear Power Stations operate with different steam generator models, this accident is analyzed with input modeling both steam generator designs. This approach ensures that the limiting cases for minimum DNBR, maximum primary pressure, and maximum secondary pressure are identified.

For an uncontrolled RCCA bank withdrawal at power accident, the analysis assumes the following assumptions:

a. Initial Nuclear Steam Supply System (NSSS) power (3672 MWt), pressure, and RCS temperatures are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the limit DNBR.

b. For reactivity coefficients, two cases are analyzed.

   1) <u>Minimum Reactivity Feedback:</u> A most-positive moderator temperature coefficient of reactivity (0 pcm/°F at 100%power, +7 pcm/°F ≤ 70% power) and a least-negative Doppler-only power coefficient form the basis for the beginning-of-life (BOL) minimum reactivity feedback assumption.

   2) <u>Maximum Reactivity Feedback:</u> A conservatively large positive moderator density coefficient of 0.54 $\Delta k/g/cm^3$ (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler-only power coefficient form the basis for the end-of-life (EOL) maximum reactivity feedback assumption.

c. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power (i.e., 118% of 3672 MWt). The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

d. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

e. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the two control banks having a maximum combined worth at a speed of 45 inches/minute or 72 steps/minute.

f. Power levels of 10%, 60%, and 100% are considered.

g. The impact of a full power RCS T_{avg} window was considered for the uncontrolled RCCA bank withdrawal at power analysis. A conservative calculation modeling the high end of the T_{avg} window was explicitly analyzed since this is limiting with respect to DNBR results.

III.10.4 Analysis Acceptance Criteria

Based on its frequency of occurrence, the uncontrolled RCCA bank withdrawal at power accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the main acceptance criteria associated with this event.

1. The fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the Safety Analysis Limit DNBR value for the entire transient.

2. Fuel integrity shall be maintained by ensuring that the maximum heat flux does not exceed the prescribed limit at any time during the transient.

3. The pressure in the reactor coolant system and main steam system shall be maintained below 110% of the design value. Peak primary pressure results, documented for the previous analysis of record, remain valid for the MUR power uprate. With respect to peak secondary pressure, the uncontrolled RCCA bank withdrawal at power accident is bounded by the Loss of Load/ Turbine Trip analysis. The loss of load/ turbine trip analysis is described in Section III.6.

4. An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently. This criterion is met by ensuring that the pressurizer does not reach a water solid condition. Pressurizer filling (water solid) is not a concern for this event since the high pressurizer water level reactor trip will trip the reactor if the pressurizer approaches a filled condition. For post reactor trip considerations, the event is bounded by the Loss of Normal Feedwater event.

5. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding. This criterion is met by demonstrating that the DNB design basis is satisfied.

The limiting single failure for this event, as defined in Appendix A to 10 CFR Part 50, is assumed to be the failure of one train of the reactor protection system. The protection function is carried out by the other train of the protection system, which remains functional, in compliance with Criterion 20 and Criterion 21

of the General Design Criteria. No single active failure would result in the loss of the protection function and removal from service of any component or channel does not result in the loss of required minimum redundancy.

III.10.5 Analysis Results

Figures III.10-1 through II.10-6 show the transient response for a rapid RCCA withdrawal incident (80 pcm/sec) starting from 100% power with minimum reactivity feedback. Reactor trip on high neutron flux occurs shortly after the start of the transient. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and margin to the safety analysis limit DNBR is maintained.

The transient response for a slow RCCA withdrawal (3 pcm/sec) from full power is shown in Figures III.10-7 and III.10-12. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the safety analysis limit value.

Figure III.10-13 shows the minimum DNBR as a function of reactivity insertion rate from 100% power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the safety analysis limit value.

Figures III.10-14 and III.10-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60% and 10% power, respectively. The results are similar to the 100% power case; however, as the initial power level decreases, the range where the overtemperature ΔT trip is effective is increased. The minimum DNBR never falls below the safety analysis limit value.

The bounding minimum DNBR values when modeling the BWI steam generators are less than those calculated when modeling the Westinghouse D5 steam generators.

The calculated sequence of events for the two uncontrolled RCCA withdrawal incidents is provided in Table III.10-1and documented in Figures III.10-1 through II.10-12.

A comparison of the results from the DNB cases analyzed for the MUR power uprate to those from the current licensing basis analysis is presented in Table III.10-2. From this comparison it can be seen that the increase in power associated with the MUR causes the minimum DNBR to decrease slightly. However, the minimum DNBR for both units remains above the Safety Analysis Limit.

III.10.6 Refer ences

III.10-1 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

III.10-2 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

Table III.10-1 Sequence of events – Uncontrolled RCCA Bank Withdrawal at Power Analysis

Case Description	Event	Time (sec)
100% Power, Minimum Feedback Fast RCCA Withdrawal (80 pcm/sec)	Initiation of uncontrolled RCCA withdrawal	0.0
	Power range neutron flux – high setpoint reached	1.6
	Rods begin to fall	2.1
	Minimum DNBR is reached	3.2
100% Power, Minimum Feedback Slow RCCA Withdrawal (3.0 pcm/sec)	Initiation of uncontrolled RCCA withdrawal	0.0
	Overtemperature ΔT setpoint reached	35.7
	Rods begin to fall	43.7
	Minimum DNBR is reached	44.3

Table III.10-2 Comparison to Analysis of Record - Uncontrolled RWAP from 100% Power Analysis DNB Limiting Case Results

	Current AOR	MUR Analysis
Case producing minimum DNBR	Min. feedback, 0.63 pcm/sec insertion rate	Min. feedback, 3.0 pcm/sec insertion rate
Minimum DNBR(1)	[ ]a,c	[ ]a,c
Case producing Maximum Heat Flux	Min. feedback, 0.63 pcm/sec insertion rate	Max. feedback, 40 pcm/sec insertion rate
Maximum Heat Flux (fraction of nominal)(2)	[ ]a,c	[ ]a,c

Notes:
1. The Safety Analysis Limit (SAL) DNBR for the current AOR is [ ]a,c; the SAL DNBR for the MUR analysis is [ ]a,c.
2. The allowable maximum heat flux is 1.1824 fraction of nominal (FON) for the current AOR and 1.19 FON for the MUR analysis.

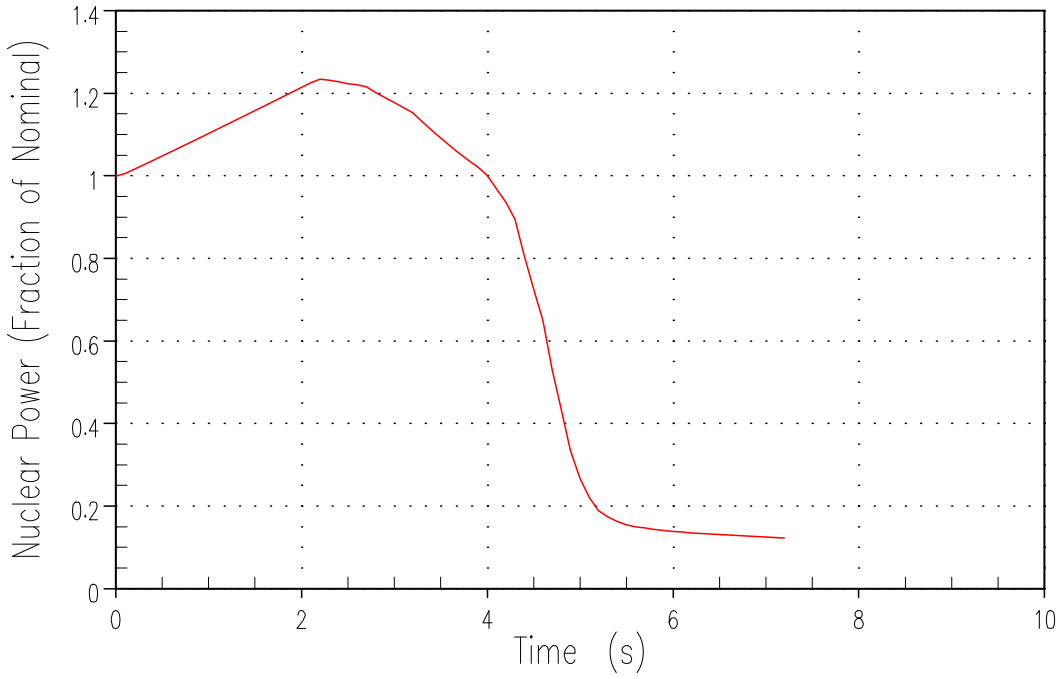

Figure III.10-1 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 80 pcm/sec, Minimum Reactivity Feedback Nuclear Power versus Time

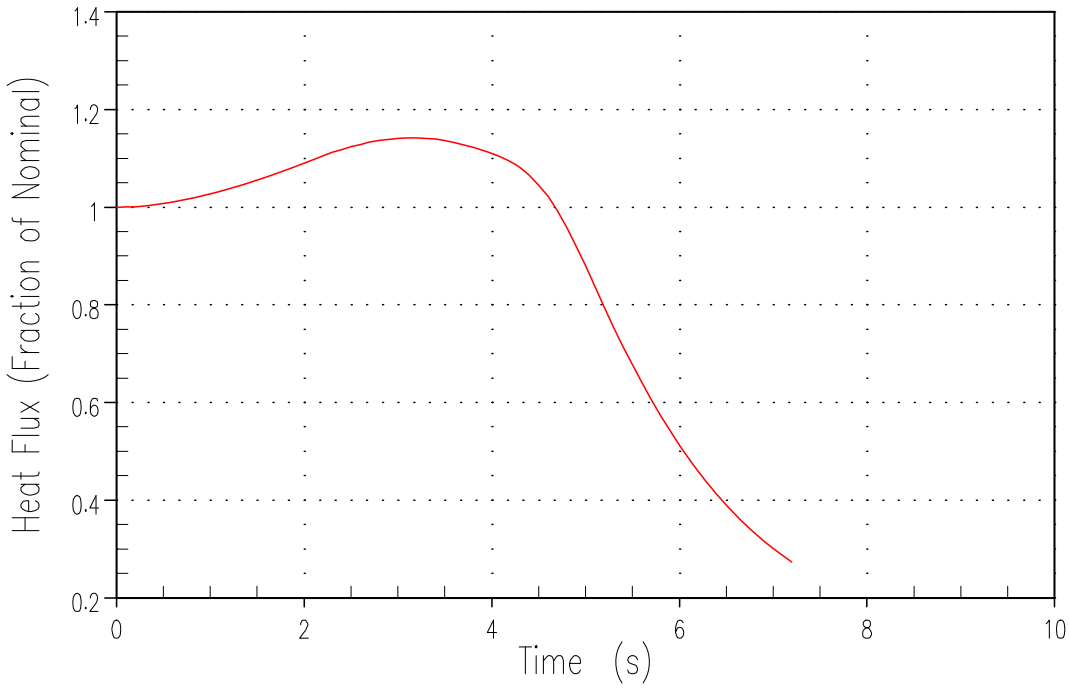

Figure III.10-2 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 80 pcm/sec, Minimum Reactivity Feedback Heat Flux versus Time

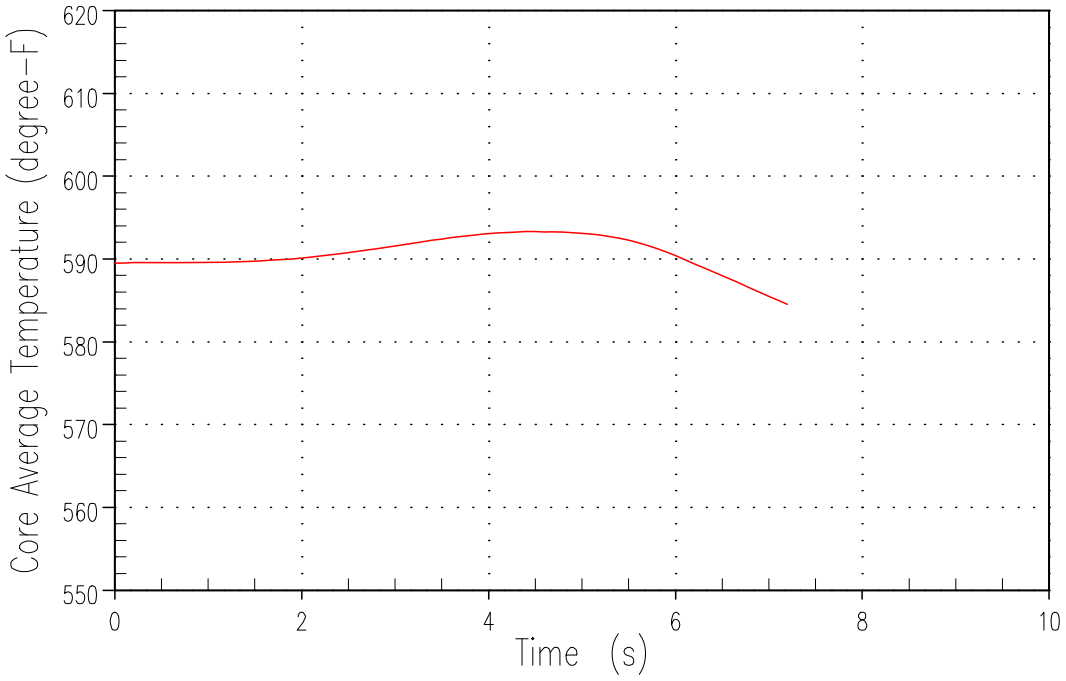

Figure III.10-3 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 80 pcm/sec, Minimum Reactivity Feedback Core Average Temperature versus Time

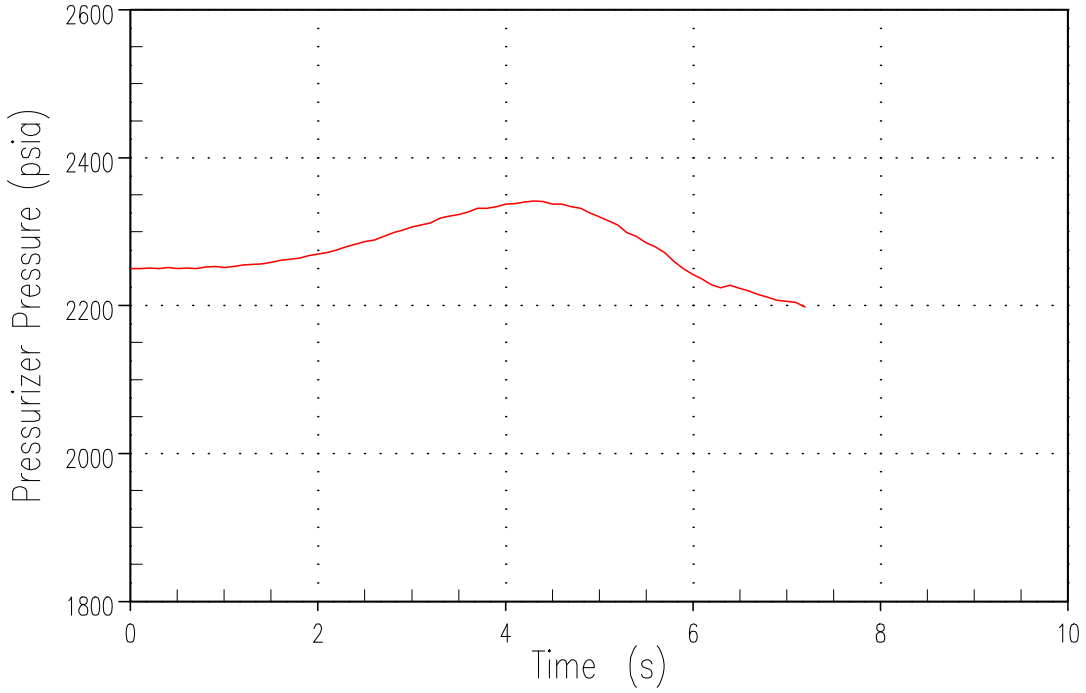

Figure III.10-4 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 80 pcm/sec, Minimum Reactivity Feedback Pressurizer Pressure versus Time

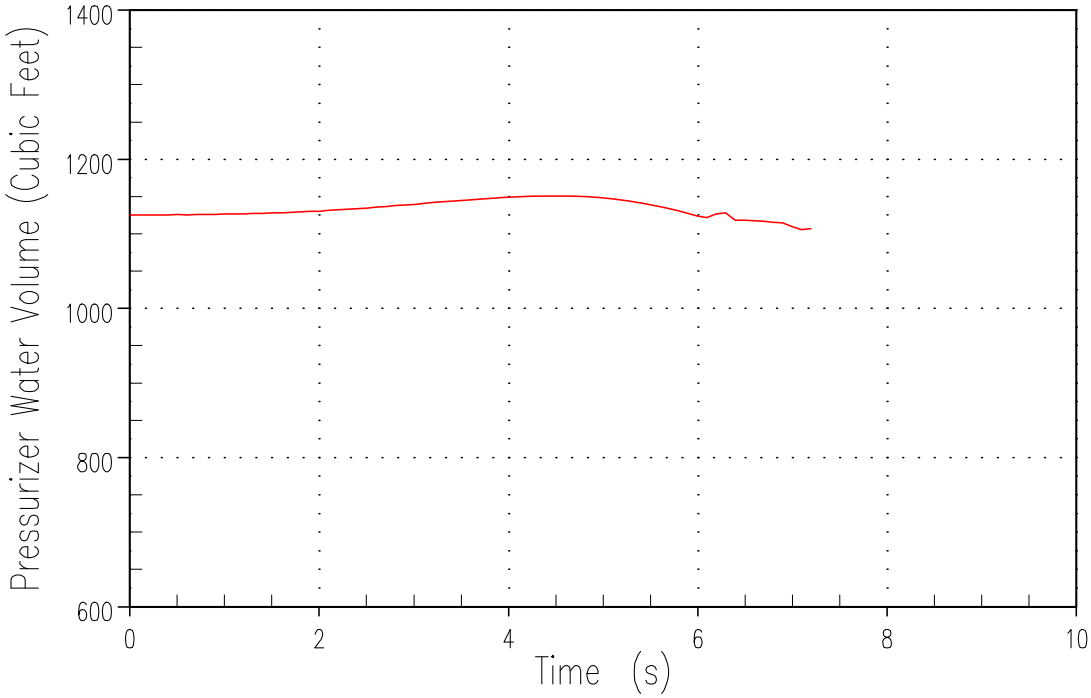

Figure III.10-5 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 80 pcm/sec, Minimum Reactivity Feedback Pressurizer Water Volume versus Time

a,c

Figure III.10-6 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 80 pcm/sec, Minimum Reactivity Feedback DNBR versus Time

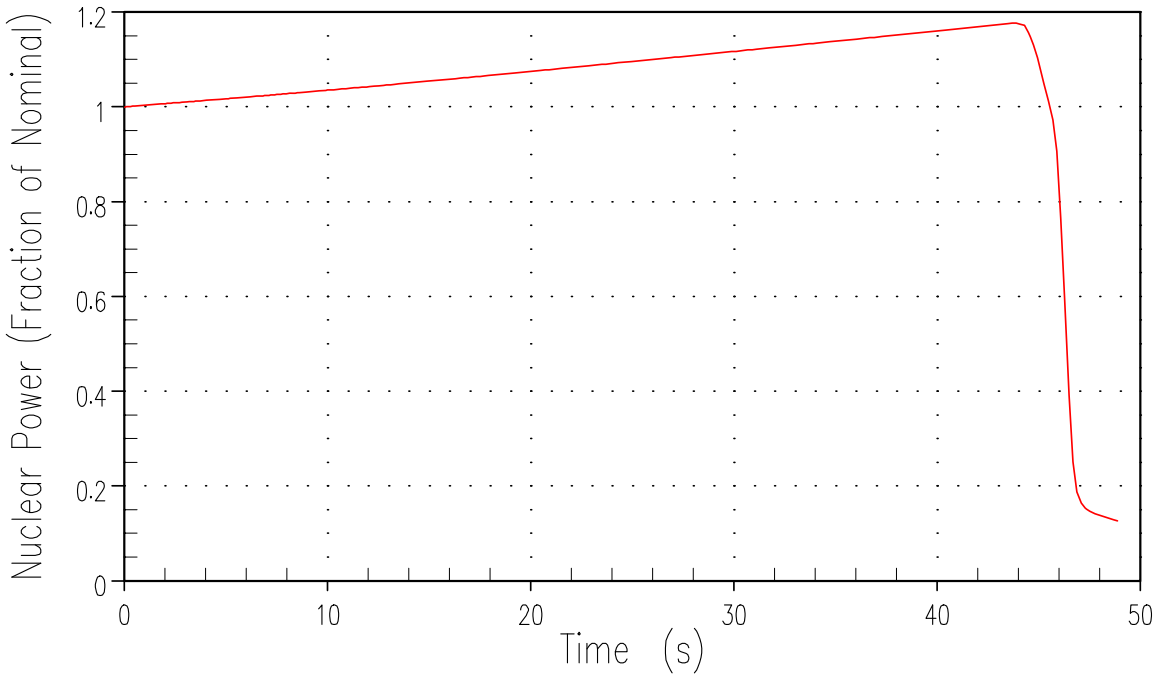

Figure III.10-7 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 3.0 pcm/sec, Minimum Reactivity Feedback Nuclear Power versus Time

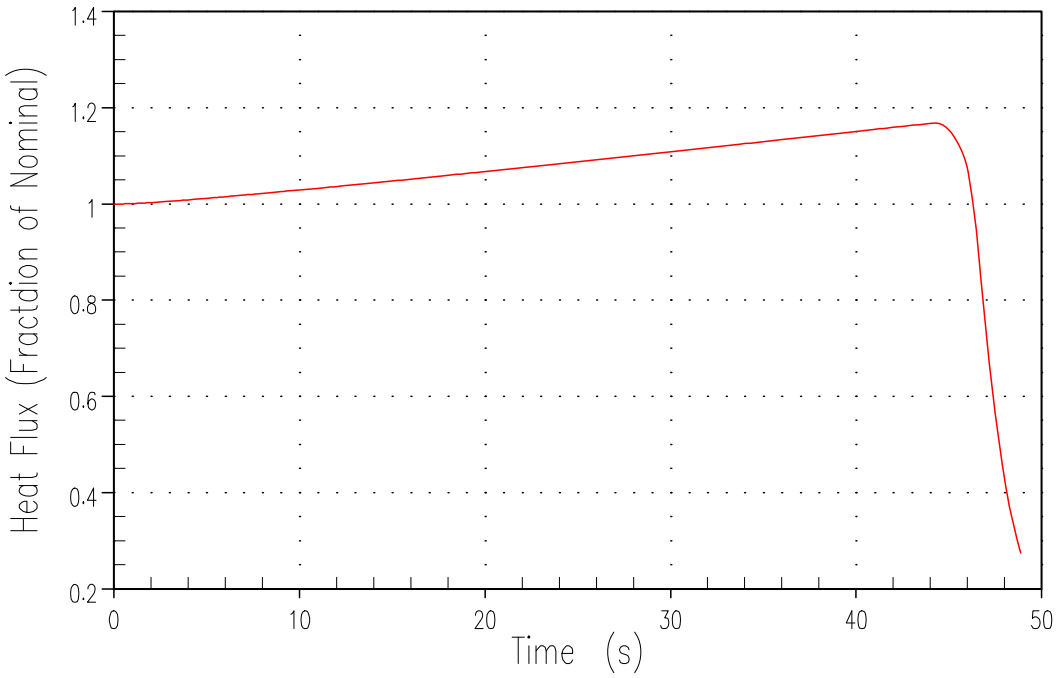

Figure III.10-8 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 3.0 pcm/sec, Minimum Reactivity Feedback Heat Flux versus Time

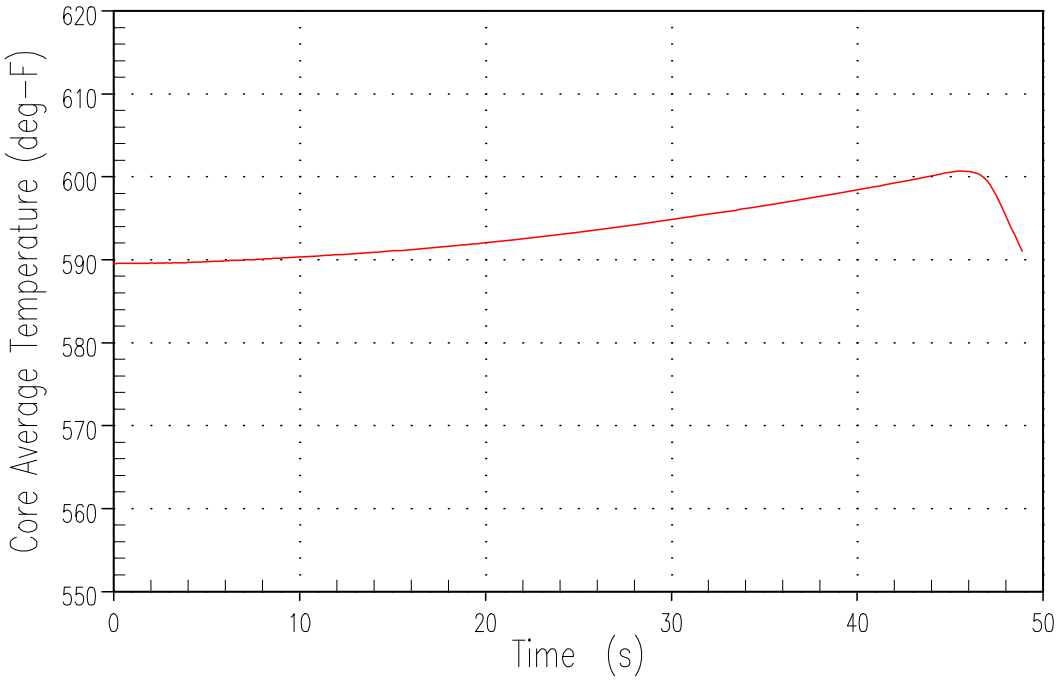

Figure III.10-9 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 3.0 pcm/sec, Minimum Reactivity Feedback Core Average Temperature versus Time

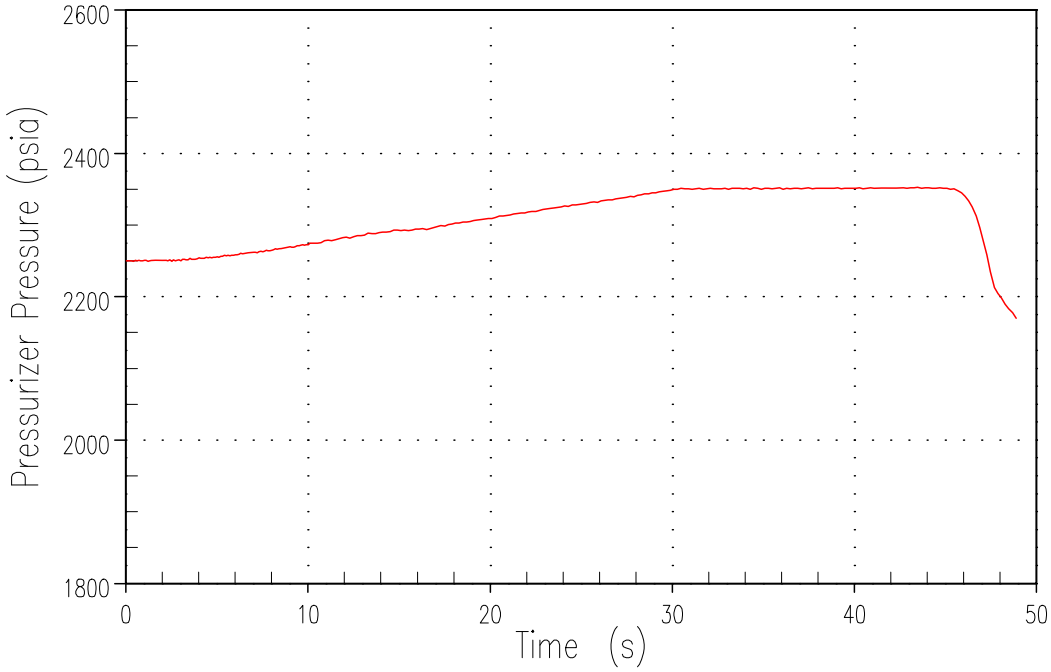

Figure III.10-10 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 3.0 pcm/sec, Minimum Reactivity Feedback Pressurizer Pressure versus Time

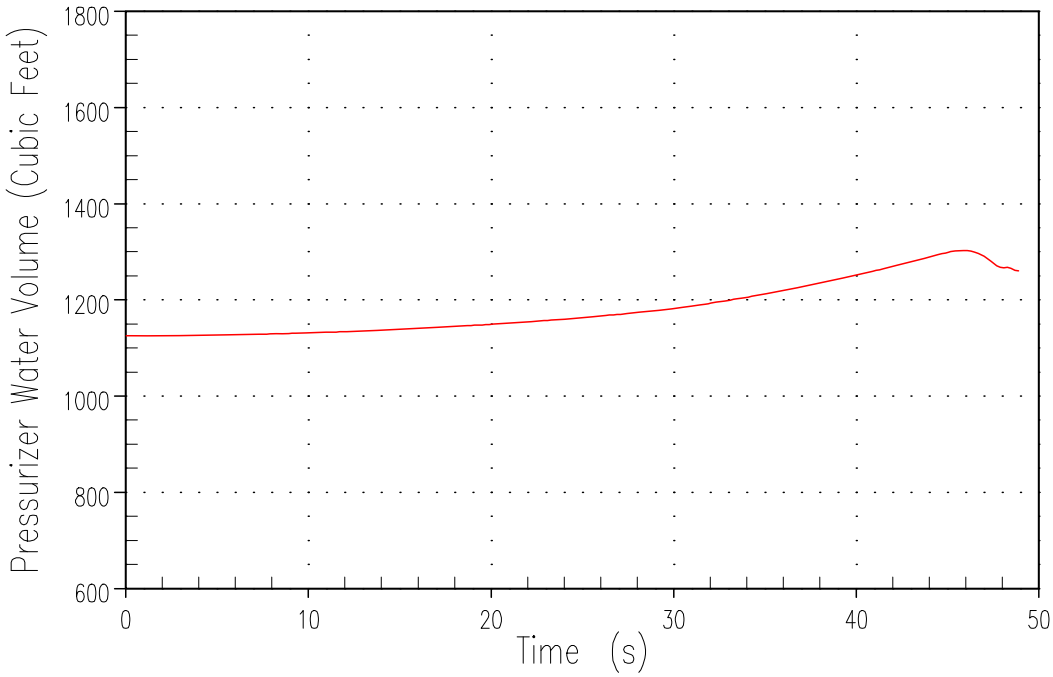

Figure III.10-11 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 3.0 pcm/sec, Minimum Reactivity Feedback Pressurizer Water Volume versus Time

a,c

Figure III.10-12 Uncontrolled RCCA Bank Withdrawal From 100% Power, Withdrawal Rate of 3.0 pcm/sec, Minimum Reactivity Feedback DNBR versus Time

a,c

Figure III.10-13 Uncontrolled RCCA Bank Withdrawal From 100% Power Minimum DNBR Versus Reactivity Insertion Rate

a,c

Figure III.10-14 Uncontrolled RCCA Bank Withdrawal From 60% Power Minimum DNBR Versus Reactivity Insertion Rate

a,c

Figure III.10-15 Uncontrolled RCCA Bank Withdrawal From 10% Power Minimum DNBR Versus Reactivity Insertion Rate

III.11 Inadvertent Operation of the Emergency Core Cooling System During Power Operation – UFSAR 15.5.1

III.11.1 Identification of Causes and Accident Description

Inadvertent operation of the emergency core cooling system (ECCS) at power could be caused by operator error, test sequence error, or a false electrical actuation signal. A spurious signal initiated after the logic circuitry in one solid-state protection system train for any of the following engineered safety feature (ESF) functions could cause this incident by actuating the ESF equipment associated with the affected train.

a. High containment pressure,

b. Low pressurizer pressure, or

c. Low steamline pressure.

Following the actuation signal, the suction of the coolant charging pumps diverts from the volume control tank to the refueling water storage tank. Simultaneously, the valves isolating the charging pumps from the injection header automatically open and the normal charging line isolation valves close. The charging pumps force the borated water from the refueling water storage tank (RWST) through the pump discharge header, the injection line, and into the cold leg of each loop. The passive accumulator tank safety injection and low head system are available. However, they do not provide flow when the reactor coolant system (RCS) is at normal pressure.

A safety injection (SI) signal normally results in a direct reactor trip and turbine trip. However, any single fault that actuates the ECCS will not necessarily produce a reactor trip. If the reactor protection system does not produce an immediate trip as a result of the spurious SI signal, the reactor experiences a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in T_{avg} and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will lessen until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by the manual trip.

Two cases are typically examined for the inadvertent operation of the ECCS at power event. One case is examined for departure from nucleate boiling (DNB) concerns while a second case explicitly addresses pressurizer overfill. As discussed below the DNB case is not bounded by the current analysis and therefore has been reanalyzed. The pressurizer overfill case is bounded by current analysis, but is included in this section for completeness.

III.11.2 Method of Analysis

DNB Case

The inadvertent operation of the ECCS at power DNB case for Byron and Braidwood Units 1 and 2 uses the NRC-approved LOFTRAN computer code (Reference III.11-1) and Revised Thermal Design

Procedure (RTDP) methodology (Reference III.11-2) to calculate a minimum departure from nucleate boiling ratio (DNBR). The current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) power uprate.

Pressurizer Overfill

The Inadvertent ECCS event results in an increase in the RCS inventory that leads to a water solid pressurizer. This event has been evaluated to assess its potential to progress into a SBLOCA event via a Pressurizer Safety Valve (PSV). The PSV's were qualified for water relief though EPRI testing performed in Reference III.11-3, which showed they would reclose following water relief.

The most limiting cases occur with the reactor at full power operation prior to the event. As the current evaluation is based on an NSSS power level of 3672.6 MWt, this evaluation remains bounding for the MUR power uprate, and the conclusions presented in the UFSAR remain valid. The NRC most recently approved the current evaluation in Reference III.11-4.

III.11.3 Analysis Inputs and Assumptions

Cases for both Byron Unit 1 and Braidwood Unit 1 with Babcock and Wilcox International (BWI) steam generators and Byron Unit 2 and Braidwood Unit 2 with Westinghouse Model D5 steam generators were analyzed at conditions designed to provide the most conservative and limiting results. Specifically, the following assumptions are made for the DNB case:

a. The event is analyzed with the RTDP methodology as described in Reference III.11-2. Initial reactor power, RCS pressure and temperature are assumed to be at the nominal full power values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.11-2.

b. The analysis assumes a zero moderator temperature coefficient and a low absolute value Doppler power coefficient at beginning of life.

c. The reactor is assumed to be in manual rod control.

d. Pressurizer heaters assumed to be inoperable. This assumption yields a higher rate of pressure decrease which is conservative. Pressurizer spray and PORVs are assumed available in order to minimize RCS pressure.

e. At the initiation of the event, two charging pumps inject borated water into the cold leg of each loop. The analysis assumes zero injection line purge volume for calculation simplicity; thus, the boration transient begins immediately in the analysis. The positive displacement charging pump is not modeled since it is not actuated by the ECCS signal

f. The turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is full open, turbine load decreases as steam pressure drops.

g. Reactor trip is initiated by a low pressurizer pressure signal at 1860 psia.

h. The decay heat has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip). A conservative residual heat generation based upon long-term operation at the initial power level is assumed.

i. Operator action is not required to mitigate the consequences of this event. Operator action is assumed to occur after the event to stabilize the plant in accordance with approved procedures to bring the plant to the applicable condition.

j. The safety valves setpoints do not impact the minimum DNBR since the PORVs are assumed available to maintain low RCS pressure; this assumption is conservative with respect to DNBR.

k. Auxiliary feedwater is not credited.

l. The main steam safety valves are assumed conservatively to open at +5% above their nominal set pressure for the DNB case. No credit for steam dump is assumed in this analysis.

III.11.4 Analysis Acceptance Criteria

The inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

a. Pressure in the reactor coolant and main steam system should be maintained below 110% of the design values,

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the DNBR limit, derived at a 95% confidence level and 95% probability, and

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

III.11.5 Analysis Results

The limiting (BWI SG case) transient response is shown in Figures III.11-1 through III.11-6. Table III.11-1 shows the calculated sequence of events.

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes T_{AVG}, pressurizer pressure, and pressurizer water level to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the pressurizer low pressure trip setpoint. The DNBR increases throughout the transient.

A comparison of the results from the limiting case analyzed for the MUR power uprate to those from the limiting case in the current licensing basis analysis is presented in Table III.11-2. The minimum DNBR values for both the MUR power uprate analysis and the current licensing basis analysis occur at time zero and increase throughout the event. The reduction in minimum DNBR was expected and can be explained by the 2% power increased modeled in the MUR. Additional power, and therefore heat, is added to the

core making the MUR conditions more limiting when compared to the uprating program for RTDP events.

III.11.6 References

III.11-1 WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

III.11-2 WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

III.11-3 EPRI Document NP-2770-LD, "EPRI/C-E PWR Safety Valve Test Report", January, 1983"

III.11-4 NRC Letter from Mr. George Dick, Jr. to Mr. Christopher Crane dated 8/26/2004, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – Issuance of Amendment, RE: Pressurizer Safety Valve Setpoints" (ML042250516)

Table III.11-1 Sequence of Events

Event	Time (seconds)
Spurious SI signal generated; two charging pumps begin injecting borated water	0.0
Turbine throttle valve wide open, load begins to drop with steam pressure	51.5
Low pressurizer pressure reactor trip setpoint reached	74.1
Control Rod Motion Begins	76.1
Minimum DNBR occurs	0.0

Table III.11-2 Comparison of Results to the Current Licensing Basis

Criteria	Current Value	MUR Value	Limit
Minimum DNBR(1)	[ ]a,c	[ ]a,c	[ ]a,c (2)

Notes:
1. Values presented are obtained from the DNB case.
2. The DNB specified acceptable fuel design limit (SAFDL) for the Current Licensing Basis is [ ]a,c

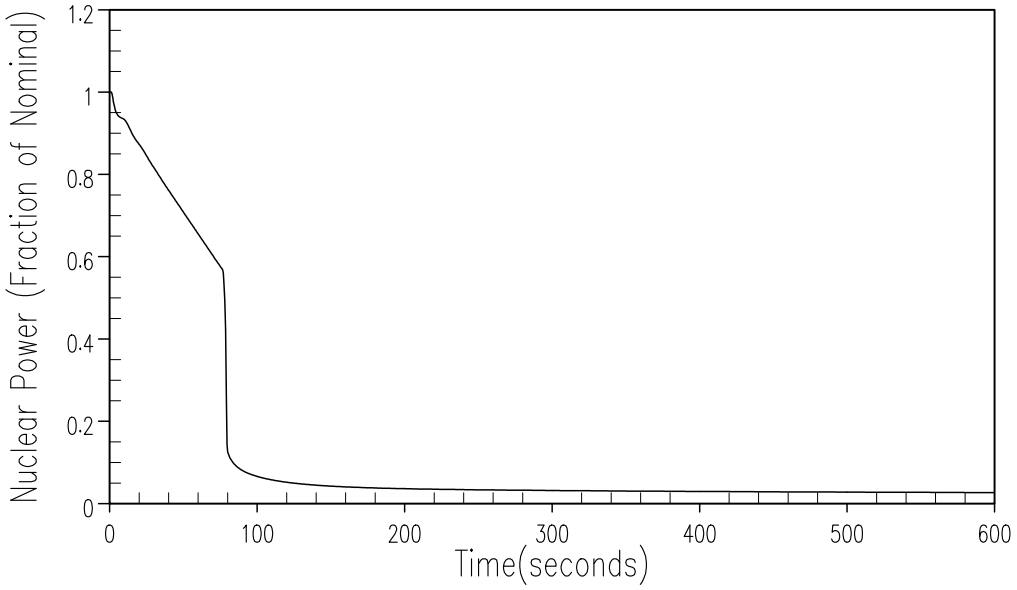

Figure III.11-1 Nuclear Power versus Time

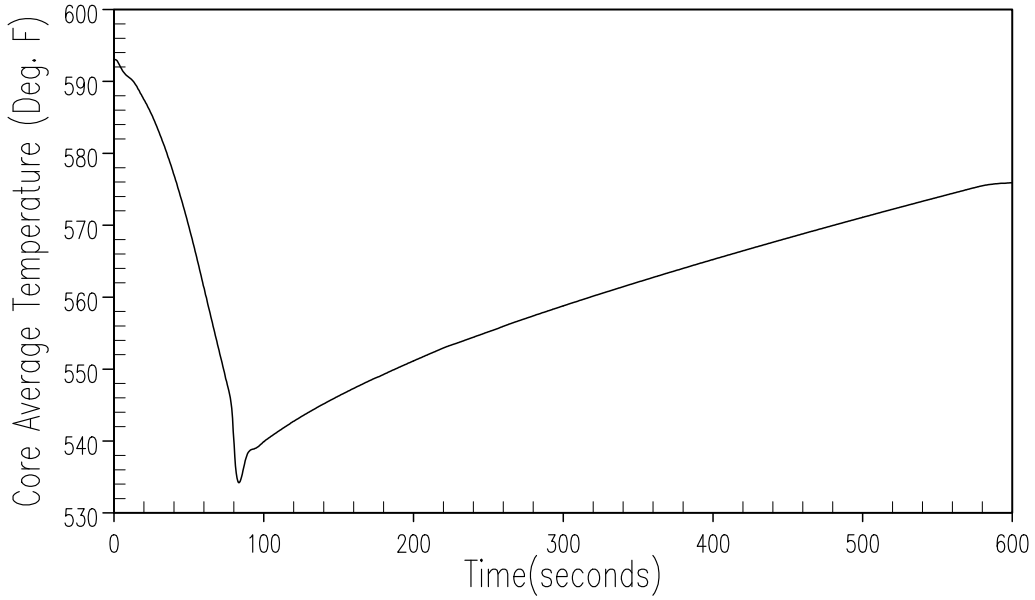

Figure III.11-2 Core Average Temperature versus Time

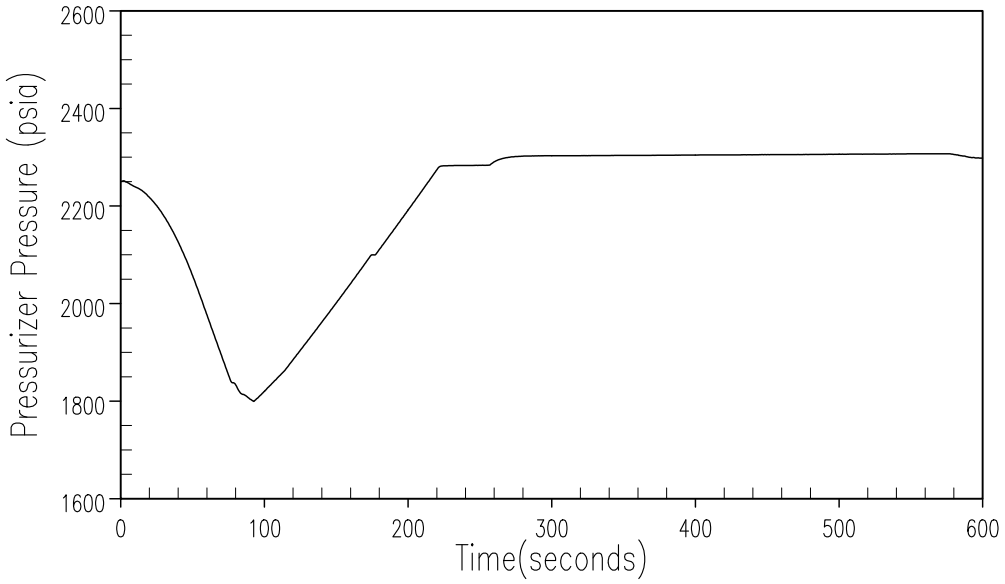

Figure III.11-3 Pressurizer Pressure versus Time

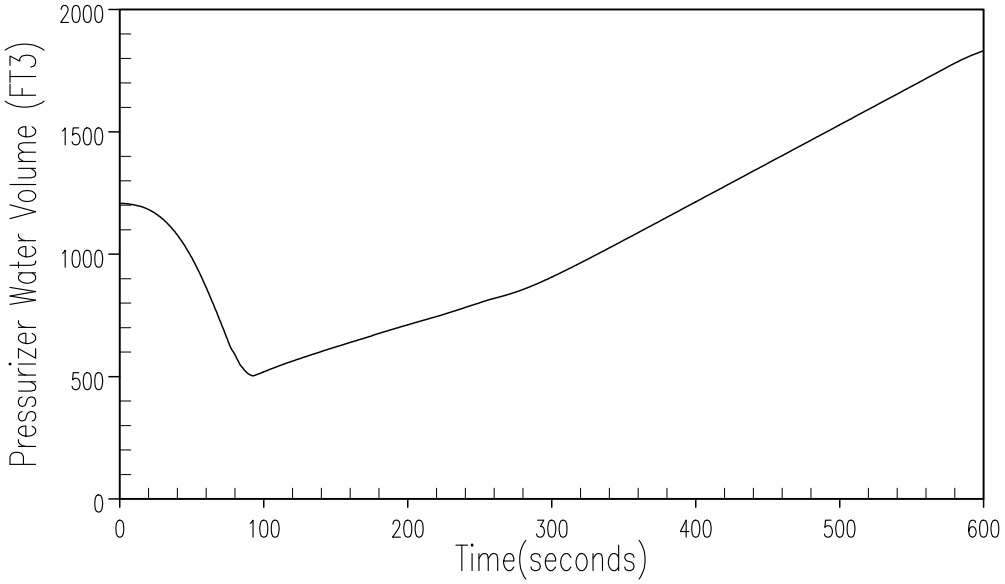

Figure III.11-4 Pressurizer Water Volume versus Time

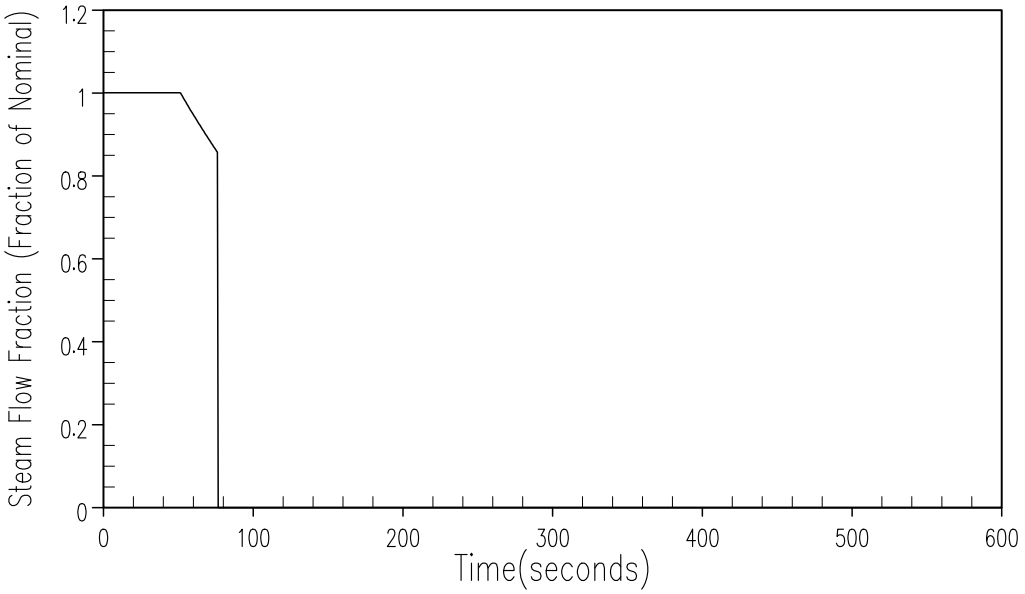

Figure III.11-5 Steam Flow Fraction versus Time

a,c

Figure III.11-6 DNBR versus Time

III.12 Inadvertent Opening of a Pressurizer Safety or Relief Valve – UFSAR 15.6.1

III.12.1 Identification of Causes and Accident Description

An accidental depressurization of the reactor coolant system (RCS) could occur as a result of an inadvertent opening of a pressurizer Power Operated Relief Valve (PORV) or a malfunction of the pressurizer spray system. However, since a Pressurizer Safety Valve (PSV) is designed to relieve approximately twice the steam flow-rate of a PORV, thereby allowing for a much more rapid depressurization upon opening, the accidental depressurization of the RCS event is conservatively analyzed to model the most severe core conditions resulting from an inadvertent opening of a PSV.

Initially the event results in a rapidly decreasing RCS pressure that could reach the hot leg saturation pressure if a reactor trip did not occur. In the presence of a positive moderator density coefficient, the effect of the pressure decrease may result in a decrease in power due to the decrease in moderator density. However, if the rod control system is in automatic mode, it functions to maintain the power and average coolant temperature until reactor trip occurs. Therefore, the accidental depressurization of the RCS analysis models a zero moderator density coefficient and manual rod control.

III.12.2 Method of Analysis

The accidental depressurization of the RCS analysis for Byron and Braidwood Units 1 and 2 uses the NRC approved LOFTRAN computer code (Reference III.12-1) and Revised Thermal Design Procedure (RTDP) methodology (Reference III.12-2) to calculate a minimum departure from nucleate boiling ratio (DNBR). The current licensing basis analysis was performed at 3600.6 MWt with no allowance for power uncertainty, consistent with RTDP methods. A reanalysis was performed at 3672 MWt (conservatively assumed to be nominal power for this analysis) to support the Measurement Uncertainty Recapture (MUR) power uprate.

III.12.3 Analysis Inputs and Assumptions

Cases for both Byron Unit 1 and Braidwood Unit 1 with Babcock and Wilcox International (BWI) steam generators (SGs) and Byron Unit 2 and Braidwood Unit 2 with Westinghouse Model D5 SGs were analyzed at conditions designed to provide the most conservative and limiting results. Specifically, the following assumptions are made:

a. Initial reactor power, pressure, and RCS temperatures (consistent with the MUR uprated power conditions) are assumed to be at their nominal values. With the exception of the RCS average temperature bias, which is explicitly modeled in the analysis, uncertainties in initial conditions are included in the DNBR limit as described in Reference III.12-2.

b. The maximum steam generator tube plugging level is assumed and both the minimum and maximum feedwater temperatures are analyzed for each steam generator design.

c. A full power moderator temperature coefficient of 0 pcm/°F corresponding to minimum feedback is assumed. This assumption is valid and appropriate even for a plant employing a positive moderator temperature coefficient (PMTC). Typically, a PMTC is allowed by the plant Technical Specifications from 0% to some part power level (in this case 70%), at

which point the PMTC linearly ramps to zero (see Technical Specifications Figure 3.1.3-1). The current licensing basis analysis assumed a PMTC of 7 pcm/°F at full power conditions to bound both a part power condition with a PMTC and a full power condition with a 0 MTC. Sensitivity studies performed since the completion of the current licensing basis analysis have shown that assuming a 0 MTC at full power bounds part power conditions with a PMTC.

d. A least negative Doppler-only power coefficient is assumed such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

e. The Low Pressurizer Pressure and Overtemperature ΔT (OTΔT) reactor trip functions are credited as being available to mitigate the effects of this event.

f. The most limiting single failure for an accidental depressurization of the RCS event is the failure of a protection train. No single active failure will prevent the reactor protection system from functioning properly.

III.12.4 Analysis Acceptance Criteria

An inadvertent opening of a pressurizer PORV or spray valve is classified as an ANS Condition II event, a fault of moderate frequency. The criterion of interest for the accidental depressurization of the RCS analysis, which conservatively models the inadvertent opening of a PSV, is that the DNB design basis is satisfied.

III.12.5 Analysis Results

The most limiting case (D5 SGs) at the maximum steam generator tube plugging (SGTP) level with minimum feedwater (FW) temperature) for an inadvertent opening of a PSV is shown on Figures III.12-1 through III.12-5. Figure III.12-1 illustrates the nuclear power transient following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on Low Pressurizer Pressure. The average temperature transient and pressure decay transient following the accident are given in Figures III.12-2 and III.12-3. Pressure drops more rapidly while core heat generation is reduced via the trip, and then slows once saturation temperature is reached in the hot leg. The DNBR decreases initially, but increases rapidly following the trip as shown in Figure III.12-5. The calculated minimum DNBR value for the MUR power uprate is [ ]^{a,c} compared to a DNBR safety analysis limit of [ ]^{a,c}. Therefore, all applicable acceptance criteria are met for the accidental depressurization of the RCS event at MUR power uprate conditions. The calculated sequence of events for the accidental depressurization of the RCS event is shown in Table III.12-1.

A comparison of the results from the limiting case analyzed for the MUR power uprate to those from the limiting case in the current licensing basis analysis is presented in Table III.12-2. From this comparison it can be seen that, despite the increase in power associated with the MUR, there was a benefit to the overall minimum DNBR calculated in the MUR analysis compared to the current licensing basis. The increase in the minimum DNBR can be primarily attributed to the difference in the moderator temperature coefficient modeling characteristics assumed in the current licensing basis analysis and those used in the MUR analysis.

III.12.6 Refer ences

III.12-1 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984.

III.12-2 WCAP-11397-P-A, “Revised Thermal Design Procedure,” April 1989.

Table III.12-1 Sequence of Events

Event	Time (Seconds)
Safety Valve Opens Fully	0.0
Low Pressurizer Pressure Trip Setpoint Reached	29.59
Rods Begin to Drop	31.59
Minimum DNBR Occurs	32.20

Table III.12-2 Comparison of Results to the Current Licensing Basis

Limiting Case Description	DNBR Safety Analysis Limit	Calculated Value
Limiting Licensing Basis Case – BWI SGs, 5% (Maximum) SGTP	[ ][a,c]	[ ][a,c]
Limiting MUR Case – D5 SGs, 10% (Maximum) SGTP, Minimum FW Temperature	[ ][a,c]	[ ][a,c]

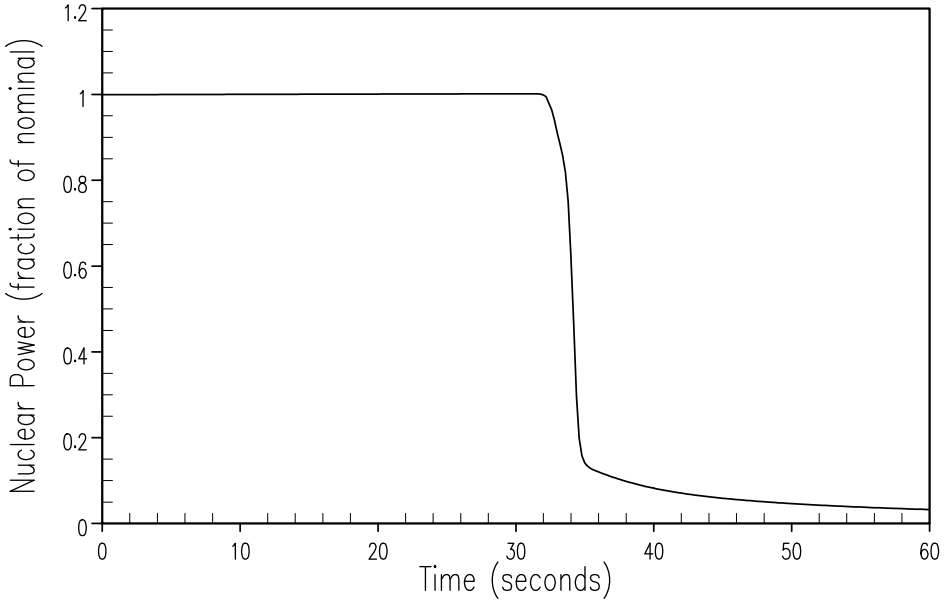

Figure III.12-1 Nuclear Power versus Time

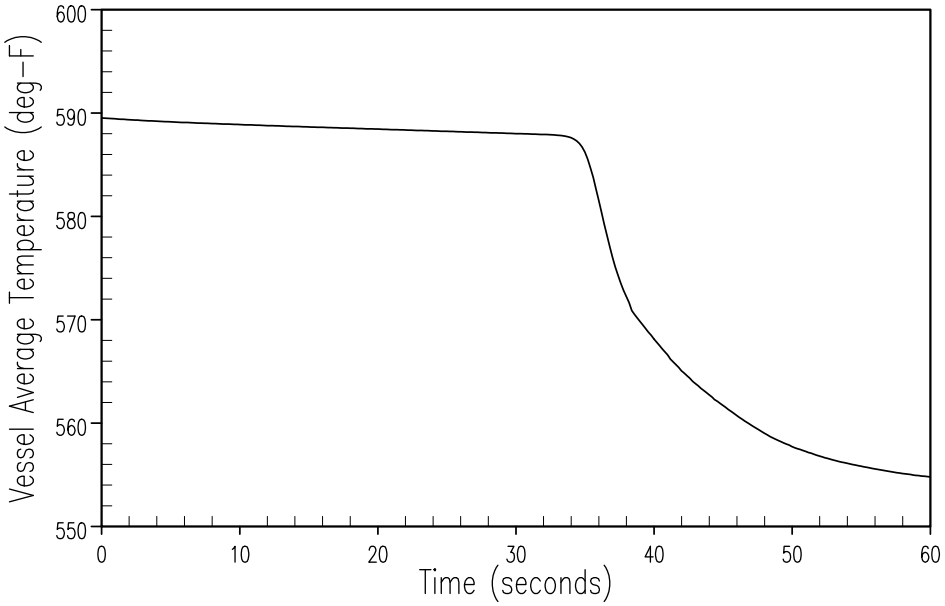

Figure III.12-2 Vessel Average Temperature versus Time

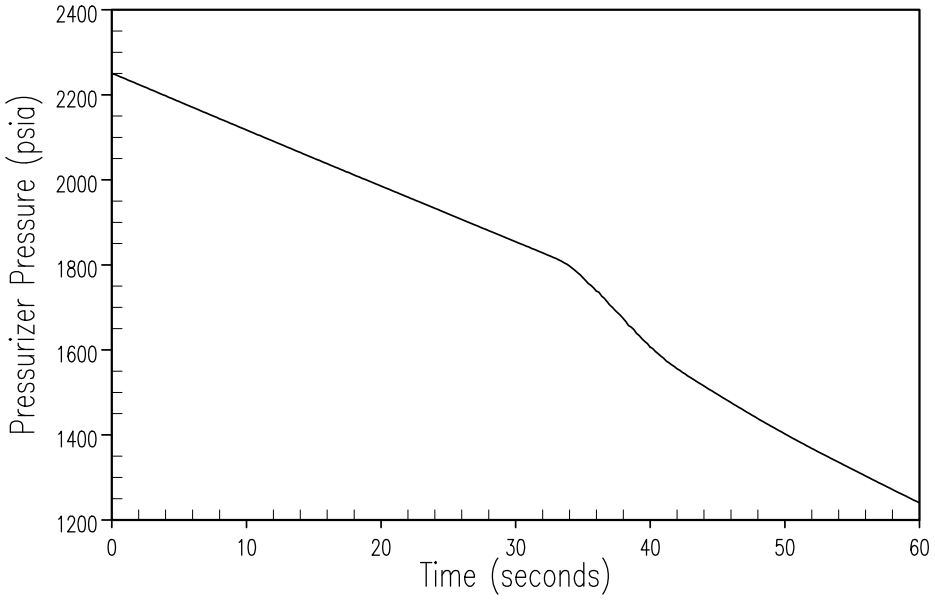

Figure III.12-3 Pressurizer Pressure versus Time

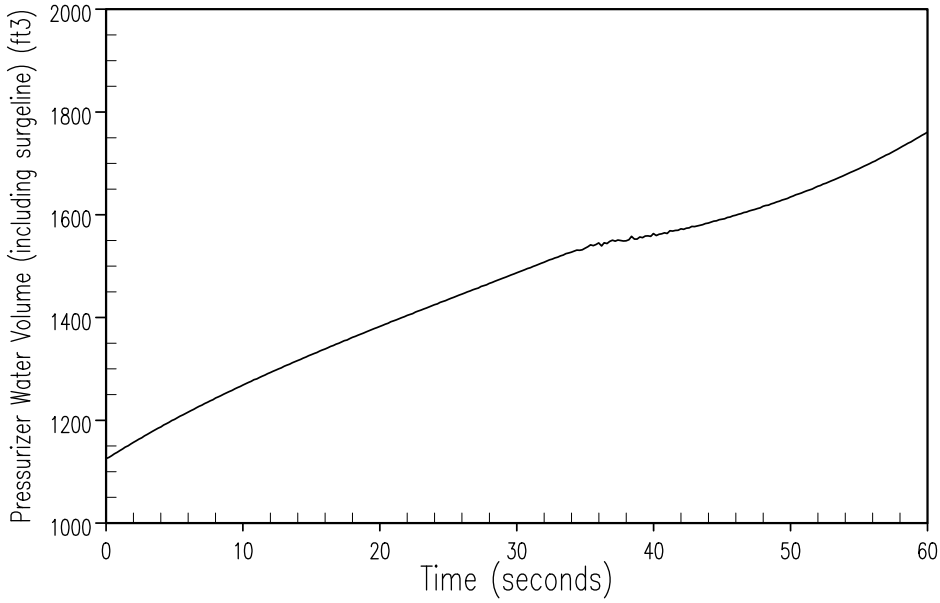

Figure III.12-4 Pressurizer Water Volume versus Time

a,c

Figure III.12-5 DNBR versus Time

III.13 Steam Generator Tube Rupture - UFSAR 15.6.3

A reanalysis of the Steam Generator Tube Rupture (SGTR) event was performed as the Margin to Overfill (MTO) results in the current analysis of record (AOR) are unacceptably small and revisions to the analysis assumptions are required. A detailed discussion of the SGTR and MTO Analysis is provided in Attachment 5a, "Steam Generator Tube Rupture and Margin to Overfill Analysis Report." A summary of the revised analysis is provided below.

The analysis addressed three major areas:

1. SGTR Margin to Steam Generator Overfill,
2. SGTR Thermal and Hydraulic Analysis for Radiological Consequences, and
3. SGTR Radiological Consequences

III.13.1 Margin to Steam Generator Overfill

III.13.1.1 Margin to Steam Generator Overfill Analysis

Analyses were performed to determine the margin to SG overfill for a design basis SGTR event for the Byron and Braidwood units. The SGTR MTO accident analysis demonstrated that SG overfill does not occur.

The analyses were performed using the LOFTTR2 program and the methodology developed in Reference III.13-1, with modifications to address NSAL-07-11 (Reference III.13-2) consistent with WCAP-16948-P (Reference III.13-3), and using plant-specific parameters. The MTO analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt. Therefore, the analyzed RTP power bounds the MUR power uprate conditions.

III.13.1.2 MTO Analysis Single Failure Assumptions

A single failure analysis was conducted for the SGTR MTO event to determine the most limiting single failure. This analysis is summarized in Attachment 5a, Sections I.1.E and II.2.E. It was determined that the most limiting failure regarding SG MTO was the failure of an intact SG PORV to open. It should be noted that the assumptions in this scenario necessitated installation of the plant modifications discussed below.

III.13.1.3 MTO Modifications

Byron and Braidwood Stations will be implementing the following plant modifications to support the Steam Generator Margin to Overfill Reanalysis single failure assumptions:

1. Install safety related air accumulator tanks to support AFW flow control,
2. Increase the capacity of the SG Power Operated Relief Valves (PORVs) (Unit 1 only)
3. Install Uninterruptible Power Supplies (UPS) on two of the four SG PORVs

4. Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

A description of these modifications is provided in Attachment 5a, Section II.2.F, "Modifications to Support MTO Single Failure Considerations." As noted above, these modifications will be installed and made operational prior to increasing power above the current licensed power level. The safety-related air accumulator tanks for AFW valve flow control, the UPS to the PORVs and the manual SI isolation valve are planned to be installed in accordance with 10 CFR 50.59; however, installation of the modification to increase the Unit 1 SG PORVs flow capacity requires NRC approval prior to installation as this modification results in more than a minimal increase in the accident dose.

The modification to install uninterruptible power supplies to the SG PORVs is prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a SGTR event. The NRC documented their position regarding these URIs in Reference III.13-4. The NRC verified that this same SGTR-related concern was also applicable to Braidwood Station as documented in Reference III.13-6. Byron Station responded to the NRC in Reference III.13-5; and Braidwood Station responded to the NRC in Reference III.13-7. In these letters, both Byron Station and Braidwood Station committed to installing the UPS modification to resolve the single failure concern. This modification places the SGTR analysis in compliance with NRC regulations and preserves the assumption in the SGTR analysis.

III.13.2 Thermal and Hydraulic Analysis for Radiological Consequences

The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in References III.13-1 and III.13-8, and using the plant-specific parameters. From these predictions, the RCS and SG water masses, the ruptured SG break flow, the fraction of this break flow that flashes directly to steam, and the steam releases from the ruptured and intact SGs through the MSSVs and PORVs are calculated for input to the dose analyses. The thermal-hydraulic analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt to generate this data. Therefore, the analyzed power bounds the MUR power uprate.

III.13.3 Radiological Consequences Analysis

The steam generator tube rupture radiological analyses are based upon the alternative source term (AST) as defined in Regulatory Guide (RG) 1.183, with acceptance criteria as specified in RG 1.183 for offsite doses and in 10 CFR 50.67 for the control room. The analyses involve the transfer of activity from the primary to the secondary side of the SGs and then to the environment. The RCS iodine and noble gas source terms are scaled to the Technical Specification Dose Equivalent Iodine-131 and Xenon-133 limits in the primary coolant, which removes the power dependence from the analysis. The various parameters from the thermal-hydraulic analyses are consistent with a core power of 3658.3 MWt, or 102% of 3586.6 MWt. The resulting doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and in the control room remain within the applicable limits as shown in Attachment 5a, Table IV-6; therefore, the results of the SGTR radiological analyses are acceptable under MUR power uprate conditions.

III.13.4 Refe rences

III.13-1 Westinghouse Report WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987

III.13-2 NSAL-07-11, "Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology," November 2007.

III.13-3 WCAP-16948-P, "Clarifications for the Westinghouse Steam Generator Tube Rupture Margin to Overfill Analysis Methodology," December 2008.

III.13-4 Letter from Steven A. Reynolds (USNRC) to Michael J. Pacilio (Exelon Generation Company, LLC), "Byron Station, Units 1 and 2 Follow Up Inspection of an Unresolved Item; 05000454/201 1010; 050004552011010," dated January 19, 2011.

III.13-5 Byron Station responded to the NRC in a letter from Timothy J. Tulon (Exelon Generation Company, LLC) to the USNRC, "Response to NRC Follow Up Inspection Report; 05000454/2011010; 05000455/2011010," dated February 18, 2011.

III.13-6 Letter from Steven A. Reynolds (USNRC) to Michael J. Pacilio (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 Verification Inspection Related to Analysis of Steam Generator Tube Rupture Event Margin to Overfill; 05000456/2011009; 050004572011009," dated February 1, 2011

III.13-7 Letter from Daniel J. Enright (Exelon Generation Company, LLC) to the USNRC, "Response to NRC Verification Inspection Report; 05000456/2011009; 050004572011009," dated March 2, 2011

III.13-8 Westinghouse Report Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," March 1986.

III.14 Anticipated Transient without Scram – UFSAR 15.8

III.14.1 Identification of Causes and Accident Description

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (such as a loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the reactor trip system to shut down the reactor. In the worst case an unmitigated ATWS might result in RCS system pressure that compromises the integrity of the RCS system.

The final ATWS rule, 10CFR50.62(c) (Reference III.14-1), requires Westinghouse-designed pressurized water reactors to incorporate a device diverse from the reactor trip system that automatically actuates the auxiliary feedwater system (AFW) and initiates a turbine trip for conditions indicative of an ATWS. The installation of the NRC-approved ATWS Mitigating System (AMS), described in UFSAR Section 7.7.1.21, satisfies this final ATWS rule.

III.14.2 Method of Analysis

As noted above, the final ATWS Rule, 10CFR50.62(c)(1) (Reference III.14-1), requires the incorporation of a diverse (from the reactor trip system) actuation of the AFW system and turbine trip for Westinghouse-designed plants. The installation of the NRC-approved AMS satisfies this final ATWS Rule. However, it must also be demonstrated that the deterministic ATWS analyses that form the basis for this rule and the AMS design remain valid for the plant. This is typically done by confirming that the analyses documented in NS-TMA-2182 (Reference III.14-2) remain valid or by performing new deterministic analyses for the proposed plant state.

To address the MUR program, the Loss of Load (LOL) and Loss of Normal Feedwater (LONF) ATWS events were reanalyzed to ensure that the analytical basis for the final ATWS rule continues to be met. The LOL and LONF ATWS events are the two most limiting RCS overpressure transients reported in NS-TMA-2182. The approach taken was to demonstrate that the ATWS unfavorable exposure time (UET) is less than 5% of an operating cycle. UET is the duration of a given cycle for which the core reactivity feedback is insufficient to preclude the RCS pressure from exceeding the ASME (Reference III.14-3) Service Level C pressure limit of 3200 psig following an ATWS event. The objective is to show that the ATWS pressure limit of 3200 psig is met for at least 95% of the cycle, and therefore the analytical basis for the final ATWS rule continues to be met.

The UET approach has been previously approved by the NRC (Reference III.14-4). The analysis must show that the UET, given the cycle design (including MTC), will be less than 5%. This 5% requirement for the UET is equivalent to the probability level in the reference analyses for the ATWS rule analytical basis (Reference III.14-2). In those analyses, the NRC required that all parameters be best-estimate values with the exception of the MTC initial condition, which is to be a full power value that is bounding for at least 95% of a given cycle. The UET approach provides a similar level of assurance for the effectiveness of the reactivity feedback.

To determine UET, the reactivity conditions of the core and plant conditions under consideration must be compared to the ATWS analysis conditions that lead to a peak RCS pressure at the ATWS pressure limit of 3200 psig. The variable conditions of significance to the resulting peak RCS pressure following the

LOL and LONF ATWS events are total reactivity feedback (primarily MTC), primary-side pressure relief capacity, and auxiliary feedwater capacity. For a given primary-side pressure relief configuration and auxiliary feedwater capacity, reactivity feedback (MTC) can be adjusted in the ATWS analysis until the peak RCS pressure during the specific ATWS event equals 3200 psig. At these specific reactivity feedback conditions, the change in power with increasing temperature represents what is defined as the Critical Power Trajectory (or heatup/shutdown characteristics) for the specific plant configuration. The heatup/shutdown characteristics of a given core at various times in the cycle can then be compared to the Critical Power Trajectory (CPT) to establish UET for the given core at the specific plant configuration conditions.

III.14.3 Analysis Inputs and Assumptions

The Loss of Normal Feedwater (LONF) and Loss of Load (LOL) ATWS events are the two most limiting RCS overpressure transients documented in Reference III.14-2; thus, these two events were analyzed for the MUR power uprate program. Byron 1 and Braidwood 1 have BWI steam generators and Byron 2 and Braidwood 2 have D5 steam generators; therefore, CPTs were generated for each steam generator type for use in determining the UET. The following analysis assumptions were used:

- Consistent with the analysis basis for the final ATWS Rule (NS-TMA-2182, Reference III.14-2):
  - Thermal Design Flow (TDF) is assumed. No uncertainties are applied to the initial power, RCS average temperature or RCS pressure.
  - 0% steam generator tube plugging (SGTP) is assumed. 0% SGTP is more limiting (i.e., results in a higher peak RCS pressure) for ATWS events.
  - Control rod insertion was not assumed.
  - 100% pressurizer power-operated relief valve capacity was assumed.
  - The analyses model a turbine trip and actuation of the AFW system as a result of an AMS signal on low steam generator water level.
- An AMS timer delay of 10 seconds was modeled.
- A 2.5 seconds delay from AMS signal on low steam generator water level to turbine trip was modeled.
- A 55 seconds delay from AMS signal on low steam generator water level until full AFW flow is reached was modeled
- A best-estimate AFW flow of 1223.2 gpm was assumed.
- The reactivity feedback (MTC) was adjusted until the peak RCS pressure during the specific ATWS event equaled 3200 psig.

III.14.4 Analysis Acceptance Criteria

To remain compliant with the basis of the final ATWS rule (10 CFR 50.62), the UET calculated for the ATWS reference conditions (no control rod insertion, nominal auxiliary feedwater flow and unblocked pressurizer power-relief valves) must be less than 5% for a given cycle.

III.14.5 Analysis Results

CPT curves for the LOL and LONF ATWS transients were generated at MUR conditions for both Byron 1 and Braidwood 1 with BWI steam generators and Byron 2 and Braidwood 2 with D5 steam generators. These CPTs are presented in tabular form in Tables III.14-1 through III.14-4.

To remain compliant with the basis of the final ATWS rule (10CFR50.62), the UET must be less than 5% for a given cycle, or equivalently, the ATWS pressure limit of 3200 psig must be met for 95% of the cycle. The UET is met for the anticipated operating conditions with a representative core design and will be checked on a cycle-specific basis; thus, the basis of the final ATW rule (10CFR50.62) is met for the MUR power uprate program.

III.14.6 Refer ences

III.14-1 10 CFR 50.62 and Supplementary Information Package, “Requirements for Reduction of Risk from ATWS Events for Light Water-Cooled Nuclear Power Plants.”

III.14-2 NS-TMA-2182, “Anticipated Transients Without Scram for Westinghouse Plants,” December 1979.

III.14-3 ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers.

III.14-4 NRC letter to D. L. Farrar (Commonwealth Edison), “Issuance of Amendments (TAC NOS. M89092, M89093, M89072 and M89091),” July 27, 1995.

Table III.14-1 Comparison of LOL ATWS Critical Power Trajectory (CPT) Data Byron and Braidwood Stations Unit 1 with Model BWI Steam Generators

T_{in} (°F)	NSSS Power (fraction of initial)	
	Current CPT Data	MUR CPT Data
554.9[1]	1.0	1.0
580	0.887	0.894
600	0.720	0.735
610	-	0.644
620	0.530	0.545
630	-	0.438
634	0.385	-
640	0.319	0.324
650	0.201	0.201
660	0.070	0.063

Note:
1. Note that the initial T_{in} in the current analysis was 555.4°F

Table III.14-2 Comparison of LONF ATWS Critical Power Trajectory (CPT) Data Byron and Braidwood Stations Unit 1 with Model BWI Steam Generators

T_{in} (°F)	NSSS Power (fraction of initial)	
	Current CPT Data	MUR CPT Data
554.9[1]	1.0	1.0
580	0.895	0.893
600	0.741	0.732
610	-	0.640
620	0.560	0.541
630	-	0.433
634	0.421	-
640	0.356	0.320
650	0.242	0.196

Note:
1. Note that the initial T_{in} in the current analysis was 555.4°F

Table III.14-3 Comparison of LOL ATWS Critical Power Trajectory (CPT) Data Byron and Braidwood Stations Unit 2 with Model D5 Steam Generators

T_{in} (°F)	NSSS Power (fraction of initial)	
	Current CPT Data	MUR CPT Data
554.9[1]	1.0	1.0
580	0.951	0.998
600	0.879	0.956
610	-	0.906
620	0.764	0.835
630	-	0.749
634	0.652	-
640	0.596	0.646
650	0.499	0.530
660	0.384	0.396

Note:
1. Note that the initial T_{in} in the current analysis was 555.4°F.

Table III.14-4 Comparison of LONF ATWS Critical Power Trajectory (CPT) Data Byron and Braidwood Stations Unit 2 with Model D5 Steam Generators

T_{in} (°F)	NSSS Power (fraction of initial)	
	Current CPT Data	MUR CPT Data
554.9[1]	1.0	1.0
580	0.954	0.943
600	0.885	0.839
610	-	0.769
620	0.772	0.687
630	-	0.590
634	0.662	-
640	0.606	0.484
650	0.509	0.366
660	0.395	0.237

Note:
1. Note that the initial T_{in} in the current analysis was 555.4°F.

III.15 LOCA Long Term Mass and Energy Release and Containment Response – UFSAR 6.2.1.3.1

III.15.1 Identification of Causes and Accident Description

A Loss-of-Coolant-Accident (LOCA) will result in release of steam and water into the containment. This will result in increases in the local subcompartment pressures and an increase in the global containment pressure and temperature.

The long-term LOCA mass and energy (M&E) release and the containment integrity analyses were reanalyzed for Byron and Braidwood Stations, Units 1 and 2, taking into consideration the MUR power uprate conditions and the recently identified inconsistency in the instantaneous mass and energy release values in the EPITOME code. In addition, the following revisions have been incorporated into the reanalysis:

- Increase in the metal mass of the lower core support plate to the input modification program (IMP) database,
- Corrections to the reactor coolant pump (RCP) homologous curves,
- Incorporation of containment spray termination at 8 hours post-LOCA, and
- Correction to SATIMP preprocessor modeling to include the barrel baffle metal mass for upflow design plants (Unit 2 only).

For the double-ended hot leg (DEHL) break, generic studies (Reference III.15-3) have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore only the mass and energy releases for the DEHL break blowdown phase have been reanalyzed.

The double-ended pump suction (DEPS) break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators (SGs). As a result, the DEPS break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment. The blowdown, reflood and post reflood for the DEPS break has been reanalyzed.

The cold leg break location has also been identified in previous studies (Reference III.15-3) to be much less limiting in terms of the overall containment energy release. These studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the DEPS break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is bounded by other breaks and has not been reanalyzed.

III.15.2 Method of Analysis

The mass and energy releases and the containment response analysis were generated using NRC approved methodologies as described in References III.15-3 through 7. The mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN78, WREFLOOD10325, FROTH, and EPITOME. Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO (Reference III.15-8). These methodologies

were previously utilized and approved for Byron and Braidwood Stations, Units 1 and 2 (Reference III.15-2).

III.15.3 Analysis Inputs and Assumptions

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment.

1. Maximum expected operating temperature of the reactor coolant system (RCS) (100% full power conditions). The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation.
2. Allowance for RCS temperature uncertainty (+9.1°F).
3. An initial RCS pressure based on a nominal value of 2250 psia plus an allowance which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since the RCS rapidly depressurizes from this value until it equilibrates with containment pressure.
4. Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty).
5. Core rated power of 3658.3 MWt (102% of 3586.6 MWt) which bounds the MUR power level.
6. Conservative heat transfer coefficient (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer).
7. Allowance in core stored energy that is based on a statistical combination of effects including fuel type, power level, manufacturing tolerances, densification and burn-up.
8. The SG metal mass was modeled to include only the portion of the SGs which is in contact with the fluid on the secondary side. Portions of the SGs such as the head, upper shell and miscellaneous upper internals have poor heat transfer due to their location above the operating level. The heat that is stored in this region is unavailable for release to containment and will not be able to effectively transfer energy to the RCS in the first 3,600 seconds. Thus this energy will be removed at a much slower rate and longer time period (>10,000 seconds).
9. An allowance for RCS initial pressure uncertainty (+43 psi).
10. A maximum containment backpressure equal to design pressure (50 psig).
11. A minimum RCS loop flow (92,000 gpm/loop).
12. A uniform steam generator tube plugging level of 0% which,
    - Maximizes reactor coolant volume and fluid release,
    - Maximizes heat transfer area across the SG tubes, and
    - Reduces coolant loop resistance, which reduces the Δp upstream of the break for the pump suction break cases and increases break flow.

A summary of the key parameters used in the LOCA M&E analysis and the containment response analysis are provided in Tables III.15-1 and 2, respectively.

Regarding safety injection flow, the mass and energy release analysis considered configurations/failures to conservatively bound respective alignments. The Minimum Safeguards Case (one Charging (CV) pump, one High Head Safety Injection (SI) pump, and one Low Head Safety Injection (RHR) pump) was analyzed. The Maximum Safeguards case, (two CV pumps, two SI pumps, and two RHR pumps) was previously shown to be considerably less limiting than the Minimum Safeguards case (Reference III.15-2). Therefore the Maximum Safeguards case was not reanalyzed.

For the containment response analysis, the Minimum Safeguards case was based upon a diesel train failure leaving only one containment spray (CS) train and 2 RCFCs available as active heat removal systems. Due to the duration of the DEHL break transient (i.e., blowdown only), no containment safeguards equipment is modeled in that analysis.

The American Nuclear Society (ANS) Standard 5.1 (Reference III.15-9) was used in the LOCA mass and energy release model for the determination of decay heat energy.

Unit 1 at each site has Babcock and Wilcox (B&W) replacement SGs, whereas Unit 2 at each site has Westinghouse model D5 SGs. Separate analytical models were generated for each steam generator type and were used for the calculations.

III.15.4 Analysis Acceptance Criteria

A large break LOCA is classified as an ANS Condition IV event (an infrequent fault). To satisfy the NRC acceptance criteria presented in the Standard Review Plan Section 6.2.1.3 (Reference III.15-10), the relevant requirements are as follows:

a. 10 CFR 50, Appendix A

b. 10 CFR 50, Appendix K, paragraph I.A

In order to meet these requirements, the following must be addressed.

1. Sources of Energy
2. Break Size and Location
3. Calculation of Each Phase of the Accident

The purpose of the containment response analysis is to demonstrate the acceptability of the containment safeguards systems to mitigate the consequences of a LOCA inside containment such that the containment pressure and temperature remain below the design limits at the MUR uprated conditions. With respect to post LOCA long term containment transient environmental conditions (pressure and temperature), equipment design and licensing criteria (e.g., qualified operating life) must be conservatively bounded.

III.15.5 Analysis Results

The consideration of the various energy sources in the long-term mass and energy release analysis, including the B&W replacement SGs and the Westinghouse D5 SGs, provide assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

The LOCA containment response analyses performed as part of the Byron and Braidwood Stations, Units 1 and 2 MUR program resulted in peak containment pressures less than the containment design pressure of 50 psig and below the Technical Specification controlled Integrated Leak Rate Test (ILRT) pressure (P_a) for Units 1 and Units 2 of 42.8 / 38.4 psig respectively. The post-LOCA containment peak air temperatures are below the containment design temperature (280°F). A comparison of the reanalysis peak containment pressures and temperatures for MUR power uprate to those from the existing analysis of record (AOR) is presented in Table III.15-3. The long-term pressures are well below 50% of the peak value within 24 hours. Based on these results assuming MUR conditions the applicable LOCA criteria of peak pressure and long term depressurization have been met.

For the Byron and Braidwood Stations, Unit 1, DEHL break blowdown reanalysis Tables III.15-4 through III.15-6 provide the sequence of events and the mass and energy balances, respectively. The resultant mass and energy release data were input into the containment response model: Figures III.15-1 and 2 show the resultant containment pressure and temperature transient curves, respectively.

For the Byron and Braidwood Stations, Unit 1, DEPS break reanalysis Tables III.15-7 through III.15-9 provide the sequence of events and the mass and energy balances, respectively. The resultant mass and energy release data were input into the containment response model: Figures III.15-3 and 4 show the resultant containment pressure and temperature transient curves, respectively.

For the Byron and Braidwood Stations, Unit 2, DEHL break blowdown reanalysis Tables III.15-10 through III.15-12 provide the sequence of events and the mass and energy balances, respectively. The resultant mass and energy release data were input into the containment response model: Figures III.15-5 and 6 show the resultant containment pressure and temperature transient curves, respectively.

For the Byron and Braidwood Stations, Unit 2, DEPS break reanalysis Tables III.15-13 through III.15-15 provide the sequence of events and the mass and energy balances, respectively. The resultant mass and energy release data were input into the containment response model: Figures III.16-7 and 8 show the resultant containment pressure and temperature transient curves, respectively.

New EQ pressure and temperature profiles were developed. The effect of the revised containment conditions on the environmental qualification of electrical equipment in containment is discussed in Section V.1.C.iii.

III.15.6 Refer ences

III.15-1 "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Nonproprietary).

III.15-2 Letter from G. F. Dick (USNRC) to O. D. Kingsley (Exelon), “Issuance of Amendments: Increase in Reactor Power, Byron Station Units 1 and 2, Braidwood Station, Units 1 and 2.” (TAC NOS. MA9428, MA9429, MA9426 and MA9427), May 4, 2001 [Accession No. ML011420274]

III.15-3 Letter from Herbert N. Berkow, Director (NRC) to James A. Gresham (Westinghouse), “Acceptance of Clarifications of Topical Report WCAP-10325-P-A, ‘Westinghouse LOCA Mass and Energy Release Model for Containment Design – March 1979 Version’ (TAC No. MC7980),” October 18, 2005 [Accession No. ML052660242].

III.15-4 "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Nonproprietary).

III.15-5 Docket No. 50-315, “Amendment No. 126, To Facility Operating License No. DPR-58 (TAC No. 71062), for D. C. Cook Nuclear Plant Unit 1,” June 9, 1989 [Accession No. ML021050051].

III.15-6 EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

III.15-7 Letter from Mr. Charles E Rossi (NRC) to Mr. William J. Johnson (W), “Acceptance for Referencing of Licensing Topical Report WCAP-10325, ‘Westinghouse LOCA Mass and Energy Release Model for Containment Design (Proprietary) - March 1979 Version’,” February 17, 1987 (Copy in Reference III.15-4).

III.15-8 "Containment Pressure Analysis Code (COCO)," WCAP-8327, July, 1974 (Proprietary), WCAP-8326, July, 1974 (Non-Proprietary).

III.15-9 ANSI/ANS-5.1 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

III.15-10 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", LWR Edition, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Chapter 6, Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs),” Revision 1, July 1981 [Accession No. ML053560191].

Table III.15-1 System Parameters Initial Conditions for Measurement Uncertainty Recapture (MUR) Uprate

Parameters	Value	
	B&W Replacement SG (Unit 1)	Westinghouse D5 – SG (Unit 2)
Core Thermal Power (MWt) including the Calorimetric Error	3658.3	3658.3
Reactor Coolant System Total Flowrate (lbm/sec)	37,590.0	37,590.0
Vessel Outlet Temperature (°F)	630.0	630.0
Core Inlet Temperature (°F)	564.2	564.2
Vessel Average Temperature (°F)	597.1	597.1
Initial Steam Generator Steam Pressure (psia)	1055.0	967.0
Steam Generator Tube Plugging (%)	0	0
Initial Steam Generator Secondary Side Mass (lbm)	136,617.8	106,484.0
Assumed Maximum Containment Backpressure (psia)	64.7	64.7
Accumulator		
Water Volume (ft^3) per accumulator	1005.2	1015.4
N_2 Cover Gas Pressure (psia)	661.7	661.7
Temperature (°F)	120.0	120.0
Safety Injection Delay, total (sec) (from beginning of event)	40.0	40.0

Note:
Core Thermal Power, RCS Total Flowrate, RCS Coolant Temperatures, and SG Secondary Side Mass include appropriate uncertainty and/or allowance.

Table III.15-2 LOCA Containment Response Analysis Parameters Measurement Uncertainty Recapture (MUR) Uprate Conditions

Service water temperature (°F)	100
RWST water temperature (°F)	105
Initial containment temperature (°F)	120
Initial containment pressure (psia)	15.7
Initial relative humidity (%)	20
Net free volume (ft^3)	2.758×10^6
Reactor Containment Fan Coolers	
Total	4
Analysis maximum	4
Analysis minimum	2
Containment Hi-1 setpoint (psig)	6.8
Delay time (sec) With Offsite Power Without Offsite Power	 27.0 65.0
Containment Spray Pumps	
Total	2
Analysis maximum	1
Analysis minimum	1
Flowrate (gpm) Injection phase (per pump) Recirculation phase (total)	 3285 3285
Containment Hi-3 setpoint (psig)	24.8
Delay time (sec) With Offsite Power (delay after High High setpoint) Without Offsite Power (total time from t=0)	 75.2 110.2
ECCS Recirculation Switchover (sec) Minimum Safeguards Maximum Safeguards	 1110. 695.
Containment Spray Recirculation Switchover, (sec) Minimum Safeguards Maximum Safeguards	 3778 3363

Table III.15-2 *(cont.)* **LOCA Containment Response Analysis Parameters Measurement Uncertainty Recapture (MUR) Uprate Conditions**

Emergency Core Cooling System (ECCS) Flows (GPM)	
Minimum ECCS Injection alignment Recirculation alignment	 5686 994.2
Maximum ECCS Injection alignment Recirculation alignment	 12,305 11,917.1
Residual Heat Removal System Heat Exchanger	
Modeled in analysis	1
Recirculation switchover time (sec) Minimum Safeguards* Maximum Safeguards	 1110 695
UA, 10^6 * BTU/hr-°F	2.16
Flows – Tube Side and Shell Side (gpm) Minimum Safeguards* Maximum Safeguards	 5000 5000
Component Cooling Water Heat Exchangers	
Modeled in analysis	1
UA, 10^6 * BTU/hr-°F	4.73
Flows – Shell Side and Tube Side (gpm) Shellside* Tubeside* (service water)	 5000 5000
Additional heat loads, (BTU/hr)	6.8×10^6

Note:
* Minimum safeguard data representing 1 EDG

Table III.15-3 LOCA Containment Response Results (Loss of Offsite Power Assumed)

Case		Peak Pressure (psig @ seconds)		Peak Steam Temp. (°F @ seconds)		Pressure (psig @ 24 hours)		Steam Temperature (°F @ 24 hours)	
		AOR	**MUR**	**AOR**	**MUR**	**AOR**	**MUR**	**AOR**	**MUR**
Unit 1	DEHL MIN SI	42.77 @ 22.116	42.6 @ 22.03	264.50 @ 22.116	264.24 @ 22.03	NA	NA	NA	NA
	DEPS MIN SI	41.84 @ 460.56	41.49 @ 1,599	262.30 @ 460.56	261.69 @ 23.2	9.074	10.23	174	178.33
Unit 2	DEHL MIN SI	38.36 @ 21.079	38.26 @ 20.12	257.57 @ 21.079	257.4 @ 20.12	NA	NA	NA	NA
	DEPS MIN SI	37.71 @ 399	38.23 @ 659.39	255.65 @ 399	256.45 @ 659.39	8.68	10.13	170.35	177.96

Table III.15-4 Double-Ended Hot Leg Break - Sequence of Events
Unit 1 with B&W Replacement Steam Generators
Minimum Safeguards at MUR Conditions

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
1.16	Containment HI-1 Pressure Setpoint Reached
3.8	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached
6.64	Containment HI-3 Pressure Setpoint Reached
15.2	Broken Loop Accumulator Begins Injecting Water
15.4	Intact Loops Accumulator Begins Injecting Water
22.02	Peak Pressure and Temperature Occur
24.2	End of Blowdown Phase
30.0	Transient Modeling Terminated

Table III.15-5 Double-Ended Hot Leg Break - Mass Balance
Unit 1 with B &W Replacement Steam Generators
Minimum Safeguards at MUR Conditions

Time (seconds)		0.00	24.20	24.20
		Mass (thousand lbm)		
Initial	In RCS and Accumulators	817.62	817.62	817.62
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
***** Total Available *****		817.62	817.62	817.62
Distribution	Reactor Coolant	568.75	67.65	105.06
	Accumulators	248.87	199.20	161.79
	Total Contents	817.62	266.85	266.85
Effluent	Break Flow	0.00	550.75	550.75
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	550.75	550.75
***** Total Accountable *****		817.62	817.60	817.60

Table III.15-6 Double-Ended Hot Leg Break - Energy Balance
Unit 1 with B &W Replacement Steam Generators
Minimum Safeguards at MUR Conditions

Time (seconds)		0.00	24.20	24.20
		Energy (millions BTU)		
Initial Energy	In RCS, Accumulators and SGs	964.82	964.82	964.82
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	8.64	8.64
	Heat From Secondary	0.00	-1.79	-1.79
	Total Added	0.00	6.85	6.85
***** Total Available *****		964.82	971.67	971.67
Distribution	Reactor Coolant	341.60	18.30	22.02
	Accumulators	22.27	17.83	14.11
	Core Stored	23.37	9.32	9.32
	Primary Metal	172.70	162.04	162.04
	Secondary Metal	92.77	90.99	90.99
	Steam Generators	312.11	304.75	304.75
	Total Contents	964.82	603.21	603.21
Effluent	Break Flow	0.00	367.85	367.85
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	367.85	367.85
***** Total Accountable *****		964.82	971.07	971.07

Table III.15-7 Double-Ended Pump Suction Break - Sequence of Events
Unit 1 with B&W Replacement Steam Generators
Minimum Safeguards at MUR Conditions

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
1.18	Containment HI-1 Pressure Setpoint Reached
4.5	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached (Safety Injection Begins Coincident with Low Pressurizer Pressure SI Setpoint)
7.8	Containment HI-3 Pressure Setpoint Reached
18.0	Broken Loop Accumulator Begins Injecting Water
18.4	Intact Loops Accumulator Begins Injecting Water
23.2	Peak Temperature Occurs
27.0	End of Blowdown Phase
43.6	Safety Injection Begins
65.0	Reactor Containment Fan Coolers Actuate
68.66	Broken Loop Accumulator Water Injection Ends
69.71	Intact Loops Accumulator Water Injection Ends
110.2	Containment Spray Pump(s) (RWST) Start
194.7	End of Reflood for MIN SI Case
1110	RHR/SI/CV Alignment for Recirculation
1599	Peak Pressure Occurs
3778	Containment Spray is Aligned for Recirculation
28,800	Containment Spray is Terminated
2.592×10^6	Transient Modeling Terminated

Table III.15-8 Double-Ended Pump Suction Break - Mass Balance
Unit 1 with B&W Replacement Steam Generators
Minimum Safeguards at MUR Conditions

		Mass Balance						
Time (seconds)		0.00	27.00	27.00	194.71	799.13	1591.26	3600.00
		Mass (thousand lbm)						
Initial	In RCS and Accumulators	817.62	817.62	817.62	817.62	817.62	817.62	817.62
Added Mass	Pumped Injection	0.00	0.00	0.00	98.20	513.58	791.83	1061.32
	Total Added	0.00	0.00	0.00	98.20	513.58	791.83	1061.32
***** Total Available *****		817.62	817.62	817.62	915.82	1331.20	1609.45	1878.94
Distribution	Reactor Coolant	568.75	47.75	77.38	140.24	140.24	140.24	140.24
	Accumulators	248.87	196.96	167.32	0.00	0.00	0.00	0.00
	Total Contents	817.62	244.71	244.71	140.24	140.24	140.24	140.24
Effluent	Break Flow	0.00	572.90	572.90	763.90	1179.28	1508.24	1777.73
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	572.90	572.90	763.90	1179.28	1508.24	1777.73
***** Total Accountable *****		817.62	817.60	817.60	904.14	1319.52	1648.48	1917.98

Table III.15-9 Double-Ended Pump Suction Break - Energy Balance
Unit 1 with B&W Replacement Steam Generators
Minimum Safeguards at MUR Conditions

		Energy Balance						
Time (seconds)		0.00	27.00	27.00	194.71	799.13	1591.26	3600.00
		Energy (million BTU)						
Initial Energy	In RCS, Accumulators and SG	964.81	964.81	964.81	964.81	964.81	964.81	964.81
Added Energy	Pumped Injection	0.00	0.00	0.00	7.17	37.50	60.02	88.77
	Decay Heat	0.00	8.82	8.82	29.68	85.21	143.90	263.64
	Heat From Secondary	0.00	15.46	15.46	15.46	15.46	15.46	15.46
	Total Added	0.00	24.28	24.28	52.31	138.17	219.38	367.88
***** Total Available *****		964.81	989.09	989.09	1017.12	1102.98	1184.19	1332.68
Distribution	Reactor Coolant	341.60	11.06	13.71	37.61	37.61	37.61	37.61
	Accumulators	22.27	17.63	14.97	0.00	0.00	0.00	0.00
	Core Stored	23.36	12.19	12.19	4.91	4.71	4.32	3.33
	Primary Metal	172.70	163.99	163.99	137.24	95.44	72.72	53.68
	Secondary Metal	92.77	91.85	91.85	84.05	62.91	44.53	32.20
	Steam Generators	312.11	327.83	327.83	296.17	214.74	148.06	105.95
	Total Contents	964.81	624.55	624.55	559.99	415.41	307.23	232.77
Effluent	Break Flow	0.00	363.96	363.96	447.26	677.70	890.64	1115.58
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	363.96	363.96	447.26	677.70	890.64	1115.58
***** Total Accountable *****		964.81	988.50	988.50	1007.25	1093.11	1197.88	1348.35

Table III.15-10 Double-Ended Hot Leg Break - Sequence of Events
Unit 2 with Westinghouse D5 Steam Generators
Minimum Safeguards at MUR Conditions

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
1.18	Containment HI-1 Pressure Setpoint Reached
3.8	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached
7.86	Containment HI-3 Pressure Setpoint Reached
13.3	Broken Loop Accumulator Begins Injecting Water
13.5	Intact Loops Accumulator Begins Injecting Water
20.12	Peak Pressure and Temperature Occur
26.2	End of Blowdown Phase
30.0	Transient Modeling Terminated

Table III.15-11 Double-Ended Hot Leg Break - Mass Balance
Unit 2 with Westinghouse D5 Steam Generators
Minimum Safeguards at MUR Conditions

Time (seconds)		0.00	26.20	26.20
		Mass (thousand lbm)		
Initial	In RCS and Accumulators	761.59	761.59	761.59
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
***** Total Available *****		761.59	761.59	761.59
Distribution	Reactor Coolant	512.72	92.11	129.52
	Accumulators	248.87	174.27	136.86
	Total Contents	761.59	266.38	266.38
Effluent	Break Flow	0.00	495.18	495.18
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	495.18	495.18
***** Total Accountable *****		761.59	761.56	761.56

Table III.15-12 Double-Ended Hot Leg Break - Energy Balance
Unit 2 with Westinghouse D5 Steam Generators
Minimum Safeguards at MUR Conditions

Time (seconds)		0.00	26.20	26.20
		Energy (million BTU)		
Initial Energy	In RCS, Accumulators and SGs	832.59	832.59	832.59
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	8.96	8.96
	Heat From Secondary	0.00	-0.67	-0.67
	Total Added	0.00	8.29	8.29
***** Total Available *****		832.59	840.88	840.88
Distribution	Reactor Coolant	309.13	21.94	25.66
	Accumulators	22.27	15.60	11.88
	Core Stored	22.71	8.73	8.73
	Primary Metal	156.71	146.69	146.69
	Secondary Metal	74.44	73.78	73.78
	Steam Generators	247.33	244.45	244.45
	Total Contents	832.59	511.19	511.19
Effluent	Break Flow	0.00	329.10	329.10
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	329.10	329.10
***** Total Accountable *****		832.59	840.28	840.28

Table III.15-13 Double-Ended Pump Suction Break - Sequence of Events
Unit 2 with Westinghouse D5 Steam Generators
Minimum Safeguards at MUR Conditions

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed
1.17	Containment HI-1 Pressure Setpoint Reached
4.4	Low Pressurizer Pressure SI Setpoint = 1715 psia Reached (Safety Injection Begins coincident with Low Pressurizer Pressure SI Setpoint)
7.77	Containment HI-3 Pressure Setpoint Reached
15.5	Broken Loop Accumulator Begins Injecting Water
15.7	Intact Loops Accumulator Begins Injecting Water
25.6	End of Blowdown Phase
44.4	Safety Injection Begins
65.0	Reactor Containment Fan Coolers Actuate
66.8	Broken Loop Accumulator Water Injection Ends
70.06	Intact Loops Accumulator Water Injection Ends
110.2	Containment Spray Pump(s) (RWST) Start
194.46	End of Reflood for MIN SI Case
659.4	Peak Pressure and Temperature Occur
1110.	RHR/SI/CV alignment for recirculation (Cold Leg Recirculation Begins)
3778.	Containment Spray is Aligned for Recirculation
28,800	Containment Spray is Terminated
$2.592X10^6$	Transient Modeling Terminated

Table III.15-14 Double-Ended Pump Suction Break - Mass Balance
Unit 2 with Westinghouse D5 Steam Generators
Minimum Safeguards at MUR Conditions

		Mass Balance						
Time (seconds)		0.00	25.60	25.60	194.46	652.36	1368.27	3600.00
		Mass (thousand lbm)						
Initial	In RCS and Accumulators	763.14	763.14	763.14	763.14	763.14	763.14	763.14
Added Mass	Pumped Injection	0.00	0.00	0.00	97.62	412.33	761.53	1060.94
	Total Added	0.00	0.00	0.00	97.62	412.33	761.53	1060.94
***** Total Available *****		763.14	763.14	763.14	860.76	1175.47	1524.67	1824.08
Distribution	Reactor Coolant	512.72	56.21	77.16	139.87	139.87	139.87	139.87
	Accumulators	250.42	192.01	171.05	0.00	0.00	0.00	0.00
	Total Contents	763.14	248.21	248.21	139.87	139.87	139.87	139.87
Effluent	Break Flow	0.00	514.91	514.91	709.27	1023.98	1399.70	1699.12
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	514.91	514.91	709.27	1023.98	1399.70	1699.12
***** Total Accountable *****		763.14	763.12	763.12	849.14	1163.85	1539.57	1838.99

Table III.15-15 Double-Ended Pump Suction Break - Energy Balance
Unit 2 with Westinghouse D5 Steam Generators
Minimum Safeguards at MUR Conditions

		Energy Balance						
Time (seconds)		0.00	25.60	25.60	194.46	652.36	1368.27	3600.00
		Energy (million BTU)						
Initial Energy	In RCS, Accumulators, and SG	832.73	832.73	832.73	832.73	832.73	832.73	832.73
Added Energy	Pumped Injection	0.00	0.00	0.00	7.13	30.11	56.55	86.33
	Decay Heat	0.00	8.56	8.56	29.61	72.96	128.34	263.64
	Heat From Secondary	0.00	18.00	18.00	18.00	18.00	18.00	18.00
	Total Added	0.00	26.56	26.56	54.75	121.07	202.89	367.97
***** Total Available *****		832.73	859.29	859.29	887.47	953.80	1035.62	1200.70
Distribution	Reactor Coolant	309.13	12.11	13.99	37.24	37.24	37.24	37.24
	Accumulators	22.41	17.18	15.30	0.00	0.00	0.00	0.00
	Core Stored	22.71	12.14	12.14	4.91	4.59	4.29	3.33
	Primary Metal	156.71	149.10	149.10	121.42	84.63	63.68	48.48
	Secondary Metal	74.44	74.58	74.58	67.08	51.47	35.02	26.36
	Steam Generators	247.33	269.09	269.09	237.09	174.85	114.30	85.00
	Total Contents	832.73	534.21	534.21	467.74	352.77	254.53	200.41
Effluent	Break Flow	0.00	324.49	324.49	409.80	591.09	784.68	1006.78
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	324.49	324.49	409.80	591.09	784.68	1006.78
***** Total Accountable *****		832.73	858.70	858.70	877.54	943.87	1039.21	1207.19

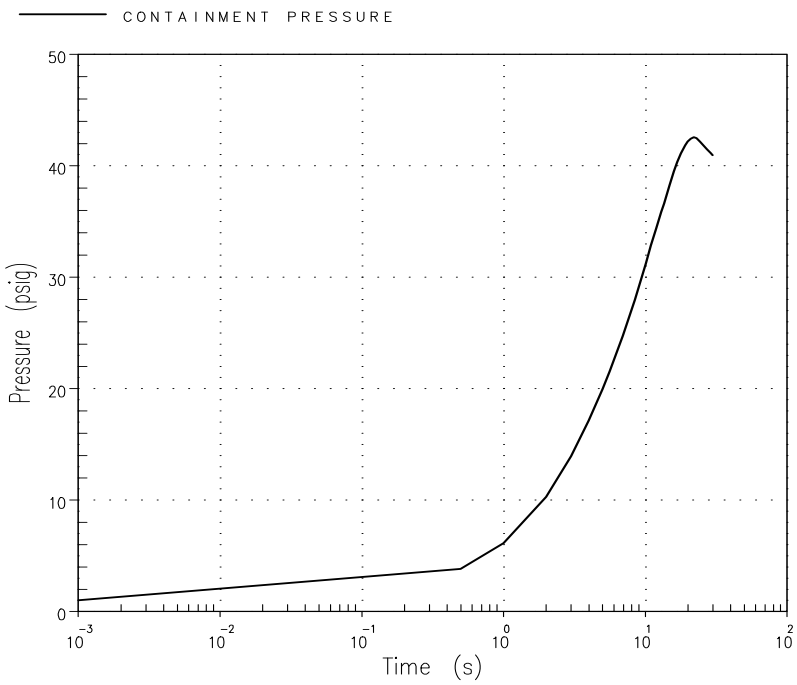

Figure III.15-1 Unit 1 MUR Analysis Double Ended Hot Leg Break with Minimum ECCS Flows - Containment Pressure Transient

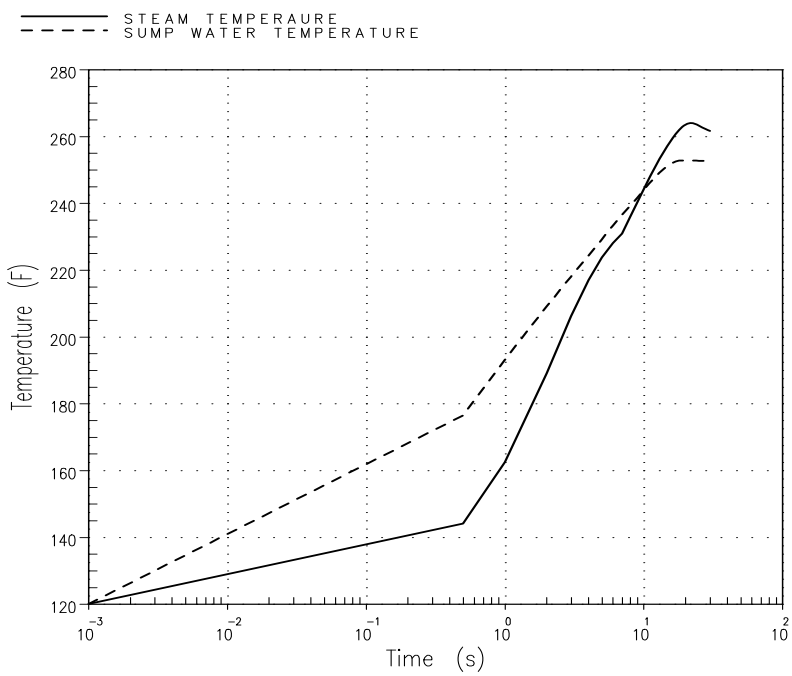

Figure III.15-2 Unit 1 MUR Analysis Double Ended Hot Leg Break with Minimum ECCS Flows - Containment Temperature Transient

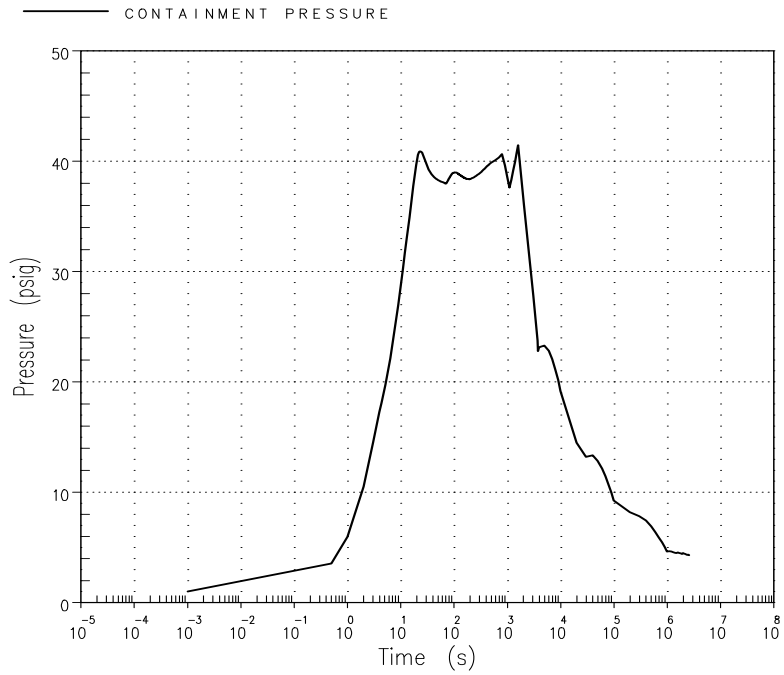

Figure III.15-3 Unit 1 MUR Analysis Double Ended Pump Suction Break with Minimum ECCS Flows Containment Pressure Transient

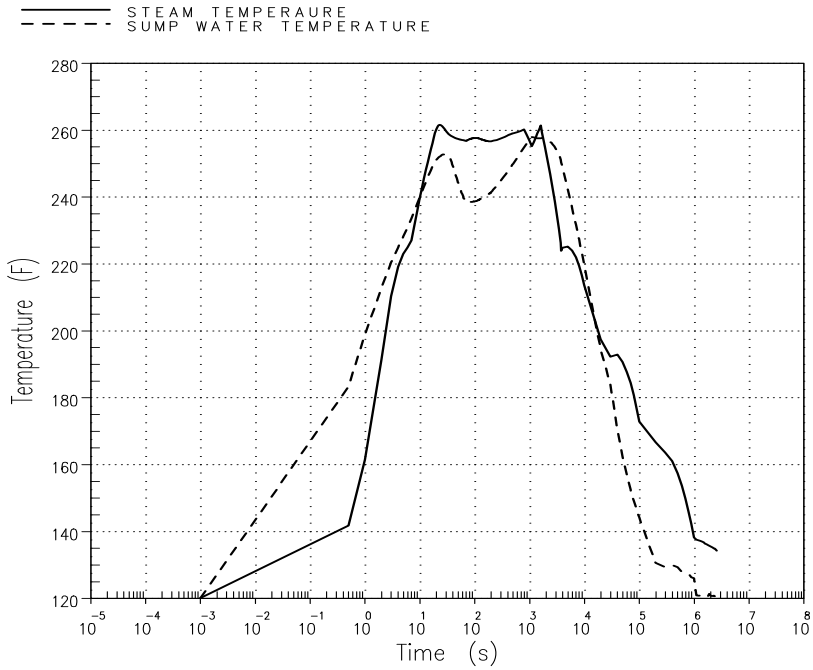

Figure III.15-4 Unit 1 MUR Analysis Double Ended Pump Suction Break with Minimum ECCS Flows Containment Temperature Transient

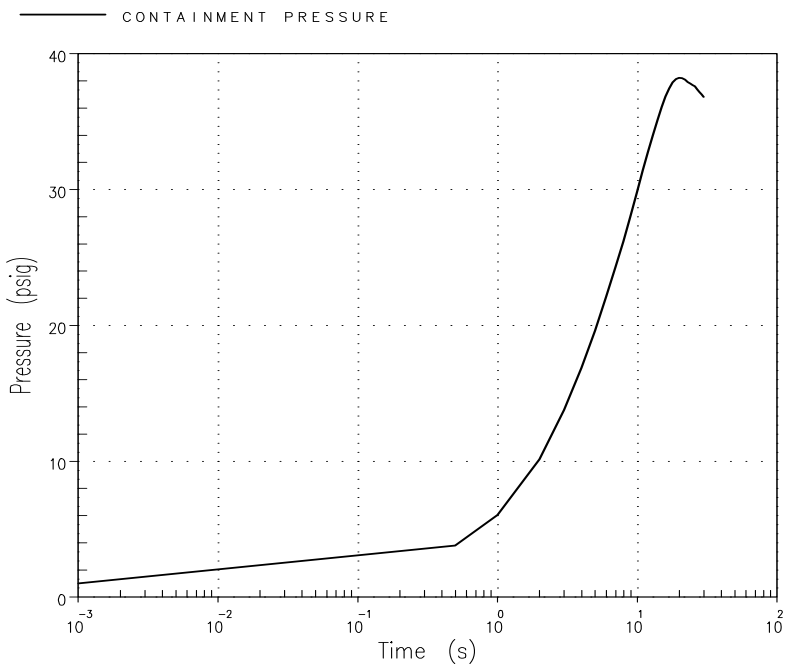

Figure III.15-5 Unit 2 MUR Analysis Double Ended Hot Leg with Minimum ECCS Flows Containment Pressure Transient

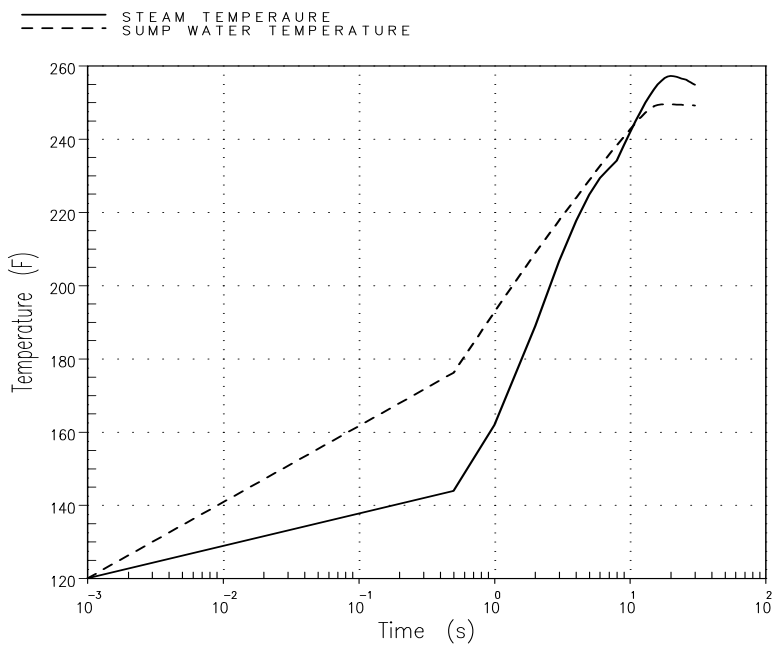

Figure III.15-6 Unit 2 MUR Analysis Double Ended Hot Leg Break with Minimum ECCS Flows Containment Temperature Transient

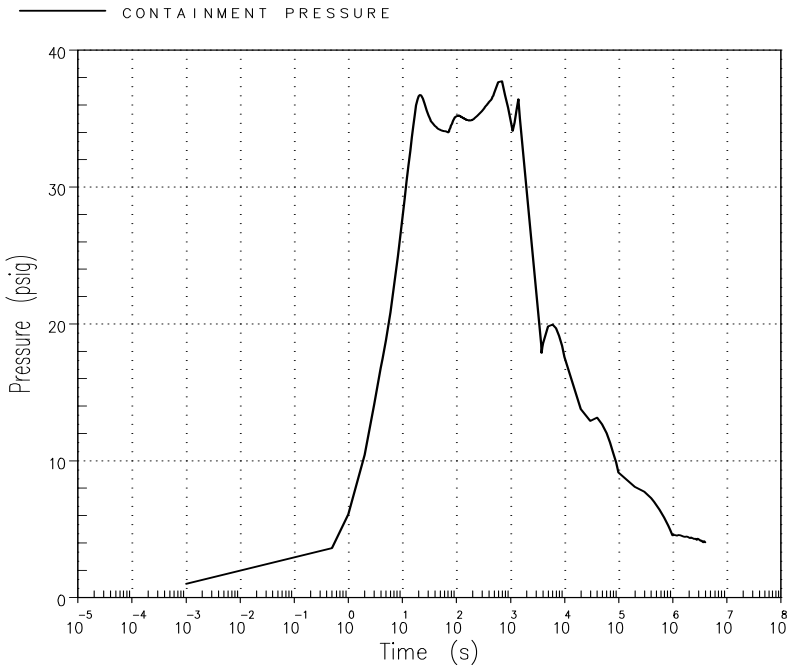

Figure III.15-7 Unit 2 MUR Analysis Double Ended Pump Suction Break with Minimum ECCS Flows Containment Pressure Transient

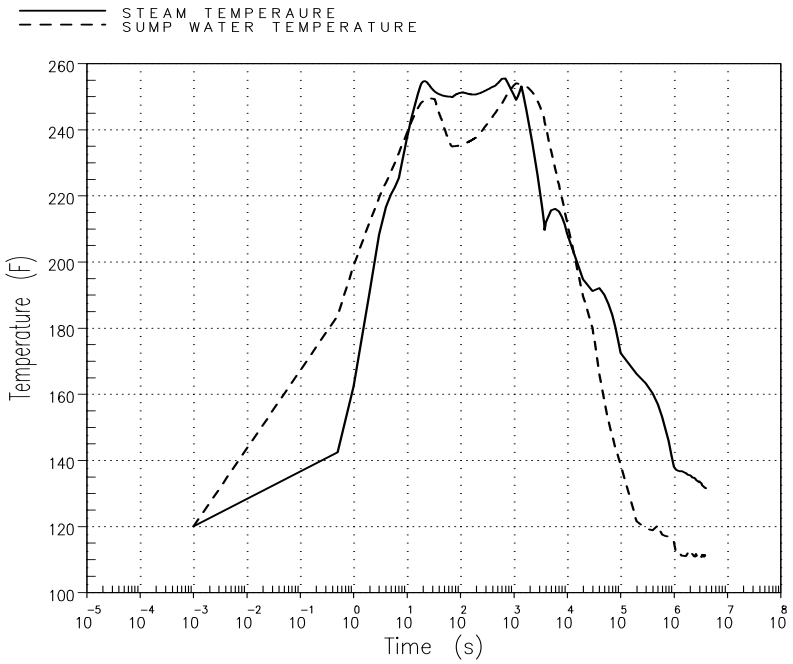

Figure III.15-8 Unit 2 MUR Analysis Double Ended Pump Suction Break with Minimum ECCS Flows Containment Temperature Transient

III.16 Main Steam Line Break Mass and Energy Releases Inside Containment – UFSAR 6.2.1.4

III.16.1 Identification of Causes and Accident Description

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the plant operating conditions and the size of the rupture as well as the configuration of the plant steam system and containment design.

The main steam line break (MSLB) mass and energy (M&E) releases used for containment temperature and pressure response is described in Section 6.2.1.4 of the UFSAR (Reference III.16-1).

III.16.2 Method of Analysis

Westinghouse mass and energy releases and containment response analyses used NRC approved methods, assuming 3672 MWt NSSS (i.e., 102.0% of 3600.6 MWt NSSS). A reduction in the calorimetric uncertainty allows for a power uprate, as long as the total NSSS power, including uncertainty, does not exceed this value. There are also small changes to other operating parameters, which were evaluated using representative cases. The peak containment temperature and pressure cases were revaluated as well as two additional hot zero power (HZP) limiting cases for each unit.

The break flows and enthalpies of the steam release through the steamline break inside containment are analyzed with the LOFTRAN computer code (Reference III.16-2). Blowdown mass and energy releases were also determined using LOFTRAN, including effects of core power generation. Calculation of containment pressure and temperature is accomplished by use of the computer code COCO (Reference III.16-3). The MSLB inside containment analysis using these methodologies have been previously approved for use in Reference III.16-4.

III.16.3 Analysis Inputs and Assumptions

The MSLB M&E releases inside containment analyses of record (AOR) assumed a calorimetric uncertainty of 2.0% that offsets the increased MUR power for this evaluation. This evaluation was performed at MUR conditions with an increase in feedwater temperature, an increase in secondary side pressure, and a longer containment spray actuation delay. Furthermore, the reanalysis of the steamline break hot zero power limiting containment integrity cases were performed with a more conservative moderator density coefficient. Parameters chosen for key hot full power (HFP) and hot zero power (HZP) cases reanalyzed are summarized in Table III.16-1.

The major assumptions associated with this evaluation are:

- Nuclear steam supply system (NSSS) power of 3672 MWt,
- Thermal design flow of 92,000 gpm/loop,
- Vessel average temperature of 588.0°F (575.0°F for T_{avg} coastdown cases),

- Maximum steam pressure of 1040 psia for Units 1 and 982 psia for Units 2, and
- Steam generator tube plugging of 0%.

Unit 1 at each site has Babcock and Wilcox (B&W) replacement SGs, whereas Unit 2 at each site has Westinghouse model D5 SGs. Separate analytical models were generated for each steam generator type and were used for the calculations.

III.16.4 Analysis Acceptance Criteria

For the MSLB inside containment analysis, the calculated containment pressure must remain less than the design pressure of 50 psig and the temperature and pressure profiles used for equipment qualification (EQ) must also be met.

III.16.5 Analysis Results

Table III.16-1 summarizes the results of the key cases selected for the reanalysis to the AOR.

For the reanalyzed cases, the resultant maximum containment pressures of 34.6 psig and 31.4 psig respectively for Byron and Braidwood Stations Units 1 and Units 2, are less than the peak containment pressures of 39.3 psig for Unit 1 and 38.3 psig for Unit 2 for the current analysis of record (AOR). These MSLB containment pressures are below the containment design pressure of 50 psig and the Technical Specification controlled Integrated Leak Rate Test (ILRT) pressures (P_a) for Units 1 and Units 2 of 42.8 and 38.4 psig respectively. The AOR pressure composite curves remain bounding as shown in Figures III.16-1 and III.16-3, for Units 1 and 2, respectively.

The maximum containment air temperature for the peak case increased by 0.6°F to 333.6°F for Units 1; the previous maximum containment temperature of 330.8°F remains applicable for Units 2. The margin between MSLB and LOCA peak containment structure temperatures is ~60°F, with the LOCA being the bounding condition for the structure temperature. Therefore, LOCA peak containment structure temperature will remain bounding following the small increase in peak containment temperature for MSLB.

The AOR temperature composite curves for Unit 2 remain bounding as shown in Figures III.16-4. The Unit 1 containment temperature response for the two 1.0 ft^2 double ended rupture (DER) cases (HFP and the HZP) result in a small increases above the AOR composite curves for the time periods as indicated in Table III.16-1 and as shown on Figure III.16-2. The margin between MSLB and LOCA peak containment structure temperatures is ~60°F, with the LOCA being the bounding condition. Therefore, LOCA peak containment structure temperature will remain bounding following the small increase in peak containment temperature for MSLB. These small increases do not impact the UFSAR conclusions for the long-term steam line break event inside containment.

New EQ pressure and temperature profiles were developed. The effect of the revised containment conditions on the environmental qualification of electrical equipment in containment is discussed in Section V.1.C.iii.

III.16.6 Refe rences

III.16-1 Byron/Braidwood Nuclear Stations UFSAR, Revision 12, Chapter 6.0, “Engineered Safety Features.”

III.16-2 WCAP-7907-P-A, “LOFTRAN Code Description,” April 1984

III.16-3 "Containment Pressure Analysis Code (COCO)," WCAP-8327, July, 1974 (Proprietary), WCAP-8326, July, 1974 (Non-Proprietary).

III.16-4 Letter from G. F. Dick (USNRC) to O. D. Kingsley (Exelon), “Issuance of Amendments: Increase in Reactor Power, Byron Station Units 1 and 2, Braidwood Station, Units 1 and 2.” (TAC NOS. MA9428, MA9429, MA9426 and MA9427), May 4, 2001 [Accession No. ML011420274]

Table III.16-1 Steamline Break Inside Containment Parameters and Comparison to AOR

Unit	Case	Power	Break	Failure	Offsite Power	Reason for Case Selection	Comparison to AOR
1	IC-1a	HZP	Full DER	RCFC	Yes	Part of Bounding Temperature Composite	No Change to Composite Curves
	IC-1b	HZP	1 ft^2 DER	RCFC	Yes	Part of Bounding Temperature Composite	Temperature > AOR (from 193 – 234 seconds post MSLB) Composite Temperature Curve Revised
	IC-1c	HFP	Full DER	FWIV	Yes	Peak Containment Pressure	Peak Pressure Bounded No Change to Composite Curves
	IC-1d	HFP	1 ft^2 DER	MSIV	No	Peak Containment Temperature	Revised Composite Temperature Curve Peak Temperature > AOR (from 333°F to 333.6°F) Temperature > AOR (Post MSLB from 11 – 27 seconds and 37 – 73 seconds)
2	IC-2a	HZP	Full DER	RCFC	Yes	Part of Bounding Temperature Composite	No Change to Composite Curves
	IC-2b	HZP	Full DER	RCFC	Yes	Peak Containment Pressure	Peak Pressure Bounded No Change to Composite Curves
	IC-2c	HZP	1 ft^2 DER	RCFC	Yes	Part of Bounding Temperature Composite	No Change to Composite Curves
	IC-2d	HFP	1 ft^2 DER	MSIV	No	Peak Containment Temperature	Peak Temperature Bounded No Change to Composite Curves

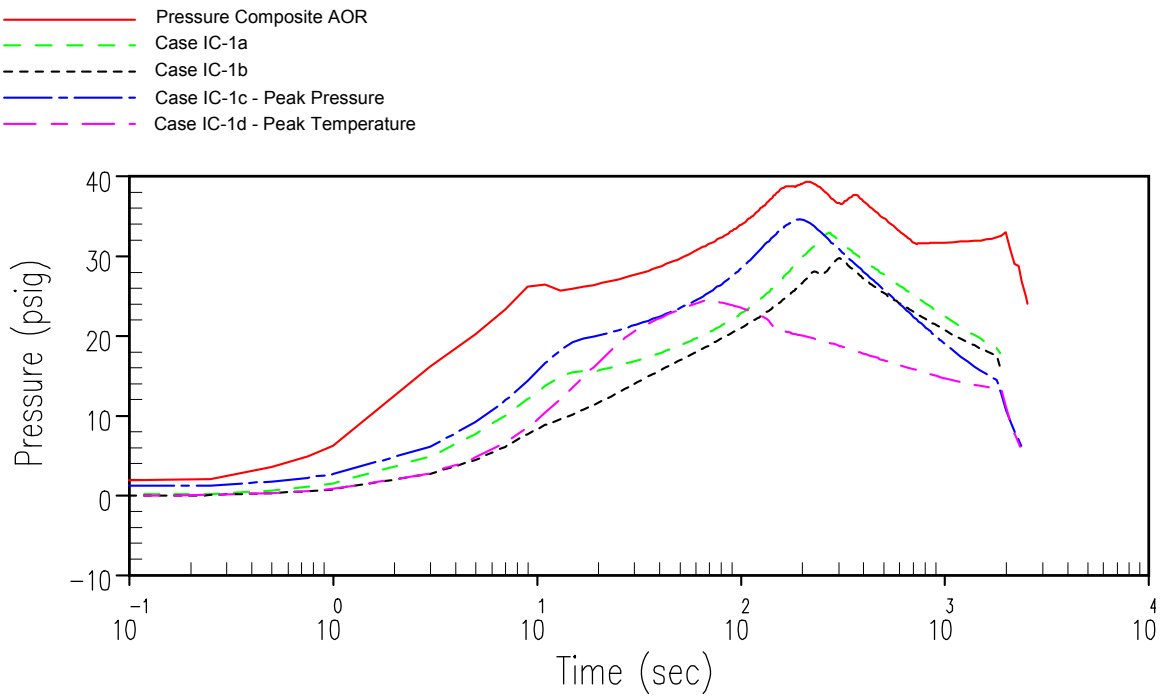

Figure III.16-1 Unit 1 MUR Analysis MSLB - Containment Pressure Transient

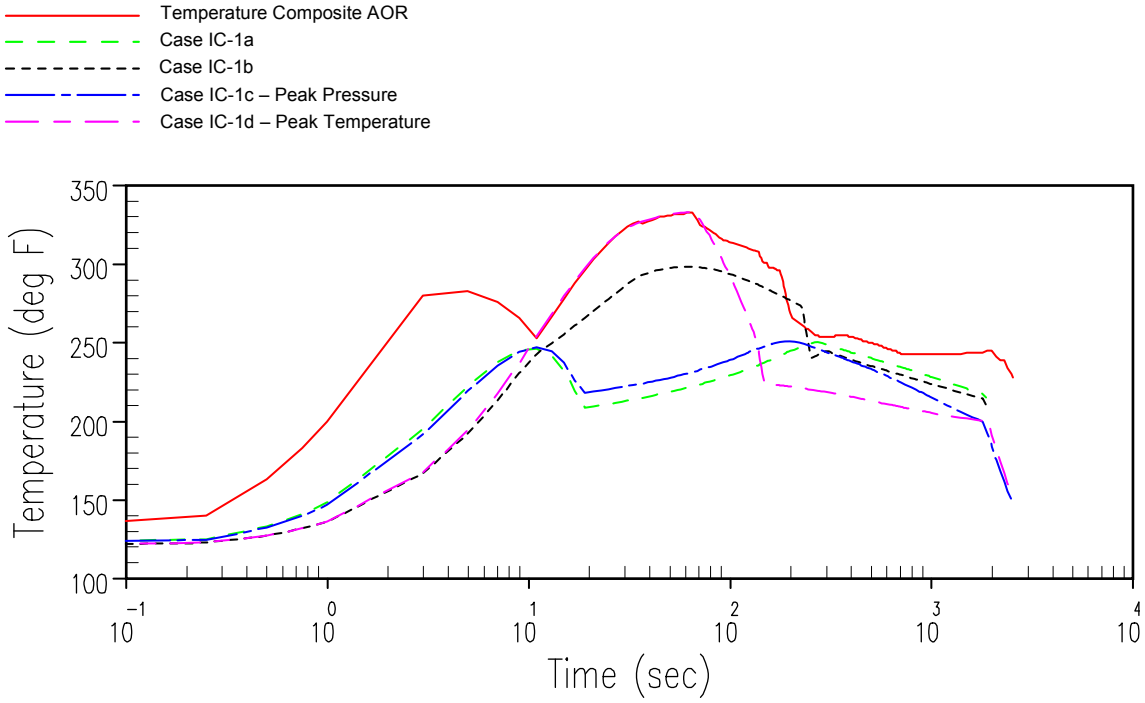

Figure III.16-2 Unit 1 MUR Analysis MSLB - Containment Temperature Transient

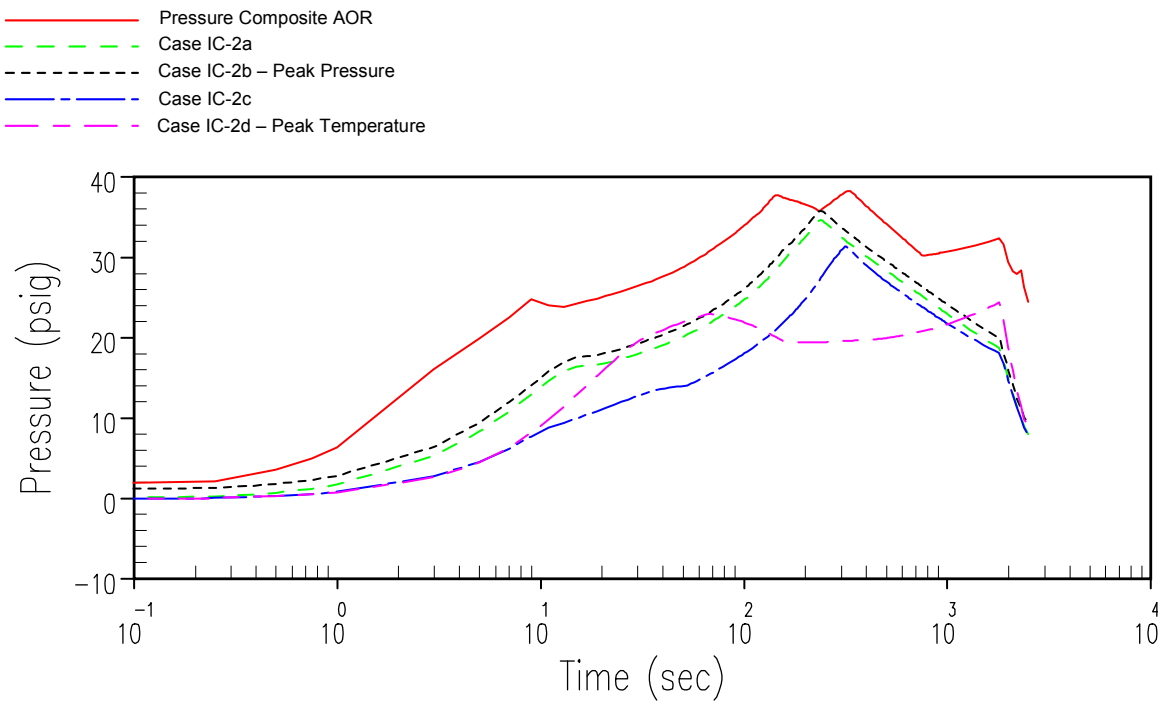

Figure III.16-3 Unit 2 MUR Analysis MSLB - Containment Pressure Transient

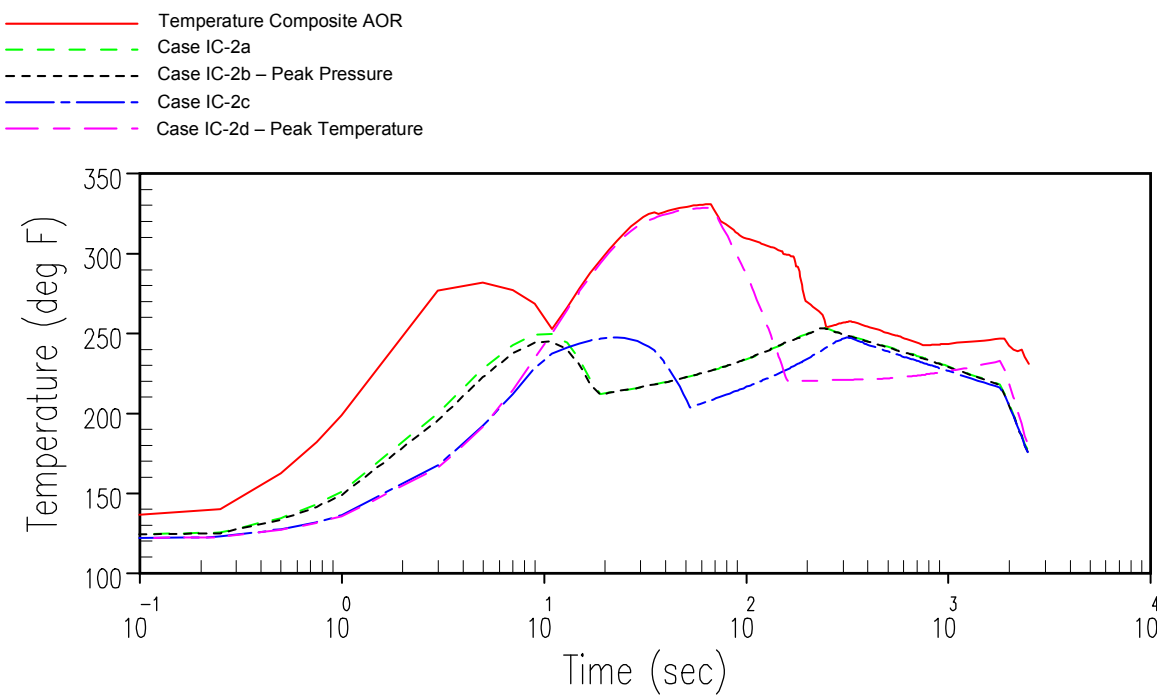

Figure III.16-4 Unit 2 MUR Analysis MSLB - Containment Temperature Transient

III.17 Radiological Consequences

III.17.1 Steam Release for Radiological Dose Analysis

III.17.1.1 Identification of Causes and Accident Description

Moderate activity releases to the environment are expected to occur several times throughout the licensed life of Byron and Braidwood Units 1 and 2. Radioactive releases to the environment are postulated via the following scenarios.

- An activity level exists in the reactor coolant system (RCS): The activity level in the RCS may be low, resulting from activated corrosion products or from the minute release of fission material from defective fuel assemblies. The activity level may also be moderate to high, resulting from fuel cladding failures and the subsequent fission product release. Cladding failures may occur from the locked rotor or steamline break events. Each of these events is classified with regard to its severity and frequency of occurrence.
- A primary-to-secondary leak occurs: The most common primary-to-secondary leak would be a leak through the wall of one or more steam generator tubes. A maximum allowable leak rate for Byron and Braidwood Units 1 and 2 is specified in the Technical Specifications, based on tube integrity requirements. The Technical Specification leakage limit is used to determine radioactivity releases to the environment.
- Secondary-side activity is released into the atmosphere: Given that a primary-to-secondary leak exists and the condenser is not available for steam dump following an accident that produces a reactor trip, steam and radioactivity will be released to the atmosphere through the steam generator relief or safety valves while the plant is being brought to a cold shutdown condition.

Vented steam releases have been calculated for the locked rotor and steamline break events to support the Byron and Braidwood Units 1 and 2 Measurement Uncertainty Recapture (MUR) power uprate program. The steam releases form part of the information documented in Table 17.1-3 (steam line break), and Table 17.3-4 (locked rotor) of Byron and Braidwood Nuclear Stations UFSAR, Chapter 15.0, “Accident Analyses” (Reference III.17-1). These steam releases are used as input to the radiological dose analysis that is required to support the Byron and Braidwood Units 1 and 2 MUR power uprate.

III.17.1.2 Method of Analysis

An energy balance determines the amount of heat that would be dissipated via steam release through the SG relief or safety valves. The energy balance considers heat generated in the core, heat released or absorbed by thick metal in the RCS and intact SGs, and heat released or absorbed within the fluids in the RCS and intact SGs. The energy that cannot be stored within the defined boundary of the RCS and intact SGs is removed via steaming (saturated liquid turning into saturated vapor), and the analysis determines the mass of steam released. The calculation considers two different time periods: from 0 to 2 hours and from 2 hours until RHR cut-in. Quasi-steady-state conditions are assumed at the beginning and end of each time period.

III.17.1.3 Analysis Inputs and Assumptions

The following key input parameters and general assumptions associated with the Byron and Braidwood MUR power uprate have been used in the calculation of the steam releases:

- Nuclear Steam Supply System (NSSS) power of 3672 MWt with 0.0% additional uncertainty,
- RCS pressure of 2250 psia,
- 0% SG tube plugging,
- Nominal RCS T_{avg} is 588.0°F + 9.1°F of additional uncertainty and bias,
- Full power pressurizer level is 60% span,
- Residual heat removal system (RHRS) cut-in temperature is 340°F,
- RHRS cut-in pressure is 300 psia, and
- SG types are BWI on Units 1 and Model D5 on Units 2.

Steam relief through the steam generator atmospheric power operated relief valves (PORVs) will be required until the reactor can be placed on the RHR system. It has been confirmed that 40 hours of steam release will occur prior to placing the plant in the RHR mode of operation. After the first 2 hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 38 hours are required to cool down and depressurize the plant from no-load conditions to the RHR operating conditions.

No explicit assumption is considered in this analysis regarding steam generator blowdown isolation. The implied assumption is that the entire inventory of the steam generators is released to the environment, so there is no loss of inventory through the blowdown line to account for. This provides a conservative calculation of the quantity of steam vented during the noted time periods.

III.17.1.4 Analysis Acceptance Criteria

There are no specific acceptance criteria associated with the calculation of the steam releases used as input to the radiological dose analyses. Tables of steam releases for each of the cooldown intervals of these transients are used as input to the radiological dose analysis in support of the Byron and Braidwood Units 1 and 2 MUR power uprate.

III.17.1.5 Analysis Results

Tables III.17-1 and III.17-2 summarizes the vented steam releases from the intact-loop steam generators for the 0 - 2 hour time period and the 2 - 40 hour time period for the steamline break and locked rotor events, respectively. These two time periods are documented to support the Byron and Braidwood Units 1 and 2 MUR power uprate. It should be noted that Westinghouse revised their methodology in that a separate time step from 2-8 hours is no longer calculated. This change in methodology has no impact on the dose results conclusion.

The steam release values used in the current Main Steam Line Break (MSLB) accident dose analysis do

not bound MUR conditions as shown in Table III.17-1. The MSLB dose calculation is subsequently being revised using the updated steam release values calculated for MUR conditions. The results of the revised MSLB dose calculation are presented below.

The current steam mass release used for the locked rotor accident included the effect of reactor coolant pump heat and a longer time to reactor trip. The MUR steam mass release as shown in Table III.17-2 decreases because the MUR reanalysis assumes an immediate reactor trip concurrent with the locked rotor event followed by a loss of offsite power that results in the loss of all reactor coolant pumps. Both of these changes result in decreased heat input which shortens the cooldown and therefore the time and mass of the steam release.

III.17.2 Dose Evaluation

III.17.2.1 Main Steam Line Break Dose Evaluation – UFSAR 15.1.5/15.1.6

The current MSLB radiological analysis is based upon the alternative source term (AST) as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The analysis involves primary coolant radiological source release to the secondary side from the steam generator (SG) and then to the environment. The source terms for equilibrium conditions with 1% failed fuel are normalized to the Technical Specifications (TS) Dose Equivalent (DE) Iodine-131(I-131) limits in the primary coolant, which removes the power dependence from the analysis.

The steam releases modeled in the MSLB analysis are consistent with a core thermal power of 3658.3 MWt (102% of 3586.6 MWt). The release pathways and dose conversion factors are unchanged from the AST license amendment request and associated safety evaluation reports (SERs). As discussed above, the steam releases have been revised for MUR power uprate. The updated mass of the steam release has been incorporated into the revised dose analysis. The atmospheric dispersion factors (χ/Q) values have been updated and incorporated into the dose analysis as per the current commitment to the NRC (RAI response letter RS-06-019). The TS DE I-131 limits and other key dose parameters are not revised as a result of the measurement uncertainty recapture (MUR) power uprate. A comparison of the MUR dose analysis results to the current analysis of record and the regulatory limits is provided in Table III.17-3.

The MUR power uprate dose analysis results in a maximum total effective dose equivalent control room dose of 2.845 rem, Exclusion Area Boundary (EAB) dose of 0.201 rem, and Low Population Zone (LPZ) dose 0.459 rem, which are less than the regulatory limits. Therefore, the MSLB accident is acceptable for the MUR power uprate.

III.17.2.2 Locked RCP Rotor Dose Evaluation – UFSAR 15.3.3

As discussed in UFSAR Section 15.3.3, the locked rotor accident (LRA) analysis is based upon the AST as defined in NUREG-1465, with acceptance criteria as specified in either 10 CFR 50.67 or Regulatory Guide 1.183. The core inventory source term used in the current locked rotor accident analysis is a function of core power, enrichment, burn-up, gap fractions for non-LOCA events from Regulatory Guide 1.183, an assumed percent of failed fuel, and an assumed radial peaking factor. The existing LRA dose evaluation was performed using the core inventory that assumes 3658.3 MWt, which is 102% of 3586.6 MWt. No changes to the assumed percent of failed fuel or assumed radial peaking factor are required to

support the MUR power uprate. The radionuclide activity in the steam release modeled in the current LRA analysis is consistent with a core thermal power of 3658.3 MWt (102% of 3586.6 MWt) and the mass of steam released bounds the LRA steam release under the MUR power uprate. The release pathways, and dose conversion factors are unchanged from the AST license amendment requests and associated SERs. The current commitment to the NRC (RAI response letter RS-06-019) is to incorporate the updated χ/Q values when the analysis is revised; however, the dose analysis was not revised because this analysis is performed at a power level that bounds the MUR power uprate power level.

Therefore, the current LRA dose analysis remains bounding for the MUR power uprate.

III.17.3 Refer ences

III.17-1 Byron/Braidwood Nuclear Stations UFSAR, Revision 12, Chapter 15.0, “Accident Analyses.”

Table III.17-1 Steamline Break - Steam Release for Dose

Current		MUR Uprate	
Interval (hours)	**Total Steam Mass Release** (lbm)	**Interval** (hours)	**Total Steam Mass Release** (lbm)
0 to 2	442,000	0 to 2	447,000
2 to 8	977,000	2 to 40	3,279,000
8 to 40	2,216,000		

Table III.17-2 Locked Rotor - Steam Release for Dose

Current		MUR Uprate	
Interval (hours)	**Total Steam Mass Release** (lbm)	**Interval** (hours)	**Total Steam Mass Release** (lbm)
0 to 2	719,000	0 to 2	457,000
2 to 8	1,109,000	2 to 40	3,323,000
8 to 40	2,664,000		

Table III.17-3 Main Steam Line Break Accident – Dose Analysis

Event Condition	Location	TEDE (rem)		
		AOR	**MUR**	**Limit**
Pre-Accident Spike (1)	**Control Room**	0.581	0.580	5.0
	EAB	0.127	0.145	25
	LPZ	0.073	0.083	25
Post Accident Spike (2)	**Control Room**	2.844	2.845	5.0
	EAB	0.175	0.201	2.5
	LPZ	0.406	0.459	2.5

Notes:

(1) Case 1: Pre-accident 60 μCi/g DE I-131 spike

(2) Case 2: Accident initiated 500 times equilibrium iodine release rate spike

IV MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

1. *A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.*

   A. *This discussion should address the following components:*

      i. *reactor vessel, nozzles and supports*

      ii. *reactor core support structures and vessel internals*

      iii. *control rod drive mechanisms*

      iv. *nuclear steam supply system (NSSS) piping, pipe supports, branch nozzles*

      v. *balance-of-plant (BOP) piping (NSSS interface systems, safety-related cooling water systems, containment systems)*

      vi. *steam generator tubes, secondary side internal support structures, shell, nozzles*

      vii. *reactor coolant pumps*

      viii. *pressurizer shell, nozzles, surge line*

      ix. *safety-r elated valves*

   B. *The discussion should identify and evaluate any changes related to the power uprate in the following areas:*

      i. *str esses*
      ii. *cumulative usage factors (fatigue)*
      iii. *flow-induced vibration*
      iv. *changes in temperature (pre- and post-uprate)*
      v. *changes in pressure (pre- and post-uprate)*
      vi. *changes in flow rates (pre- and post-uprate)*
      vii. *high-energy line break locations*
      viii. *jet impingement and thrust forces*

C. The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel with respect to:

i. pressurized thermal shock calculations
ii. fluence evaluation
iii. heatup and cooldown pressure-temperature limit curves
iv. low-temperature overpressure protection
v. upper -shelf energy
vi. surveillance capsule withdrawal schedule

D. The discussion should identify the code of record being used in the associated analyses, and any changes to the code of record.

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs, and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.

F. The discussion should address whether the effect of the power uprate on steam generator tube cycle fatigue is consistent with NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes, February 5, 1988.

RESPONSE TO IV MECHANICAL/STRUCTURAL/MATERIAL COMPONENT INTEGRITY AND DESIGN

IV.1.A.i Reactor Vessel

<u>RV Structural</u>

The inputs to the reactor vessel component stress and cumulative fatigue usage factors were evaluated at the uprated operating conditions. The key inputs for the MUR conditions were the NSSS design parameters, NSSS design transients and the interface loads associated with the various reactor vessel components. The Byron and Braidwood reactor vessels were previously analyzed with a minimum normal operating inlet temperature of 538.2°F and a maximum normal operating outlet temperature of 620.3°F. Due to operational restrictions, the MUR minimum vessel inlet temperature is 538.2°F and maximum vessel outlet temperature is 618.4°F. The MUR temperature values (538.2°F - 618.4°F) are bounded by the values in the previous analysis (538.2°F - 620.3°F).

The NSSS design transients associated with the reactor vessel components remain unchanged for the MUR. Service condition interface loads did not change for the MUR; however, the lifting lug loads are updated per the interface load review. Therefore, existing reactor vessel component stress and maximum cumulative fatigue usage factors did not change for the MUR except for the lift lug.

The lift lug loads were evaluated. All lift lug stress limits are met for MUR conditions.

The code of record is listed in Section IV.1.D and remains unchanged. The reactor vessel meets the stress and fatigue analysis requirements of ASME B&PV Code, Section III, for plant operation at the uprated power conditions.

Integrated Head Package CRDM Seismic Support Assembly

The MUR project has no effect on the structural qualification of the Integrated Head Package CRDM Seismic Support Assembly since the revised loads are bounded by the existing design basis loads. The structural qualification of the Integrated Head Package CRDM Seismic Support Assembly, as documented in the component stress report (Reference IV1.A.i -1), is still bounding.

The code of record for the Integrated Head Package CRDM Seismic Support Assembly (as reported in the abstract section of Reference IV.1.A.i-2) is listed in Table IV.1.D-1 and remains unchanged.

References

IV.1.A.i-1 WCAP-9610, revision 1, Stress Report, 4-Loop Integrated Head Package, CRDM Seismic Support Assembly, for Commonwealth Edison Company, Byron Units 1 and 2, Braidwood Units 1 and 2.

IV.1.A.i-2 955138, revision 2, Westinghouse Equipment Specification for Commonwealth Edison, Byron Units 1 and 2 and Braidwood Units 1 and 2 Nuclear Plants, Integrated Head Package, Control Rod Drive Mechanism Seismic Support Assembly.

IV.1.A.ii Reactor Vessel Internals

The revised design conditions due to MUR were evaluated for impact on the current analyses of record for reactor vessel internals and the results of these assessments are as follows.

IV.1.A.ii.a Core Bypass Flow

The design core bypass flow limit for the reactor pressure vessel system is 8.3% of the total reactor vessel flow with the elimination thimble plugging devices. This core bypass flow limit remains unchanged and valid for MUR power uprate conditions. The MUR power uprate RCS conditions have an insignificant effect on the core bypass flow and the calculated core bypass flow remains below the 8.3% design limit.

IV.1.A.ii.b Rod Control Cluster Assembly Drop Time

RCCA drop time is affected by changes to the RCCA driveline, fuel assembly thermal hydraulic characteristics, and/or plant operating conditions. The MUR power uprate does not change the RCCA driveline or fuel assembly thermal-hydraulic characteristics. The only change is in the plant operating conditions. The increased power level results in a decrease in core inlet temperature of about 0.6°F. This decrease in temperature results in a small increase in the calculated RCCA drop time. However, the Technical Specification limit of 2.7 seconds remains bounding and applicable for the MUR power uprate conditions.

IV.1.A.ii.c Hydraulic Lift Forces and Pressure Losses

A comparison of the parameters used to determine hydraulic lift forces and pressure losses on various reactor internal components under MUR conditions to current conditions was performed. Since the input used remains unchanged from the current analysis of record, the existing hydraulic lift forces and pressure losses remain bounding for the MUR power uprate conditions.

IV.1.A.ii.d Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or fuel rod vibration caused by a high-velocity water jet. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the core. The baffle jetting phenomenon could lead to fuel cladding damage.

A comparison of the parameters used to determine baffle joint momentum flux and fuel rod stability under MUR conditions to current conditions was performed. Since the input used remains unchanged from the current analysis of record, the existing baffle joint momentum flux and fuel rod stability remain bounding for the MUR power uprate conditions.

IV.1.A.ii.e Mechanical Evaluation

The MUR power uprate conditions do not affect the current design bases for seismic and loss of-coolant-accident (LOCA) loads. The FIV stress levels on the core barrel assembly and upper internals are below the material high-cycle fatigue endurance limit. Therefore, the MUR uprated conditions do not affect the structural margin for FIV.

IV.1.A.ii.f Structural Evaluation

Evaluations were performed to demonstrate that the structural integrity of reactor internal components is not adversely affected by the MUR power uprate. For reactor internal components, the stresses and cumulative fatigue usage factor of the previous analyses remain bounding at MUR power uprate.

IV.1.A.ii.f.1 Lower Core Plate Structural Analysis

The lower core plate (LCP) is subjected to the effects of heat generation rates (HGRs), due to its proximity to the core. Structural evaluations were performed to demonstrate that the LCP structural integrity was not adversely affected by the revised design conditions. The LCP maximum primary plus secondary stress intensity and cumulative fatigue usage factor, including the effect of increased HGRs, are acceptable. The LCP is structurally adequate for the MUR power uprate conditions.

IV.1.A.ii.f.2 Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel with installed baffle plates. Bolting connects former plates to the baffle and core barrel. This bolting restrains baffle plate motion. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads, and secondary loads consisting of preload and thermal loads resulting from RCS temperatures and gamma heating rates.

Since the input used in this analysis remain unchanged from the current analysis of record, the existing baffle-barrel region thermal and structural analysis results remain bounding for the MUR power uprate conditions.

IV.1.A.ii.f.3 Upper Core Plate Structural Evaluation

The upper core plate (UCP) is subjected to the effects of heat generation rates (HGRs), due to its proximity to the core. Structural evaluations were performed to demonstrate that the UCP structural integrity was not adversely affected by the revised design conditions. The UCP maximum primary plus secondary stress intensity and cumulative fatigue usage factor, including the effect of increased HGRs, are acceptable. The UCP is structurally adequate for the MUR power uprate conditions.

IV.1.A.ii.g Conclusions

The reactor vessel internals evaluations conclude that the reactor internal components continue to meet their design criteria at the MUR power uprate conditions.

IV.1.A.iii Control Rod Drive Mechanism

The control rod drive mechanisms (CRDMs) use electro-magnetic coils to position the rod cluster control assembly (RCCA) within the reactor core. The updated design conditions (design parameters and nuclear steam supply system design transients) were reviewed for their impact on the existing CRDM design basis analyses.

CRDMs are subjected to T_{cold} temperatures and reactor coolant system (RCS) pressures. These are the only design parameters considered in the CRDM evaluation. The maximum T_{cold} from the MUR power uprate design parameters for any case is 555.1°F. The maximum T_{cold} from the analysis of record is 558.4°F. As a result, the analysis of record remains bounding and applicable. No changes in RCS design or operating pressure were made as part of the MUR power uprate. The temperature and pressure transients are unaffected by the MUR power uprate. Since the transients are unchanged, they do not alter the stress results or the bending moment allowables. Therefore, the original transient analysis remains bounding and applicable to the MUR power uprate conditions. The stress intensity limits are based on a design temperature of 650°F and a pressure of 2,500 psia, which are unchanged by the MUR power uprate. Updated seismic and loss of coolant accident loads remain less than the allowable loads provided in the analysis of record. The code of record is listed in Section IV.1.D and remains unchanged.

IV.1.A.iv Reactor Coolant Piping and Supports

The revised design conditions were evaluated for impact on the existing design basis analyses for the reactor coolant loop piping, primary equipment nozzles (reactor pressure vessel inlet and outlet, SG inlet and outlet, and RCP suction and discharge), primary equipment supports (reactor pressure vessel nozzle supports, SG upper lateral, and lower lateral supports, SG columns, SG snubbers and SG lateral bumpers, pressurizer supports, and RCP lateral supports and columns and tie rods), reactor coolant loop branch nozzles, and Class 1 auxiliary piping systems attached to the reactor coolant loop. There are no significant changes to the reactor coolant loop thermal analysis, deadweight and seismic analysis, reactor coolant loop piping fatigue evaluations, and main steam line break analysis. The existing design

transients remain valid for the uprated conditions. The T_{hot} and T_{cold} variations are conservative and bounding for the MUR power uprate temperature ranges. There were no changes to existing pressurizer design transient parameter responses. There are no significant changes to the pressurizer surge line operating conditions.

In conclusion, there are no significant changes to the reactor coolant loop LOCA or main steam line break analyses with respect to the design basis. The current design basis reactor coolant loop piping system deadweight, thermal, and seismic analyses remain applicable for the MUR power uprate conditions. There are no changes to the following: reactor coolant loop displacements at the Class 1 auxiliary line connections to the reactor coolant loop, Class 1 auxiliary lines, primary equipment nozzle qualification, branch nozzle qualification, and primary equipment supports loads. The maximum primary and secondary stresses and maximum usage factors for the deadweight, thermal, and seismic analyses remain valid. The code of record is listed in Table IV.1.D-1.

IV.1.A.v Balance-of-Plant Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems and Containment Systems)

BOP piping includes the following systems:

- Main Steam System
- Extraction Steam System
- Condensate System
- Condensate Booster System
- Heater Drains System
- Feedwater System
- Steam Generator Blowdown System

NSSS interface systems are further discussed in Section VI.1.A. Safety-Related cooling water systems and related issues concerning Generic Letter 96-06 are discussed in Sections VI.1.C and VII.6.E.iii, respectively. Containment systems are discussed in section VI.1.B.

The MUR uprate operating conditions for the BOP piping systems listed above were reviewed for impact based on system operating parameters, the existing piping design/analytical ratings, flow velocities for Flow Accelerated Corrosion (FAC) and thermal expansion effects.

A review and subsequent evaluation of BOP piping comparing MUR operating temperatures with existing operating and design temperatures concluded these lines were acceptable under MUR power uprate operating conditions.

All lines were reviewed in the BOP piping systems and either conform to their current piping design and/or operating pressures or were subsequently evaluated and found to be acceptable for MUR power uprate operating conditions..

Based on the evaluation of flow velocity, lines in the Main Steam, Extraction Steam, Condensate, Condensate Booster, Feedwater and Steam Generator Blowdown Systems operate with flow velocities in excess of guideline flow velocities at the MUR uprate power level. These lines will be addressed via the FAC program as appropriate.

IV.1.A.vi Steam Generators

The following evaluation addresses the Unit 1 and Unit 2 Steam Generators (SGs) at Byron and Braidwood Stations. Note that the Unit 1 SGs and Unit 2 SGs at Byron and Braidwood Stations are different models. The Unit 1 SGs at Byron Station and Braidwood Station are the same model; i.e., BWI Replacement Steam Generators (RSGs). The Unit 2 SGs at Byron Station and Braidwood Station are the same model; i.e., Westinghouse model D-5.

IV.1.A.vi.1 Unit 1 Steam Generators

IV.1.A.vi.1.a Steam Generator Thermal-Hydraulic Evaluation

Thermal-hydraulic analyses have been completed for Byron and Braidwood Stations Unit 1 Replacement Steam Generators (RSGs) at the MUR conditions with a nuclear steam supply system (NSSS) power level of 3672 MWt. These analyses were documented in the "Exelon Byron and Braidwood Stations Unit 1, Replacement Steam Generators MUR Power Uprate Thermal-Hydraulic Performance Report," [Reference IV.1.A.vi.1.a-2]. The analyses determined the steam generator thermal-hydraulic characteristics and inventories, and provided input used to evaluate the potential for tube wear and flow-induced vibration (FIV). The results show that the steam generators have satisfactory thermal-hydraulic performance for the MUR conditions provided in the Certified Design Specification [Reference IV.1.A.vi.1.a-1] and Customer supplied Design Information [Reference IV.1.A.vi.1.a-3].

The thermal-hydraulic performance evaluation consisted of a steady-state, one-dimensional thermal-hydraulic simulation using Babcox and Wilcox (B&W) CIRC code and three-dimensional thermal-hydraulic simulation using B&W ATHOSBWI code.

The results of the thermal-hydraulic analyses show that there is only a minor change in thermal hydraulic conditions due to MUR power uprate and all thermal-hydraulic acceptance criteria for the Byron and Braidwood Unit 1 RSGs continue to be met.

Moisture carry over (MCO) was reviewed for the MUR conditions using the current plant configuration and historical test data. The review concluded that all steam generators' MCO were below the design limits. Since plant configuration changes such as steam generator tube plugging and system modifications also impact MCO, any future changes to those parameters would be evaluated at the time of the change.

References

IV.1.A.vi.1.a-1 Areva NP Inc. Specification 18-1229648-008, "Certified Design Specification for Replacement Steam Generator Byron and Braidwood Stations Unit 1," September 2010.

IV.1.A.vi.1.a-2 B&W Canada Report 236R-PR-01, Rev. 00, "Exelon Byron and Braidwood Stations Unit 1, Replacement Steam Generators MUR Power Uprate Thermal-Hydraulic Performance Report," October 2010.

IV.1.A.vi.1.a-3 Exelon Transmittal of Design Information No. BRW-BYR-MUR-043 dated April 16, 2010 from Dan Milroy to Roy McGillivray/Steve Fluit of B&W Canada, "Byron/Braidwood TODI PU-2010-040 - Responses to DIR BYR/BRW-RFI-001 Request for Thermal Hydraulic Inputs" (includes Measured Calorimetric Data for Byron and Braidwood Unit 1 Steam Generators in TODI PU-2010-01 and TODI PU-2010-02 respectively).

IV.1.A.vi.1.b Steam Generator Structural Integrity

Structural Integrity analyses have been completed for Byron and Braidwood Stations Unit 1 Replacement Steam Generators (RSGs) at the MUR conditions with a nuclear steam supply system (NSSS) power level of 3672 MWt as specified in the Certified Design Specification (Reference IV.1.A.vi.1.b-1]). These analyses were documented in the "Exelon, Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Structural Analysis Report," (Reference IV.1.A.vi.1.b-2). The analysis addresses ASME Section III A, B, C and D service levels and considers the thermal transients, external RSG interface loads and internal pressure boundary attachment loads. The scope of the reconciliation was the entire steam generator pressure boundary including the steam drums, internal and external pressure boundary attachments, lower base support, and all internal components.

The review of the MUR conditions revealed that the maximum pressure and temperature loadings are bounded by the original evaluations for 100% power (NSSS power level of 3425 MWt). Similarly, external loadings were not affected by MUR. The reconciliation analysis confirms that all internal components remain acceptable for the MUR Condition. Therefore, the Design Condition analyses remain valid.

The results of the evaluation demonstrated that the steam generator pressure boundary continue to comply with the structural criteria of the ASME Code, Section III, Class 1, Subsection NB and NF for operation at the MUR conditions. The stresses and fatigue usage factors for internal components are also shown to meet ASME Code limits.

References

IV.1.A.vi.1.b-1 Areva NP Inc. Specification 18-1229648-008, "Certified Design Specification for Replacement Steam Generator Byron and Braidwood Stations Unit 1," September 2010.

IV.1.A.vi.1.b-2 B&W Canada Report 236R-SR-01, Rev. 00, "Exelon, Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Structural Analysis Report," December 2010.

IV.1.A.vi.1.c Steam Generator Flow Induced Vibration and Wear, and Chemistry

Flow Induced Vibration and Wear

Flow induced vibration (FIV) and tube wear analyses have been completed for Byron and Braidwood Stations Unit 1 Replacement Steam Generators (RSGs) at the MUR conditions with a nuclear steam supply system (NSSS) power level of 3672 MWt. These analyses were documented in the "Exelon, Byron and Braidwood Stations Unit 1 Replacement Steam Generator Flow-Induced Vibration and Wear Analysis Report," (Reference IV.1.A.vi.1.c-1).

A flow-induced vibration (FIV) and wear analysis was performed with a bounding analysis at the lower RCS T_{hot} limit with end-of-life plugging and fouling. The FIV analysis was performed for the critical tubes which exhibited the highest FIV responses in the previous analyses of record. The critical tubes selected are analyzed for the following three potential FIV mechanisms: fluid elastic instability, vortex shedding resonance and random turbulence excitation. The critical tube selection also includes the peripheral U-bend tube in the first tube row adjacent to the tube-free-lane which was found to have ineffective hot-leg collector bar support.

All evaluated RSG tube cases meet the FIV limits for fluidelastic instability, vortex shedding resonance and random turbulence excitation. Wear calculations also provided results which satisfy the 40% allowable tube wall wear limit. Wear associated with tube touching in the U-bend will remain within acceptable limits for MUR operating conditions. It is therefore concluded that the Byron and Braidwood Unit 1 RSG tube bundles are adequately designed and supported for the prevention of detrimental flow-induced vibration and tube fretting wear at MUR uprated power conditions for the 40 year design life of the RSGs.

The concerns associated with high cycle fatigue in steam generator tube bundles addressed in NRC Bulletin 88-02 are not applicable to Byron and Braidwood Unit 1 RSGs.

References

IV.1.A.vi.1.c-1 B&W Canada Report 236R-FIV-01, Rev. 00, "Exelon, Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Flow-Induced Vibration and Wear Analysis Report," December 2010.

Chemistry

The changes in temperatures resulting from a power uprate have the potential to affect steam generator primary and secondary water chemistries. Based on the Certified Design Specification for Replacement Steam Generator (RSG) of Byron and Braidwood Stations Unit 1 (References. IV.1.A.vi.1.c-2 and IV.1.A.vi.1.c-3), the temperature range at the steam generator primary side inlet under normal operating conditions has changed from 600ºF-618.4ºF to 608.6ºF-618.4ºF at the MUR power uprate condition. The maximum primary side temperature is unchanged. The steam temperature on the RSG secondary side at MUR power uprate normal operating condition (522.1ºF to 546.9ºF) is very close to that (523.7ºF to 545.7ºF) before the uprate. These temperature changes on both primary and secondary side are

considered to be small and will not significantly affect water chemistries. Therefore, a revision to B&W recommendations on water chemistry control is not required as a consequence of the MUR.

References

IV.1.A.vi.1.c-2 Areva NP Inc. Specification 18-1229648-008, "Certified Design Specification for Replacement Steam Generator Byron and Braidwood Stations Unit 1," September 2010.

IV.1.A.vi.1.c-3 B&W Canada Report 236R-PR-01, Rev. 00, "Exelon Byron and Braidwood Stations Unit 1, Replacement Steam Generators MUR Power Uprate Thermal-Hydraulic Performance Report," October 2010.

IV.1.A.vi.1.d Steam Generator Steam Drum Evaluation

The structural evaluation of the steam drum and its internals is addressed in Section IV.1.A.vi.1.b.

IV.1.A.vi.1.e Steam Generator Mechanical Repair Hardware

Tube hardware refers to components such as plugs, sleeves, and stabilizers that are installed in the steam generators (SGs) to address tube degradation.

Evaluation results show that mechanical plug designs satisfy applicable stress, fatigue and retention acceptance criteria for operation at Measurement Uncertainty Recapture (MUR) uprate conditions. The evaluation concluded that the revised stresses were within the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code allowable values. Actual tube plugging levels are 0.08% for Byron Unit 1 and 0.32% for Braidwood Unit 1.

The fatigue usage values, when adjusted for the MUR power uprate conditions, remained less than the 1.0 fatigue limit.

The ribbed mechanical plug remains qualified for the MUR power uprate conditions. The ribbed mechanical plugs also meet the ASME Section XI IWA-4713 requirements. The evaluation of the straight leg cable stabilizers concluded that the stabilizer parameters that are affected by the MUR uprating (stability ratios, tube displacements, turbulence induced bending stresses, and fatigue) will remain within the specified acceptance criteria following the implementation of the MUR power uprate. Therefore, the straight leg cable stabilizers remain qualified for the MUR uprate conditions. The qualification of the 0.5 inch outer diameter straight leg collared-cable-stabilizer is based solely on geometric parameters and the relative wear coefficients between the stabilizer collars and the host tube materials. These parameters remain unchanged due to the MUR uprate and thus the straight leg collared-cable stabilizer remains qualified for the MUR uprate condition.

Therefore, SG repair hardware continues to meet ASME B&PV Code limits for plant operation at MUR uprate conditions.

IV.1.A.vi.1.f Steam Generator Loose Parts

Analyses have been completed to determine the effect of the proposed MUR power uprate on the potential of Foreign Objects to cause tube damage in the Byron Unit 1 and Braidwood Unit 1 Replacement Steam Generators (RSGs). These analyses were documented in the "Exelon Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Tube Damage from Foreign Objects Report," (Reference IV.1.A.vi.1.f-1).

The Thermal-Hydraulic conditions taken from the MUR Power Uprate Thermal Hydraulic Analysis Report (Reference IV.1.A.vi.1.f-2) were compared to the current power operating case (NSSS power level of 3600.6 MWt) for potential tube wear from the previous Tube Wear Analysis Report. The Foreign Object Wear Assessment shows that there is only a minor change in the potential for Foreign Object wear due to operation at MUR conditions and all acceptance criteria for the Byron and Braidwood Unit 1 RSGs continue to be met. All assessments of known objects remain acceptable for two operating cycles.

References

IV.1.A.vi.1.f-1 B&W Canada Report 236R-FIV-02, Rev. 00, "Exelon Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Tube Damage from Foreign Objects Report," December 2010.

IV.1.A.vi.1.f-2 B&W Canada Report 236R-PR-01, Rev. 00, "Exelon Byron and Braidwood Stations Unit 1, Replacement Steam Generators MUR Power Uprate Thermal-Hydraulic Performance Report," October 2010.

IV.1.A.vi.1.g Regulatory Guide 1.121 Analysis

NRC Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe tube degradation beyond which tubes found defective by in-service inspection must be repaired or removed from service. The acceptable degradation level is called the repair limit.

The Regulatory Guide 1.121 evaluation defines the structural limit for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. Steam generator (SG) tubing structural limits were determined by previous analysis (Reference IV.1.A.vi.1.g-1), for an assumed uniform thinning degradation mode in both the axial and circumferential directions. The allowable stress limits were taken from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III analysis of record (Stress Report WNET-153, Volume 5). The limiting stresses during Normal operation (Level A) and Upset (Level B) service conditions are the primary membrane stresses due to the primary-to-secondary pressure differential across the tube wall. The postulated accident condition loads for the Faulted (Level D) service condition are the loss-of-coolant-accident (LOCA), steam line break, feedline break, and design basis earthquake (DBE).

The allowable tube repair limit is established by adjusting the structural limit per Draft Regulatory Guide 1.121 to take into account uncertainties in eddy current measurement, and an operational allowance for continued tube degradation until the next scheduled inspection. Previous analyses were performed to

establish the structural limit for the tube straight-leg (free span) region for degradation over an unlimited axial extent, and for degradation over a limited axial extent at the tube support plate and anti-vibration bar intersections (Reference IV.1.A.vi.1.g-1).

Regulatory Guide 1.121 analyses have been completed for Byron and Braidwood Stations Unit 1 Replacement Steam Generators (RSGs) at the MUR power uprate conditions with a nuclear steam supply system (NSSS) power level of 3672 MWt (References IV.1.A.vi.1.g-2). These analyses were documented in the "Exelon, Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Structural Analysis Report," (Reference IV.1.A.vi.1.g-3). The analysis consists of a reconciliation to address the changes in loading conditions which occur as a consequence of MUR conditions.

It was concluded that the predicted tube leakage limits presented in the original Westinghouse analysis (Reference IV.1.A.vi.1.g-1) remain valid but the tube structural limit based on design conditions has decreased slightly at MUR power uprate design conditions. The reduction in the design structural limit is driven by the reduced secondary side minimum design pressure, however the RSGs do not operate near the design low pressure limits. Calculations based on operating conditions, where the secondary side pressures are higher than the design secondary side pressure do not result in a reduction of the structural limit. Therefore, additional structural limits were calculated for operating cases with higher secondary side pressure which may be used when appropriate.

References

IV.1.A.vi.1.g-1 Westinghouse Proprietary Report, WCAP-14977, Rev. 1, "Steam Generator Tube Plugging Limits Analysis for the Byron 1 / Braidwood 1 Replacement steam Generators."

IV.1.A.vi.1.g-2 Areva NP Inc. Specification 18-1229648-008, "Certified Design Specification for Replacement Steam Generator Byron and Braidwood Stations Unit 1," September 2010.

IV.1.A.vi.1.g-3 B&W Canada Report 236R-SR-01, Rev. 00, "Exelon, Byron and Braidwood Stations Unit 1 Replacement Steam Generators MUR Power Uprate Structural Analysis Report," December 2010.

IV.1.A.vi.2 Unit 2 Steam Generators

IV.1.A.vi.2.a Steam Generator Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluation focused on changes to secondary side operating characteristics at MUR power uprate conditions. SG secondary side performance characteristics such as steam pressure and flow, circulation ratio, bundle mixture flow, heat flux, secondary side pressure drop, moisture carryover, hydrodynamic stability, secondary side mass and others are affected by increases in power level.

Moisture carry over (MCO) was reviewed for the MUR conditions using the current plant configuration and historical test data. The review concluded that all steam generators' MCO were below the design limits. Since plant configuration changes such as steam generator tube plugging and system modifications also impact MCO, any future changes to those parameters would be evaluated at the time of the change.

Secondary side performance characteristics were calculated using the SG performance code GENP (secondary side characteristics except DNB). GENP analyses were performed for the design parameter cases. A separate analysis was performed using the three-dimensional (3-D) flow analysis code ATHOS (DNB parameters) to determine the detailed flow parameters throughout the tube bundle. The evaluation concluded that the Model D5 steam generator thermal-hydraulic operating characteristics remain acceptable for the MUR power uprate at Byron and Braidwood Units 2.

IV.1.A.vi.2.b Steam Generator Structural Integrity

The structural evaluation focused on the critical steam generator (SG) components as determined by the design basis analyses stress ratios and fatigue usages.

The structural analysis impact of the uprate on the Byron Unit 2 and Braidwood Unit 2 Model D-5 steam generators is based on changes in the pressure differential for the primary side components, and changes in the secondary side steam temperature and pressure for secondary side components with some components also affected by changes in feedwater temperature. Following a comparison of the MUR power uprate parameters to those used for the analysis-of-record, it was demonstrated that the MUR power uprate inputs are equal to, or enveloped by, those used in the analysis-of-record. Therefore, the current design basis analysis remains applicable for the MUR power uprate and the steam generator components continue to meet the ASME B&PV Code limits.

An analysis was performed to determine if the ASME B&PV Code limits on design primary-to-secondary difference in pressure (ΔP) would be exceeded for any applicable transient at power uprate conditions. The analysis for Byron Unit 2 and Braidwood Unit 2 Model D5 steam generators determined that the maximum primary-to-secondary side differential pressures during Normal operating transients are 1425 psi and 1553 psi for high T_{avg} and low T_{avg} temperatures, respectively. The maximum primary-to-secondary side differential pressures during Upset condition transients are 1629 psi and 1716 psi for high T_{avg} and low T_{avg} temperatures, respectively. These values are below the applicable design pressure limits of 1600 psi and 1760 psi for Normal and Upset conditions, respectively. Therefore, the ASME B&PV Code design pressure requirements are satisfied.

IV.1.A.vi.2.c Steam Generator Tube Bundle Integrity, Flow Induced Vibration and Wear, H*, and Chemistry

<u>Tube Integrity</u>

The Byron Unit 2 and Braidwood Unit 2 Model D5 SGs contain thermally-treated Alloy 600TT tubing and 405 stainless steel tube support plates (TSP) with broached quatrefoil holes. The quatrefoil tube hole configuration results in reduced potential for contaminant concentration at tube support plate intersections by reducing the crevice area. The first nine tube rows were heat treated after bending to relieve stresses. Hydraulic tube expansion in the tubesheet region results in reduced residual stresses compared to mechanical roll expansion and a more uniform expansion compared to explosively expanded tubes. Thermally-treated Alloy 600 is highly resistant to stress corrosion cracking. In 2003 Braidwood Unit 2 exhibited three TSP outside diameter stress corrosion cracking (ODSCC) indications after 10 cycles of operation; three other tubes were preventively plugged due to their higher potential to develop ODSCC.

There has been no recurrence of this degradation mechanism in Braidwood Unit 2 nor has Byron Unit 2 experienced TSP ODSCC.

Actual tube plugging levels are 2.08% for Byron Unit 2 and 1.42% for Braidwood Unit 2. The predominant degradation mechanisms responsible for tube plugging are mechanical wear mainly due to wear at anti-vibration bars, foreign objects interaction with straight tubes, and preventive plugging arising from pre-heater repair in one steam generator in Byron Unit 2; administrative plugging of tubes as a conservative response to non-corrosion related eddy current signals reported in top of tubesheet expansion transitions also contributed to the total. Both plants have experienced primary water stress corrosion cracking in the first inch from the hot leg tube ends. During SG monitoring and operational assessments, the degradation mechanisms cited as existing in the Model D5 SGs were wear at anti-vibration bars, wear due to foreign objects, wear at pre-heater tube intersections, and primary water stress corrosion cracking. Outside diameter stress corrosion cracking has not occurred in six of the eight SGs Byron Unit 2 and Braidwood Unit 2; pitting has not been observed in any of the eight (8) SGs. These potential mechanisms are nevertheless consistently addressed in the inspection planning for each SG.

On the basis of T_{hot} increase alone, the mechanical wear processes are predicted to be insignificant. The increased reactor coolant system (RCS) temperature effects on primary water stress corrosion cracking are predicted to be negligible because of the licensing of alternate repair criteria (H*), an alternate basis for tube plugging for flaws found in a hydraulically expanded tube/tubesheet joint. The small RCS temperature increases contemplated for the MUR power uprate are predicted to cause insignificant change in the rates of primary water stress corrosion cracking initiation and propagation; the licensing of H* alternate repair criteria on a permanent basis would reduce the plugging of tubes due to primary water cracking in the hot leg tubesheet, the only region to exhibit such cracking in the Byron Unit 2 and Braidwood Unit 2 steam generators.

Growth rates of currently observed tube wear mechanisms at Byron Unit 2 and Braidwood Unit 2 may be slightly increased; however, the magnitude of this increase is sufficiently small that SG tube integrity performance criteria defined by Reference IV.1.A.vi.2-1.c will not be challenged under the MUR power uprate conditions.

Comparisons with industry predictions for Model D5 SGs equipped with Alloy 600TT tubes are favorable with respect to Byron Unit 2 and Braidwood Unit 2.

Reference

IV.1.A.vi.2-1.c NEI 97-06, Revision 2, “Steam Generator Program Guidelines,” Nuclear Energy Institute, May 2005.

<u>Flow Induced Vibration and Wear</u>

The effect of operating the Byron and Braidwood Unit 2 Model D5 steam generators at MUR power uprate conditions was evaluated for several issues associated with FIV and tube wear. These include:

- Tube stability ratio, peak turbulent displacements and vortex shedding
- Anti-Vibration Bar (AVB) wear of inactive tubes

- AVB wear of active tubes
- Wear of preheater tubes
- Loose part wear in the downcomer, preheater and at the top of tubesheet (TTS)
- High cycle fatigue

Results show that there will be small increases in the tube stability ratio, peak turbulent displacements and vortex shedding displacements but all remain within specified acceptance criteria. The wear rate of inactive tubes from AVBs is expected to increase slightly once MUR power uprate conditions are implemented. However, the number of tubes that will require monitoring during scheduled inspections is small and therefore acceptable. Power uprate growth rate (PUGR) factors that can be applied to tube wear at various locations in the tube bundle for eight different MUR uprate operating conditions were also calculated. The PUGR factors address AVB wear of active tubes and wear of preheater tubes from increased feedwater flow. They also address loose part wear in the preheater, at the top of the tubesheet and in the downcomer region. High cycle fatigue in the upper tube bundle was also addressed at MUR power uprate conditions in accordance with NRC Bulletin 88-02. Based on evaluations previously performed for Byron and Braidwood Unit 2, there are no concerns that high cycle fatigue will occur while operating at MUR power uprate conditions. Therefore, operation at MUR power uprate conditions will not result in rapid rates of tube wear or high levels of tube vibration in the steam generator tube bundle.

H* Evaluation

The key operating parameters associated with the MUR power uprate were evaluated for H* lengths and leakage factors. It was concluded that there is no impact on the H* lengths or leakage factors at MUR power uprate conditions for the Byron and Braidwood Unit 2 Model D5 Steam Generators.

Chemistry

An evaluation considering the Byron and Braidwood Strategic Water Plans required by the EPRI Guidelines and the design parameters specific for Byron Unit 2 and Braidwood Unit 2 D-5 steam generators was performed to assess the potential for changes in steam generator chemistry due to MUR power uprate. The scope is limited to the chemistry of the bulk water in the steam generators and does not include any fuel considerations or other primary system considerations.

No significant changes in the bulk steam generator water chemistry of either the primary or secondary side are expected due to the uprating because the bulk chemistry will continue to be controlled after the MUR power uprate by plant procedures and specifications conforming to industry accepted guidelines and embodied in the Primary and Secondary Strategic Water Chemistry Plans for Re-circulating Steam Generator Plants. In addition, design temperatures are in the range where other plants control bulk chemistry based on the same industry guidelines.

IV.1.A.vi.2.d Steam Generator Steam Drum Evaluation

Erosion-corrosion has been detected in several components of the Byron Unit 2 and Braidwood Unit 2 Model D5 steam generator's steam drum internals, and estimates of the rates of degradation are made by

comparing the results of sequential inspections. Observed measurement variability indicates that there may be re-deposition of magnetite on the back surface of the components and that there is a considerable difficulty in making measurements of the thickness of these components. However, it is clear from the inspection data obtained that thinning is occurring in some upper internal components.

Erosion-corrosion in the SG steam drum region depends on numerous factors, including material composition, fluid velocity and turbulence, and secondary side water chemistry. Due to the increased steam flows at MUR power uprate conditions, the fluid velocity is the variable of interest following uprate. The increased velocities at MUR power uprate conditions are estimated to increase current estimated degradation rates up to 25%. Because the degradation rate may increase under MUR power uprate conditions, continued careful monitoring is required. Exelon will continue to perform periodic steam drum component inspections to evaluate the impact of any potential accelerated wear rates in the steam drum.

IV.1.A.vi.2.e Steam Generator Mechanical Repair Hardware

Tube hardware refers to components such as plugs, sleeves, and stabilizers that are installed in the steam generators (SGs) to address tube degradation.

Evaluation results show that mechanical plug designs satisfy applicable stress, fatigue and retention acceptance criteria for operation at Measurement Uncertainty Recapture (MUR) power uprate conditions. There are no Alloy 600 ribbed mechanical plugs in either Byron Unit 2 or Braidwood Unit 2 and no Alloy 600 ribbed mechanical or welded plugs will be installed in the future, so existing NRC rules on Alloy 600 tube plugs are not applicable. The NPT-88 field installed weld plug may be used in applications that cannot employ a mechanical plug. Both the NPT-23 (a tapered plug), and NPT-88 (a thimble plug) shop and field weld plugs remain qualified at the MUR power uprate conditions. Field machining SG tube ends is a possibility for modifications and tube repair (i.e., plugging, sleeving, and tube end reopening). The evaluation concluded that the revised stresses were within the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code allowable values.

The fatigue usage values, when adjusted for the MUR power uprate conditions, remained less than the 1.0 fatigue limit.

The evaluation of the straight leg cable stabilizers concluded that the only stabilizer parameters that are affected by the MUR power uprate (stability ratio and tube displacements) will remain within the specified acceptance criteria following the implementation of the MUR power uprate. Therefore, the straight leg cable stabilizers remain qualified for the MUR power uprate conditions. The qualification of the 0.5 inch outer diameter straight leg collared-cable-stabilizer is based solely on geometric parameters and the relative wear coefficients between the stabilizer collars and the host tube materials. These parameters remain unchanged due to the MUR power uprate and thus the straight leg collared-cable-stabilizer remains qualified for the MUR power uprate condition.

Therefore, SG repair hardware continues to meet ASME B&PV Code limits for plant operation at MUR uprate conditions.

IV.1.A.vi.2.f Steam Generator Loose Parts

An evaluation was done to determine the effect of the 3672 MWt NSSS Power MUR power uprate on Byron Unit 2 and Braidwood Unit 2 D5 steam generators loose parts.

The revised wear time calculations for the limiting objects in the pre-heater location and tubesheet location for each steam generator are summarized. These are evaluated taking the limiting objects for the pre-heater location and tubesheet location for each steam generator from the B2R15 spring 2010 Byron Unit 2 outage, A2R14 fall 2009 Braidwood Unit 2 outage and the A2R12 fall 2006 Braidwood Unit 2 outage. The wear times are compared based on calculations before and after the uprate. The wear times after the uprate remain greater than or equal to two fuel cycles (3 years).

The previous loose part evaluations were reviewed to determine the power uprate effects on the objects projected wear times. Although there was no indication of wear present on any tubes adjacent to the limiting foreign objects, the wear time analyses were performed by conservatively assuming 20% initial tube wear on the limiting tube location. The steam generator secondary side conditions will change as a result of the MUR power uprate operating conditions, however, these changes do not affect the previous evaluation conclusions. The operation at the MUR power uprate conditions is acceptable.

The analysis determined that the amount of time required for the limiting foreign object orientation to wear a tube down to a minimum allowable tube wall thickness under conservative secondary side conditions is greater than or equal to 3 years or 2 operational cycles.

A review of outage close-out letters for Byron Unit 2 reveals that some existing objects in the steam generators have caused wear on the tubing during past cycles. These objects are termed “unknown objects or inaccessible objects” since the support plates locations are difficult to access. It is determined that even with the change in conditions due to the MUR power uprate, the inspection criteria for these objects can remain the same as previously defined in the closeout letters. Thus, the disposition of the foreign objects is not affected by the MUR power uprate.

IV.1.A.vi.2.g Regulatory Guide 1.121 Analysis

NRC Draft Regulatory Guide 1.121 describes an acceptable method for establishing the limiting safe tube degradation beyond which tubes found defective by in-service inspection must be repaired or removed from service. The acceptable degradation level is called the repair limit.

The Regulatory Guide 1.121 evaluation defines the structural limit for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. Steam generator (SG) tubing structural limits were determined by previous analysis, for an assumed uniform thinning degradation mode in both the axial and circumferential directions. The allowable stress limits were taken from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III analysis of record (Stress Report WNET-153, Volume 5). The limiting stresses during Normal operation (Level A) and Upset (Level B) service conditions are the primary membrane stresses due to the primary-to-secondary pressure differential across the tube wall. The postulated accident condition loads for the Faulted (Level D) service condition are the loss-of-coolant-accident (LOCA), steam line break, feedline break, and design basis earthquake (DBE).

The allowable tube repair limit is established by adjusting the structural limit per Draft Regulatory Guide 1.121 to take into account uncertainties in eddy current measurement, and an operational allowance for continued tube degradation until the next scheduled inspection. Previous analyses were performed to establish the structural limit for the tube straight-leg (free span) region for degradation over an unlimited axial extent and for degradation over a limited axial extent at the tube support plate and anti-vibration bar intersections. All of the loading conditions considered in the Regulatory Guide 1.121 analysis to determine the tube structural limits are unchanged from those utilized in the analysis of record. Therefore, the analysis of record remains valid and the existing structural limits continue to apply. The existing tube repair limit is unaffected by the MUR power uprate and remains valid at uprate conditions.

IV.1.A.vii Reactor Coolant Pumps and Reactor Coolant Pump Motors

Revised RCS conditions were reviewed for impact on the existing RCP design basis analyses. The NSSS design parameters considered in the RCP evaluation are the pump inlet temperature and RCS pressure. The pump inlet temperature (equivalent to the SG outlet temperature) is considered because the RCP design specification lists a specific value for inlet temperature. No changes in RCS design or operating pressure were made as part of the MUR power uprate. The maximum steam generator outlet temperature for any NSSS design parameters case is 554.8°F. This temperature is lower than the pump inlet temperature of 556.7°F considered in the RCP design specification and the existing analysis of the RCPs. Due to lower allowable design stress limits, higher temperatures are more limiting for RCP structural design qualification and the NSSS parameter change for the MUR power uprate is therefore conservative. The MUR power uprate conditions remain bounded by the original design conditions and previously evaluated conditions. The existing NSSS primary side design transients that have previously been evaluated for Byron and Braidwood Units 1 and 2 remain valid for the MUR power uprate. There are also no changes to nozzle or support foot loads for the MUR power uprate that would affect the existing RCP structural analyses.

The RCP motors were evaluated for horsepower loading at continuous hot and cold operation, starting ability of the motor, and loads on the thrust bearings. The RCP motors are acceptable for operation at MUR power uprate conditions. The maximum pump brake horsepower at hot loop condition for the MUR power uprate remains below the nameplate rating of the motor. Revised horsepower loading at cold loop operation will cause only a minimal impact to the insulation life and adequate service life remains. Previous evaluations for the Byron and Braidwood Units 1 and 2 motors evaluated the starting ability under cold conditions and minimum voltage against reverse flow. This evaluation remains applicable for the revised RCS conditions. Changes in thrust loads due to the MUR power uprate were concluded to be minor in comparison to the available stress margin in the bearing shoes, and are therefore acceptable for MUR power uprate conditions.

The revised RCS conditions are acceptable for the RCP with respect to ASME B&PV Code structural integrity. The original code of record, 1971 Edition with Addenda through Winter 1972, remains unchanged (Table IV.1.D-1). Therefore, the revised MUR power uprate conditions remain bounded by the previously evaluated design conditions.

IV.1.A.viii Pressu rizer Structural Evaluation

The Measurement Uncertainty Recapture (MUR) power uprate operating conditions were reviewed for impact on the existing pressurizer design basis analysis. The limiting pressurizer conditions occur when the Reactor Coolant System (RCS) pressure is high and the RCS T_{hot} and T_{cold} are low. No changes were made in RCS design or operating pressure as part of the power uprate. The minimum T_{hot} and T_{cold} values from the design parameter cases were used in the pressurizer evaluation. At the normal operating pressure of 2250 psia, the revised T_{hot} and T_{cold} temperature differences for normal operation are bounded by the original analysis.

The Nuclear Steam Supply System (NSSS) design transients did not change and were enveloped by the existing design transients. Pressure fluctuations during the uprate transients are the same as the original evaluations. The maximum pressure within each load category (Normal, Upset, Faulted and Test) has not changed from the value used in the original evaluations. Thus, the uprate transients have no effect on the primary stress evaluations previously performed. The Byron and Braidwood pressurizer lower heads were previously evaluated for insurge/outsurge transient effects related to both design transients and operational transients that were not considered in the original design. The revised design parameters were evaluated for their effect on the previous evaluation conclusions. The revised design parameters have an insignificant impact on the previous fatigue results and they remain valid. Therefore, the pressurizer meets the stress/fatigue analysis requirements for plant operation at the MUR power uprate conditions. The codes of record are listed in Table IV.1.D-1.

The effect of the Byron and Braidwood Units 1 and 2 MUR power uprate on pressurizer nozzle weld overlays was evaluated. That evaluation determined that the MUR power uprate has a negligible impact on the qualification of the pressurizer surge, spray, safety and relief nozzle Structural Weld Overlay (SWOL) designs.

IV.1.A.ix Safety Related Valves

The revised operating conditions were reviewed for impact on the design basis of existing safety-related valves. No changes in RCS design or operating pressure were made as part of the power uprate. The evaluations concluded that the temperature changes due to the power uprate have, at most, an insignificant effect on the differential pressures used in the existing analyses. Safety-related valves were reviewed within the applicable system (Section VI) and program (Section VII.6.E) evaluations. None of the safety-related valves required a change to their design or operation as a result of the MUR power uprate.

IV.1.A.x Loop Stop Isolation Valves

The revised design conditions were reviewed for impact on the existing loop stop isolation valve design basis analyses previously performed. No changes in previously evaluated RCS design or operating pressure were made as part of the power uprate. The loop stop isolation valves are located in each RCS hot leg and cold leg. Higher temperatures are more limiting for the design qualification, so the hot leg valves were chosen to bound both applications. The maximum allowable T_{hot} is limited to 618.4°F. This value is the overall limiting temperature for all the components that are subjected to RCS operating conditions. Thus, the limiting hot leg temperature is bounded by the design evaluations previously

performed for the loop stop isolation valve. In addition, the T_{hot} variations as they presently exist in the component analyses are conservative and bounding.

In addition to the evaluation for the stresses, the MUR transients were compared to the current transients used in the fatigue analysis. Based on the review the current transients bound the MUR transients. Therefore the MUR transients are acceptable.

Therefore, the previously analyzed loop stop isolation valve evaluations remain bounding and applicable to the design parameters and NSSS design transients at MUR power uprate conditions. The code of record remains unchanged and is listed in Table IV.1.D-1.

IV.1.B.i Stress es

Evaluations were performed to demonstrate that the revised design conditions for the NSSS components, piping, and interface systems were within the existing structural design basis analyses. Stress evaluations are discussed in Sections IV.1.A.i (Reactor Vessel), IV.1.A.ii (Reactor Vessel Internals), IV.1.A.iii (Control Rod Drive Mechanism), IV.1.A.iv (Reactor Coolant Piping and Supports), IV.1.A.v (BOP Piping), IV.1.A.vi (Steam Generators), IV.1.A.vii (Reactor Coolant Pumps and Reactor Coolant Motors), IV.1.A.viii (Pressurizer Structural Evaluation), IV.1.A.ix (Safety-Related Valves), and IV.1.A.x (Loop Stop Isolation Valves).

IV.1.B.ii Cumulative Usage Factors

Evaluations were performed to demonstrate that the revised design conditions for the NSSS components, piping, and interface systems were within the existing structural design basis analyses. Cumulative usage factors (fatigue evaluations) are discussed in Sections IV.1.A.i (Reactor Vessel), IV.1.A.ii (Reactor Vessel Internals), IV.1.A.iv (Reactor Coolant Piping and Supports), IV.1.A.vi (Steam Generators), IV.1.A.vii (Reactor Coolant Pumps and Reactor Coolant Motors), IV.1.A.viii (Pressurizer Structural Evaluation), and IV.1.A.x (Loop Stop Isolation Valves).

IV.1.B.iii Flow-Induced Vibration

SG flow-induced vibration (FIV) is discussed in Section IV.1.A.vi.1.c for Unit 1 Steam Generators and in Section IV.1.A.vi.2.c for Unit 2 Steam Generators. Reactor vessel internal components were also evaluated for FIV impact under MUR power uprate conditions and found to be acceptable.

IV.1.B.iv Temp erature Effects

IV.1.B.iv.1 Changes in Temperature (Pre- and Post-uprate)

Calculations were completed to define the RCS and SG design conditions for the Byron and Braidwood MUR power uprate. The operating temperature changes are shown in LAR Attachment 1 Table 3-1 for Byron and Braidwood Stations Unit 1 and Table 3-2 for Byron and Braidwood Stations Unit 2. Specific calculation outputs include T_{hot} and T_{cold}. The current T_{avg} window has been maintained at 575°F-588°F. There is an approximate 1.20°F increase in temperature across the core (T_{hot} increases approximately

0.6°F and T_{cold} decreases approximately 0.6°F) from current operating conditions due to the power uprate. There is no change to the RCS average temperature limit in Technical Specification 3.4.1 and the COLR.

Changes in main steam and feedwater system temperatures are discussed in Sections VI.1.A.i and VI.1.A.iv respectively.

IV.1.B.iv.2 Evaluation of Potential for Thermal Stratification

NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems (RCS)," addresses thermal stresses in piping attached to the RCS that cannot be isolated. This bulletin is mentioned because it introduces the issue of thermal stratification; however the surge line falls under Bulletin 88-11.

NRC Bulletin No. 88-11, "Pressurizer Surge Line Thermal Stratification," addresses surge line thermal stratification. Surge line thermal stratification is driven by the temperature difference between the RCS hot leg and the pressurizer. The current hot leg operating temperatures for the upper and lower bound cases (based on different levels of steam generator tube plugging) will either stay the same or increase by 0.6°F for the proposed MUR power uprate operating conditions. A higher hot leg temperature lowers the temperature differential between the hot leg and pressurizer, which reduces the stratification effects. There are no significant changes to the surge line operating conditions and therefore no significant changes to the pressurizer stratification loading.

IV.1.B.v Changes in Pressure (pre-and post-uprate)

Calculations were completed to define the RCS and SG conditions for Byron and Braidwood Stations MUR power uprate. There will be no change in RCS operating pressure as a result of the MUR power uprate. The nominal operating pressure is 2250 psig (LAR Attachment 1, Table 3-1). There is no change to the RCS pressure limit in Technical Specifications 2.1.2 or 3.4.1.

Changes in main steam and feedwater system pressure, as well as other NSSS interface systems, are discussed in Section VI.1.A.

IV.1.B.vi Changes in Flow Rates (pre- and post-uprate)

Calculations were completed to define the RCS and SG conditions for Byron and Braidwood Stations MUR power uprate. The mechanical design RCS flow is shown in LAR Attachment 1, Table 3-1 and remains unchanged for the power uprate. As discussed in LAR Attachment 1 the minimum RCS flow given in Technical Specification 3.4.1 is being increased from 380,900 gpm to 386,000 gpm to address revised DNBR analyses conditions.

Changes in main steam and feedwater system flow rates, as well as other NSSS interface systems, are discussed in Section VI.1.A.

IV.1.B.vii High-Energy Line Break

IV.1.B.vii.1 High Energy Line Break Locations

A review was performed to determine the power uprate impact on high energy line break (HELB) program. MUR power uprate operating temperatures, pressures, and mass flow rates were compared to the analyzed conditions. The review concluded that overall, the total pipe stresses were not significantly impacted. Therefore, the MUR power uprate does not result in any new or revised pipe break locations, and the existing design basis for pipe break, jet impingement and pipe whip remains valid.

IV.1.B.vii.2 Leak-Before-Break Evaluation

The existing leak-before-break (LBB) analyses justified eliminating large primary loop pipe rupture from the Byron and Braidwood Units 1 and 2 Nuclear Power Plants structural design basis in WCAP-14559 Revision 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants " (Reference IV.1.B.vii.2-1). The applicable pipe loadings, normal operating pressure, and temperature parameters at Byron and Braidwood Units 1 and 2 for MUR power uprate conditions were used to evaluate LBB. The LBB acceptance criteria are based on Nuclear Regulatory Commission Standard Review Plan, Section 3.6.3, "Leak-Before-Break Procedures" (Reference IV.1.B.vii.2-2). The LBB acceptance criteria are satisfied for the Byron and Braidwood Units 1 and 2 primary loop piping for the MUR power uprate conditions. All the recommended margins are satisfied, and the existing analyses conclusions remain valid. It is therefore concluded that the dynamic effects of the primary loop piping breaks for Byron and Braidwood Units 1 and 2 need not be considered in the structural design basis at the MUR power uprate conditions.

References

IV.1.B.vii.2-1 WCAP-14559 Revision 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Byron and Braidwood Units 1 and 2 Nuclear Power Plants," April 1996.

IV.1.B.vii.2-2 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol.52, No. 167/ Notices, pp. 32626-32633/Friday, August 28, 1987

IV.1.B.viii LOCA Forces Including Jet Impingement and Thrust

A LOCA hydraulic forces analysis generates the hydraulic forcing functions and hydraulic loads that occur on RCS components due to a postulated LOCA. No changes in RCS design or operating pressure were made as part of the MUR power uprate**.** LOCA hydraulic forces increase with lower temperatures, so they are predominantly influenced by T_{cold}. The currently supported operating conditions for LOCA hydraulic forces on the Byron and Braidwood Units 1 and 2 loop piping and steam generators were evaluated to be sufficient to address the proposed initial conditions for the MUR power uprate based on conservatisms in these analyses. The currently supported operating conditions for LOCA hydraulic forces on the Byron and Braidwood Units 1 and 2 vessel/internals were also evaluated to be sufficient to address the proposed initial conditions for the MUR power uprate based on conservatisms in these analyses.

Therefore, the analyses for vessel/internals, loop, and steam generator LOCA hydraulic forcing functions described above remain valid for the MUR design conditions. There are no changes to methodology or results with respect to LOCA hydraulic forces as a result of the MUR conditions.

IV.1.B.ix Seismi c Qualification

Byron and Braidwood safety-related structures, systems and components are designed for seismic events as described in UFSAR Sections 3.2, 3.7, 3.8, and 3.10. The primary input motions due to the design basis earthquake are not affected by the MUR PU. Seismic design is not impacted, because seismic requirements remain unchanged. Therefore, the seismic qualification of essential equipment supports is unaffected.

The mechanical and electrical equipment seismic qualification review demonstrated that the equipment will continue to meet the current licensing basis.

IV.1.C Reactor Vessel Integrity

IV.1.C.i Pressu rized Thermal Shock

Byron Units 1 and 2

The Pressurized Thermal Shock (PTS) evaluation provides a means for assessing the susceptibility of reactor vessel beltline materials to failure during a PTS event, to ensure that adequate fracture toughness exists during reactor operation. 10 CFR 50.61 (Reference IV.1.C.i-1) provides the requirements, methods of evaluation, and safety criteria for PTS assessments.

PTS screening calculations were performed for the Byron Units 1 and 2 reactor vessel beltline materials using the current 40 year end of license (EOL) neutron fluence values. It was determined that all the Byron Units 1 and 2 reactor vessel beltline materials will continue to meet the 10 CFR 50.61 PTS screening criteria (270°F for plates, forgings, and axial welds, and 300°F for circumferential welds). For Byron Unit 1, the limiting RT_{PTS} value for the forgings is 109°F, which corresponds to the Intermediate Shell Forging (using non-credible surveillance data). For Byron Unit 2, the limiting RT_{PTS} value for the forgings (using credible surveillance data) is 62°F, which corresponds to the Nozzle Shell Forging. The limiting circumferential weld material is the Intermediate to Lower Shell Forging Circumferential Weld Seam (using credible surveillance data) with RT_{PTS} values of 74°F and 114°F for Byron Units 1 and 2, respectively. These limiting materials are unchanged from those provided in the Byron Units 1 and 2 respective Pressure and Temperature Limits Report (References IV.1.C.i-2 and IV.1.C.i-3).

The PTS screening calculations performed at the end of the current operating license result in RT_{PTS} values that are consistent with those documented in the vessel integrity analyses of record. The MUR power uprate has no impact on 10 CFR 50.61 compliance. The reactor vessels will remain within their PTS limits after implementation of the MUR power uprate.

References

IV.1.C.i-1 Code of Federal Regulations, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

IV.1.C.i-2 Pressure and Temperature Limits Report, "Byron Unit 1 Pressure and Temperature Limits Report (PTLR)," December 2006.

IV.1.C.i-3 Pressure and Temperature Limits Report, "Byron Unit 2 Pressure and Temperature Limits Report (PTLR)," December 2006.

Braidwood Units 1 and 2

The Pressurized Thermal Shock (PTS) evaluation provides a means for assessing the susceptibility of reactor vessel beltline materials to failure during a PTS event, to ensure that adequate fracture toughness exists during reactor operation. 10 CFR 50.61 (Reference IV.1.C.i-4) provides the requirements, methods of evaluation, and safety criteria for PTS assessments.

PTS screening calculations were performed for the Braidwood Units 1 and 2 reactor vessel beltline materials using the current 40-year end-of-license (EOL) neutron fluence values. It was determined that all the Braidwood Units 1 and 2 reactor vessel beltline materials will continue to meet the 10 CFR 50.61 PTS screening criteria (270°F for plates, forgings, and axial welds, and 300°F for circumferential welds). For Braidwood Unit 1, the limiting RT_{PTS} value for the forgings (using credible surveillance data) is 54°F, which corresponds to the Nozzle Shell Forging. For Braidwood Unit 2, the limiting RT_{PTS} value for the forgings (using non-credible surveillance data) is 74°F, which also corresponds to the Nozzle Shell Forging. The limiting circumferential weld material is the Intermediate to Lower Shell Forging Circumferential Weld Seam (using credible surveillance data) with a RT_{PTS} value of 98°F for both Braidwood Units 1 and 2. These limiting materials are unchanged from those provided in the Braidwood Units 1 and 2 respective Pressure and Temperature Limits Report (References IV.1.C.i-5 and IV.1.C.i-6).

The PTS screening calculations performed at the end of the current operating license result in RT_{PTS} values that are consistent with those documented in the vessel integrity analyses of record. The MUR power uprate has no impact on 10 CFR 50.61 compliance. The reactor vessels will remain within their PTS limits after implementation of the MUR power uprate.

References

IV.1.C.i-4 Code of Federal Regulations, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

IV.1.C.i-5 Pressure and Temperature Limits Report, "Braidwood Unit 1 Pressure and Temperature Limits Report (PTLR)," Revision 4.

IV.1.C.i-6 Pressure and Temperature Limits Report, "Braidwood Unit 2 Pressure and Temperature Limits Report (PTLR)," Revision 4.

IV.1.C.ii Fluence Evaluation

Fluence calculations were based on the NRC-approved methodologies described in References IV.1.C.ii-1 and IV.1.C.ii -2. These methodologies follow the guidance and meet the requirements of Regulatory Guide 1.190 (IV.1.C.ii-3). The evaluation complies with Regulatory Guide 1.190, because the acceptance criteria are derived directly from Regulatory Guide 1.190, Section 1.4.3. This section states that a vessel fluence uncertainty of 20% (one sigma, 1σ) is acceptable for RT_{PTS} and RT_{NDT} determination. The NRC-approved methodology used for Byron Units 1 and 2 and Braidwood Units 1 and 2 fluence evaluations has been demonstrated to satisfy this criterion. The Regulatory Guide 1.190 specific requirements incorporated in this methodology are:

- The calculations use neutron transport cross sections from the Evaluated Nuclear Data Files (ENDF/B-VI).
- A P5 expansion of the scattering cross sections is used in the discrete ordinates calculations. This exceeds the minimum requirement of Regulatory Guide 1.190.
- An S16 order of angular quadrature is used in the discrete ordinates calculations. This exceeds the minimum requirement of Regulatory Guide 1.190.
- An uncertainty analysis that included calculation comparisons with test and power reactor benchmarks and an analytical uncertainty study has been completed and documented in NRC-approved topical reports. The transport calculations' overall uncertainty was demonstrated to be 13% (one sigma, 1σ). This uncertainty level meets the Regulatory Guide 1.190 requirement of 20% (one sigma, 1 σ).

Byron Unit 1

The calculations for Cycles 1 through 16 (20.2 EFPY) represent the neutron exposure to the pressure vessel and surveillance capsules based on spatial power distribution and a core power as follows:

- Cycles 1 through 10 – 3411 MWt
- Cycle 11 – 3518.4 MWt
- Cycles 12 through 16 – 3586.6 MWt

A previous power uprate from 3411 MWt to 3518.6 MWt occurred during Cycle 11. The power level listed above for Cycle 11 (3518.4 MWt) represents a time-weighted average of 3411 MWt and 3586.6 MWt.

Projections beyond Cycle 16 were based on a bounding uprated core power level of 3658 MWt and the uprate fuel cycle design.

Byron Unit 2

The calculations for Cycles 1 through 15 (20.1 EFPY) represent the neutron exposure to the pressure vessel and surveillance capsules based on spatial power distribution and a core power as follows:

- Cycles 1 through 9 – 3411 MWt
- Cycle 10 – 3583.5 MWt

- Cycles 11 through 15 – 3586.6 MWt

A previous power uprate from 3411 MWt to 3586.6 MWt occurred during Cycle 10. The power level listed above for Cycle 10 (3583.5 MWt) represents a time-weighted average of 3411 MWt and 3586.6 MWt.

Projections beyond Cycle 15 were based on a bounding uprated core power level of 3658 MWt and the uprate fuel cycle design.

Braidwood Unit 1

The calculations for Cycles 1 through 14 (17.7 EFPY) represent the neutron exposure to the pressure vessel and surveillance capsules based on spatial power distribution and a core power as follows:

- Cycles 1 through 8 – 3411 MWt
- Cycle 9 – 3458 MWt
- Cycles 10 through 14 – 3586.6 MWt

A previous power uprate from 3411 MWt to 3586.6 MWt occurred during Cycle 9. The power level listed above for Cycle 9 (3458 MWt) represents a time-weighted average of 3411 MWt and 3586.6 MWt.

Projections beyond Cycle 14 were based on a bounding uprated core power level of 3658 MWt and the uprate fuel cycle design.

Braidwood Unit 2

The calculations for Cycles 1 through 14 (18.4 EFPY) represent the neutron exposure to the pressure vessel and surveillance capsules based on spatial power distribution and a core power as follows:

- Cycles 1 through 8 – 3411 MWt
- Cycle 9 – 3528 MWt
- Cycles 10 through 14 – 3586.6 MWt

A previous power uprate from 3411 MWt to 3586.6 MWt occurred during Cycle 9. The power level listed above for Cycle 9 (3528 MWt) represents a time-weighted average of 3411 MWt and 3586.6 MWt.

Projections beyond Cycle 14 were based on a bounding uprated core power level of 3658 MWt and the uprate fuel cycle design.

Peak fast neutron fluence (E > 1.0 MeV) values for all Byron/Braidwood units were provided to the NRC in IV.1.C.ii-4. The peak reactor vessel inner surface fluence (E > 1.0 MeV) values reported in Reference IV.1.C.ii-4 and the MUR power uprate fluence values for the same time period are shown in Table IV.1.C.ii-1.

The previously calculated maximum fluence values are conservative (higher in value) compared to those calculated for the MUR power uprate to 3658 MWt starting at Byron 1 Cycle 17, at Byron 2 Cycle 16, at Braidwood 1 Cycle 15, and at Braidwood 2 Cycle 15.

Given the uncertainties associated with the two NRC-approved methodologies, both analyses meet the 20% (one sigma, 1σ) Regulatory Guide 1.190 (Reference IV.1.C.ii -3) requirement. Therefore, the results of either calculation are acceptable.

References

IV.1.C.ii -1 WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., May 2004.

IV.1.C.ii -2 WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," S. L. Anderson, May 2006.

IV.1.C.ii -3 Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

IV.1.C.ii -4 RS-00-38, Letter from R. M. Krich to USNRC Document Control Desk, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," July 2000.

IV.1.C.ii -5. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

Table IV.1.C.ii-1 Peak Reactor Vessel Inner Surface Fluence

Unit	Maximum Analysis of Record Fluence	Methodology	Years Exposed	MUR Maximum Fluence	Methodology	Years Exposed
Byron 1	2.02 E19 n/cm^2	Reference IV.1.C.ii-5	32 EFPY	1.77 E19 n/cm^2	References IV.1.C.ii-1 and IV.1.C.ii-2	32 EFPY
Byron 2	2.06 E19 n/cm^2	Reference IV.1.C.ii-5	32 EFPY	1.76 E19 n/cm^2	References IV.1.C.ii-1 and IV.1.C.ii-2	32 EFPY
Braidwood 1	2.05 E19 n/cm^2	Reference IV.1.C.ii-5	32 EFPY	1.76 E19 n/cm^2	References IV.1.C.ii-1 and IV.1.C.ii-2	32 EFPY
Braidwood 2	1.96 E19 n/cm^2	Reference IV.1.C.ii-5	32 EFPY	1.73 E19 n/cm^2	References IV.1.C.ii-1 and IV.1.C.ii-2	32 EFPY

IV.1.C.iii Heatup and Cooldown Pressure/Temperature Limit Curves

Byron Units 1 and 2

10 CFR 50, Appendix G (Reference IV.1.C.iii-3) provides fracture toughness requirements for ferritic low alloy steel or carbon steel materials in the reactor coolant system pressure boundary. It also includes the requirements on Upper-Shelf Energy values used for assessing the safety margins of reactor vessel materials against ductile tearing, and for calculating plant pressure-temperature (P-T) limits. These P-T limits are established to ensure the structural integrity of reactor coolant system pressure boundary ferritic components during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

The current heatup and cooldown curves (Pressure and Temperature Limits Report (PTLR) Figures 2.1 and 2.2 (References IV.1.C.iii-1 and IV.1.C.iii -2) are licensed through the first 32 effective full power years (EFPY) for Byron Units 1 and 2. Adjusted Reference Temperature (ART) or RT_{NDT} calculations have been performed per Regulatory Guide 1.99, Revision 2 (Reference IV.1.C.iii-4) for the Byron Units 1 and 2 reactor vessel beltline materials at the EOL neutron fluence values corresponding to 32 EFPY. The fluence methodology follows the guidance and meets the requirements of Regulatory Guide 1.190 (Reference IV.1.C.iii-5). Furthermore, the reactor vessel inlet temperatures for Byron Units 1 and 2 remain within the accepted range identified in Regulatory Guide 1.99, Revision 2, Position 1.3. Therefore, the embrittlement correlations in the Regulatory Guide used to perform the ART calculations are applicable to the Byron Units 1 and 2 reactor vessels for the MUR uprate program.

For Unit 1, the limiting ART values used in the development of the current P-T limit curves at 32 EFPY bound the MUR power uprate limiting ART values (at 32 EFPY). Therefore, the current heatup and cooldown curves for Byron Unit 1 are valid through EOL (32 EFPY) with the MUR power uprate and do not require an update, because the limiting ART values from which the curves were developed remain applicable.

For Unit 2, the limiting ART values used in the development of the current P-T limit curves at 32 EFPY are slightly lower than the MUR power uprate limiting ART values (at 32 EFPY). Therefore, the applicability date for which the current heatup and cooldown curves for Byron Unit 2 were developed decreased from 32 EFPY to 30.5 EFPY. The Byron Unit 2 PTLR will be updated to reflect the new applicability date of 30.5 EFPY for both the heatup and cooldown limit curves.

References

IV.1.C.iii -1 Pressure and Temperature Limits Report, “Byron Unit 1 Pressure and Temperature Limits Report (PTLR),” December 2006.

IV.1.C.iii -2 Pressure and Temperature Limits Report, “Byron Unit 2 Pressure and Temperature Limits Report (PTLR),” December 2006.

IV.1.C.iii -3 Code of Federal Regulations, 10 CFR 50, Appendix G, “Fracture Toughness Requirements.”

IV.1.C.iii -4 NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

IV.1.C.iii -5 NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

Braidwood Units 1 and 2

10 CFR 50, Appendix G (Reference IV.1.C.iii-8) provides fracture toughness requirements for ferritic low alloy steel or carbon steel materials in the reactor coolant system pressure boundary. It also includes the requirements on Upper-Shelf Energy values used for assessing the safety margins of reactor vessel materials against ductile tearing, and for calculating plant pressure-temperature (P-T) limits. These P-T limits are established to ensure the structural integrity of reactor coolant system pressure boundary ferritic components during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

The current heatup and cooldown curves (Pressure and Temperature Limits Report (PTLR) Figures 2.1 and 2.2 (References IV.1.C.iii-6 and IV.1.C.iii-7) are licensed through the first 32 effective full power years (EFPY) for Braidwood Units 1 and 2. Adjusted Reference Temperature (ART) or RT_{NDT} calculations have been performed per Regulatory Guide 1.99, Revision 2 (Reference IV.1.C.iii-9) for the Braidwood Units 1 and 2 reactor vessel beltline materials at the EOL neutron fluence values corresponding to 32 EFPY. The fluence methodology follows the guidance and meets the requirements of Regulatory Guide 1.190 (Reference IV.1.C.iii-10). Furthermore, the reactor vessel inlet temperatures for Braidwood Units 1 and 2 remain within the accepted range identified in Regulatory Guide 1.99, Revision 2, Position 1.3. Therefore, the embrittlement correlations in the Regulatory Guide used to perform the ART calculations are applicable to the Braidwood Units 1 and 2 reactor vessels for the MUR power uprate.

The limiting ART values used in the development of the current P-T limit curves at 32 EFPY bound the MUR power uprate limiting ART values (at 32 EFPY) for both Units. Therefore, the current heatup and cooldown curves are valid through EOL (32 EFPY) with the MUR power uprate and do not require an update, because the limiting ART values from which the curves were developed remain applicable.

References

IV.1.C.iii-6 Pressure and Temperature Limits Report, "Braidwood Unit 1 Pressure and Temperature Limits Report (PTLR)," Revision 4.

IV.1.C.iii-7 Pressure and Temperature Limits Report, "Braidwood Unit 2 Pressure and Temperature Limits Report (PTLR)," Revision 4.

IV.1.C.iii-8 Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

IV.1.C.iii-9 NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

IV.1.C.iii-10 NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March 2001.

IV.1.C.iv Low-Temperature Overpressure Protection

None of the critical inputs for the low temperature overpressure protection system setpoints are changing for the MUR power uprate program, including the pressure-temperature limits described in Section IV.1.C.iii. The current low temperature overpressure protection setpoints are therefore bounding through EOL with the MUR power uprate and do not require update.

IV.1.C.v Effect on Upper-Shelf Energy Calculation

Byron Units 1 and 2

Upper-Shelf Energy (USE) was evaluated to ensure compliance with 10 CFR 50, Appendix G (Reference IV.1.C.v-1). If the limiting reactor vessel beltline material's Charpy USE is projected to fall below 50 ft-lbs, an equivalent margins assessment must be performed. The projected EOL Charpy USE decreases due to MUR power uprate fluence at the 1/4-T location were calculated per the Regulatory Guide 1.99, Revision 2 trend curves (Reference IV.1.C.v-2). The reactor vessel inlet temperatures for Byron Units 1 and 2 remain within the accepted range identified in Regulatory Guide 1.99, Revision 2, Position 1.3. Therefore, the embrittlement correlations in the Regulatory Guide used to perform the USE calculations are applicable to the Byron Units 1 and 2 reactor vessels for the MUR power uprate.

It was determined that all of the Byron Units 1 and 2 reactor vessel beltline materials will continue to remain above 50 ft-lbs. For Byron Unit 1, the limiting projected 1/4-T USE value is 65 ft-lbs, which corresponds to the Nozzle to Intermediate Shell Forging Circumferential Weld Seam. For Byron Unit 2, the limiting projected 1/4-T USE value is 68 ft-lbs, which also corresponds to the Nozzle to Intermediate Shell Forging Circumferential Weld Seam.

The Charpy USE decrease calculations performed at the end of the current operating license result in projected USE values that are consistent with those documented in the vessel integrity analyses of record. The 1/4-T USE values for the Byron Units 1 and 2 beltline materials meet the 50 ft-lb acceptance criterion of 10 CFR 50, Appendix G at the end of the current 40-year license period, including the MUR power uprate. The MUR power uprate has no impact on 10 CFR 50, Appendix G compliance.

References

IV.1.C.v-1 Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

IV.1.C.v-2 NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

Braidwood Units 1 and 2

Upper-Shelf Energy (USE) was evaluated to ensure compliance with 10 CFR 50, Appendix G (Reference IV.1.C.v-1). If the limiting reactor vessel beltline material's Charpy USE is projected to fall below 50 ft-lb, an equivalent margins assessment must be performed. The projected EOL Charpy USE decreases due to MUR power uprate fluence at the 1/4-T location were calculated per the Regulatory Guide 1.99, Revision 2 trend curves (Reference IV.1.C.v-2). The reactor vessel inlet temperatures for Braidwood Units 1 and 2 remain within the accepted range identified in Regulatory Guide 1.99, Revision 2, Position 1.3. Therefore, the embrittlement correlations in the Regulatory Guide used to perform the USE calculations are applicable to the Braidwood Units 1 and 2 reactor vessels for the MUR power uprate.

It was determined that all of the Braidwood Units 1 and 2 reactor vessel beltline materials will continue to remain above 50 ft-lbs. For Braidwood Unit 1, the limiting projected 1/4-T USE value is 75 ft-lbs, which corresponds to the Intermediate to Lower Shell Forging Circumferential Weld Seam (using surveillance data). For Braidwood Unit 2, the limiting projected 1/4-T USE value is 66 ft-lbs, which also corresponds to the Intermediate to Lower Shell Forging Circumferential Weld Seam (using surveillance data).

The Charpy USE decrease calculations performed at the end of the current operating license result in projected USE values that are consistent with those documented in the vessel integrity analyses of record. The 1/4-T USE values for the Braidwood Units 1 and 2 beltline materials meet the 50 ft-lb acceptance criterion of 10 CFR 50, Appendix G at the end of the current 40-year license period, including the MUR power uprate. The MUR power uprate has no impact on 10 CFR 50, Appendix G compliance.

References

IV.1.C.v-3 Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

IV.1.C.v-4 NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

IV.1.C.vi Surveillance Capsule Withdrawal Schedule

Byron Units 1 and 2

The reactor vessel material surveillance program provides a means for determining and monitoring the reactor vessel beltline material fracture toughness, to support analyses for ensuring the structural integrity of reactor vessel ferritic components.

A withdrawal schedule has been established to periodically remove surveillance capsules from each Byron Unit's reactor vessel, to monitor the reactor vessel materials condition under actual operating conditions. The schedules are consistent with ASTM E-185-82 (Reference IV.1.C.vi-3) and are based on the projected neutron fluence in the analyses of record. After a review of the withdrawal schedule contained in each Unit's Pressure and Temperature Limits Report (PTLR) (References IV.1.C.vi-1 and IV.1.C.vi-2), the surveillance capsule monitoring program requirements are satisfied through EOL, including the MUR power uprate fluence projections. The three required in-vessel surveillance capsules

have been withdrawn and tested to date from both Units and used in the PTS evaluation described in Section IV.1.C.i above. The other three capsules for both Units have also been withdrawn, but have not been tested, and are stored in the spent fuel pool. Since all of the surveillance capsules have been withdrawn from the Byron Units 1 and 2 reactor vessels, there is no longer a need to recommend withdrawal schedules. However, the current capsule withdrawal schedule shown in each Unit's PTLR will be updated to reflect the latest capsule fluence, lead factor, and withdrawal EFPY associated with each capsule. The surveillance capsule withdrawal schedules for Byron Units 1 and 2 are contained in Tables IV.1.C.vi-1 and IV.1.C.vi-2, respectively.

References

IV.1.C.vi-1 Pressure and Temperature Limits Report, "Byron Unit 1 Pressure and Temperature Limits Report (PTLR)," December 2006.

IV.1.C.vi-2 Pressure and Temperature Limits Report, "Byron Unit 2 Pressure and Temperature Limits Report (PTLR)," December 2006.

IV.1.C.vi-3 American Society for Testing and Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

Table IV.1.C.vi-1 Byron Unit 1 Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor[(b)]	Withdrawal EFPY[(a)]	Fluence[(b)] (n/cm^2, E > 1.0 MeV)
U	58.5°	4.05	1.18	0.409×10^{19}
X	238.5°	4.09	5.67	1.49×10^{19}
W	121.5°	4.08	9.27	2.26×10^{19}
Z[(c)]	301.5°	4.11	14.59[(c)]	3.34×10^{19}
V[(c)]	61.0°	3.89	14.59[(c)]	3.16×10^{19}
Y[(c)]	241.0°	3.85	18.81[(c)]	3.97×10^{19}

Notes:
(a) Effective Full Power Years (EFPY) from plant startup.
(b) Updated as part of the MUR uprate fluence evaluation.
(c) Standby Capsules Z, V, and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules.

Table IV.1.C.vi-2 Byron Unit 2 Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor[(b)]	Withdrawal EFPY[(a)]	Fluence[(b)] (n/cm^2, E > 1.0 MeV)
U	58.5°	4.02	1.19	0.406×10^{19}
W	121.5°	4.07	4.67	1.20×10^{19}
X	238.5°	4.14	8.63	2.18×10^{19}
Z[(c)]	301.5°	4.11	14.28[(c)]	3.25×10^{19}
V[(c)]	61.0°	3.88	14.28[(c)]	3.07×10^{19})
Y[(c)]	241.0°	3.88	20.05[(c)]	4.19×10^{19}

Notes:
(a) Effective Full Power Years (EFPY) from plant startup.
(b) Updated as part of the MUR uprate fluence evaluation.
(c) Standby Capsules Z, V, and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules.

Braidwood Units 1 and 2

The reactor vessel material surveillance program provides a means for determining and monitoring the reactor vessel beltline material fracture toughness, to support analyses for ensuring the structural integrity of reactor vessel ferritic components.

A withdrawal schedule has been established to periodically remove surveillance capsules from each of the Braidwood Unit's reactor vessels, to monitor the reactor vessel materials condition under actual operating conditions. The schedules are consistent with ASTM E-185-82 (Reference IV.1.C.vi -6) and are based on the projected neutron fluence in the analyses of record. After a review of the withdrawal schedule contained in each Unit's Pressure and Temperature Limits Report (PTLR) (References IV.1.C.vi-4 and IV.1.C.vi-5), the surveillance capsule monitoring program requirements are satisfied through EOL, including the MUR power uprate fluence projections. The three required in-vessel surveillance capsules have been withdrawn and tested to date from both Units and used in the PTS evaluation described in Section IV.1.C.i above. The other three capsules for both Units have also been withdrawn, but have not been tested, and are stored in the spent fuel pool. Since all of the surveillance capsules have been withdrawn from the Braidwood Units 1 and 2 reactor vessels, there is no longer a need to recommend withdrawal schedules. However, the current capsule withdrawal schedule shown in each Unit's PTLR will be updated to reflect the latest capsule fluence, lead factor, and withdrawal EFPY associated with each capsule. The surveillance capsule withdrawal summaries for Braidwood Units 1 and 2 are contained in Tables IV.1.C.vi-3 and IV.1.C.vi-4, respectively.

References

IV.1.C.vi-4 Pressure and Temperature Limits Report, "Braidwood Unit 1 Pressure and Temperature Limits Report (PTLR)," Revision 4.

IV.1.C.vi-5 Pressure and Temperature Limits Report, "Braidwood Unit 2 Pressure and Temperature Limits Report (PTLR)," Revision 4.

IV.1.C.vi-6 American Society for Testing and Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

Table IV.1.C.vi-3 Braidwood Unit 1 Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Lead Factor[(b)]	Withdrawal EFPY[(a)]	Fluence[(b)] (n/cm^2, E > 1.0 MeV)
U	58.5°	4.02	1.16	0.388×10^{19}
X	238.5°	4.06	4.30	1.17×10^{19}
W	121.5°	4.05	7.79	1.98×10^{19}
Z[(c)]	301.5°	4.09	12.01[(c)]	2.79×10^{19}
V[(c)]	61.0°	3.92	17.69[(c)]	3.71×10^{19}
Y[(c)]	241.0°	3.81	12.01[(c)]	2.60×10^{19}

Notes:
(a) Effective Full Power Years (EFPY) from plant startup.
(b) Updated as part of the MUR uprate fluence evaluation.
(c) Standby Capsules Z, V, and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules.

Table IV.1.C.vi-4 Braidwood Unit 2 Surveillance Capsule Withdrawal Summary

Capsule	Capsule Location	Lead Factor[(b)]	Withdrawal EFPY[(a)]	Fluence[(b)] (n/cm^2, E > 1.0 MeV)
U	58.5°	4.08	1.18	0.388×10^{19}
X	238.5°	4.03	4.24	1.15×10^{19}
W	121.5°	4.06	8.56	2.07×10^{19}
Z[(c)]	301.5°	4.14	12.78[(c)]	2.83×10^{19}
V[(c)]	61.0°	3.92	18.42[(c)]	3.73×10^{19}
Y[(c)]	241.0°	3.89	12.78[(c)]	2.66×10^{19}

Notes:
(a) Effective Full Power Years (EFPY) from plant startup.
(b) Updated as part of the MUR uprate fluence evaluation.
(c) Standby Capsules Z, V, and Y were removed and placed in the spent fuel pool. No testing or analysis has been performed on these capsules.

IV.1.D Codes of Record

Table IV.1.D-1 Codes of Record

Component	Code	Code Class	Edition and Addenda
Piping and Supports	ASME Section III	1	1974 Edition, and Addenda through Summer 1975
Steam Generator *(Unit 1)*			
Tube side	ASME Section III	1	1986 Edition with no Addenda Subsection NB and NF.
Shell side	ASME Section III	1	1986 Edition with no Addenda Subsection NB and NF.
Steam Generator[(1)] *(Unit 2)*			
Tube side	ASME Section III	1	1971 Edition plus Addenda through Summer 1972, and selected paragraphs of the Winter 1974 Addendum.
Shell side	ASME Section III	1[(2)]	1971 Edition plus Addenda through Summer 1972, and selected paragraphs of the Winter 1974 Addendum.
Reactor Vessel	ASME Section III	1	1971 Edition through Summer 1973 Addenda
Integrated Head Package CRDM Seismic Support Assembly	ASME Section III	NF	1977 Edition through Winter 1978 Addenda[(3)]
Reactor Coolant Pumps	ASME Section III	1	1971 Edition, and Addenda through Winter 1972
CRDM	ASME Section III	A	1974 Edition through Summer 1974 Addenda
Pressurizer	ASME Section III	1	1971 Edition through Summer 1973 Addendum
Loop Stop Valves			
Byron Units 1 and 2	ASME Section III	1	1971 Edition through Winter 1973
Braidwood Units 1 and 2	ASME Section III	1	1974 Edition through Winter 1975

Notes:

1. Code edition is for Class 1 Stress Reports. Code Edition applies only to the Byron Unit 2/Braidwood Unit 2 Model D-5 steam generators.
2. Code design requirements assigned are in excess of the requirement dictated by the applicable Safety Class.
3. The equipment is designed in accordance with Westinghouse Equipment Specification 955138, revision 2 Westinghouse Equipment Specification for Commonwealth Edison, Byron Units 1 and 2 and Braidwood Units 1 and 2 Nuclear Plants, Integrated Head Package, Control Rod Drive Mechanism Seismic Support Assembly.

IV.1.E Changes to Component Inspection and Testing Program

IV.1.E.i Inservice Testing Program

10 CFR 50.55a(f), Inservice Testing Requirements, mandates the development and implementation of an IST Program. Byron and Braidwood Stations have developed and implemented an IST Program for pumps and valves per the applicable requirements. Byron and Braidwood Technical Specification 5.5.8 describes the surveillance requirements that apply to the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

The applicable system analyses were reviewed to determine if the MUR power uprate would impact the existing IST Program. There are no significant changes to the maximum operating conditions and no changes to the design basis requirements that would affect component performance or test acceptance criteria. Therefore, the MUR power uprate has no impact on the testing required by the IST Program.

IV.1.E.ii In-Service Inspection Program

10 CFR 50.55a(g), Inservice Inspection Requirements, mandates the development and implementation of an ISI Program. The applicable program requirements are specified in ASME B&PV Code, Section XI. Byron and Braidwood Stations have developed and are implementing an ISI Program per these requirements. The ISI program is documented in the Station ISI Program plan. UFSAR Section 6.6 describes the ISI Program as it relates to Class 2 and 3 components. Class 1 components are discussed in the UFSAR within the various sections which describe the components. The MUR analyses were reviewed to determine if the MUR power uprate would impact the existing ISI Program. System classifications and boundaries, required procedures, and inspection frequencies for ASME Class 1, 2, and 3 systems are not affected.

IV.1.E.iii Erosion/Corrosion Program

Byron and Braidwood Stations have established and maintain a Flow Accelerated Corrosion (FAC) Program per NRC Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning. The FAC Program meets the intent of EPRI NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program," and INPO EPG-06, "Engineering Program Guide -Flow Accelerated Corrosion (FAC)." This program provides a standardized method of identifying, inspecting, and tracking components susceptible to FAC wear in both single and two-phase flow conditions. Program elements include: FAC susceptibility analysis and modeling, FAC inspection and evaluation, operational experience reviews, and crossover/crossunder main steam piping and moisture separators/reheaters inspections and evaluations. In general, plant systems are considered susceptible to FAC unless excluded by defined criteria. The criteria includes: material, moisture content, temperature, dissolved oxygen, frequency of system usage, plant-specific operating experience, and industry operating experience. Byron and Braidwood utilize the CHECWORKS Steam/Feedwater Application (SFA) FAC monitoring computer code to predict and track FAC susceptible components. The CHECWORKS SFA computer code has been used to create unit-specific databases. Once the database has been built, the application is used to perform analysis and data interpretation. These analytical models result in Wear Rate Analysis that rank components in order of predicted FAC wear and predicted time to reach minimum allowable wall thickness. The Byron and Braidwood Stations Unit 1 and 2 CHECKWORKS SFA models will be

updated to incorporate the changes associated with the power uprate. An evaluation was performed to identify piping that may be affected by MUR power uprate conditions and were deemed acceptable with proper FAC program inspection and monitoring. The following piping lines have been recommended for FAC review.

- Main Steam System header piping to Turbine Stop Valves
- Extraction Steam System supply piping from Low Pressure Turbines to the Low Pressure Heaters
- Extraction Steam System supply piping from the High Pressure Turbines to the #7 High Pressure Heaters
- Condensate System piping from the outlet of the Gland Steam Condenser to the Condensate Booster Pump suction header
- Condensate Booster System piping from the discharge of the Condensate Booster Pumps to the suction of the Main Feed Pumps
- Motor Driven and Turbine Driven Feed Pump discharge piping
- Steam Generator Blowdown System piping on the inlet header to the Blowdown Condensers

These components will be added as appropriate to the FAC program for future monitoring.

IV.1.F Impact of NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

NRC Bulletin 88-02 is discussed in Section IV.1.A.vi.1c for Unit 1 Steam Generators and in Section IV.1.A.vi.2.c for Unit 2 Steam Generators.

V ELECTRICAL EQUIPMENT DESIGN

1. *A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:*

   A emergency diesel generators

   B. station blackout equipment

   C. environmental qualification of electrical equipment

   D. grid stability

RESPONSE TO V ELECTRICAL EQUIPMENT DESIGN

V.1.A Emergency Diesel Generators

The onsite (emergency) AC power system for each unit consists of two diesel generators, one for each ESF division. The diesel generators provide an independent emergency source of power in the event of a complete loss of offsite power. The diesel generator supplies all of the electrical loads which are required for reactor safe shutdown either with or without a loss-of-coolant accident (LOCA).

The station electrical loads that change as a result of the power uprate are not fed from the emergency diesel generator (EDG) system. There are no increases to the emergency bus loads supported by the EDGs. The EDGs system equipment capacity and capability for plant operation at the uprate conditions are bounded by the EDG loading tables. The EDG loading tables are supported by the existing analysis of record. Both the bounding analysis and the EDG loading tables demonstrate that the EDG system has adequate capacity and capability to provide onsite standby power for safety-related loads following a loss of offsite power (LOOP) with or without a concurrent accident. Therefore, the EDG system is not affected by the MUR power uprate.

V.1.B Station Blackout Program

10 CFR 50.63 requires each light water cooled nuclear power plant to withstand and recover from a loss of all AC power, referred to as Station Blackout (SBO). Byron and Braidwood Stations coping duration is four hours. This is based on an evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability. The offsite power design characteristics include the expected frequency of a grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power. The evaluation was completed per NUMARC 87-00 and NRC Regulatory Guide 1.155.

The MUR power uprate has no impact on the current SBO coping duration of four hours. The MUR power uprate was evaluated for impact on the alternate AC power source and the following SBO coping issues: reactor coolant inventory, condensate storage tank inventory, Class 1E battery capacity, ventilation, compressed air, and containment isolation.

V.1.B.i Alternate AC Power Source

The Alternate AC Power Source consists of the excess capacity of the running EDG on the non-blacked out unit. The running EDG can be cross-tied to the bus of the same electrical division on the blacked out Unit from the Main Control Room within 10 minutes. This provides additional assurance that AC power will remain available. There are no increases to the emergency buses' loads supported by the EDGs as a result of the MUR power uprate. The total loading on the EDG for SBO will remain within the 2000-hour rating of the EDG. Therefore, the Alternate AC Power Source has sufficient capacity to operate systems necessary for coping with a SBO event for the required coping period.

V.1.B.ii Reactor Coolant Inventory

The non-blacked-out unit's available EDG provides power to one charging (CV) pump per unit. The CV pump will provide the water required for maintaining reactor inventory at an adequate level to ensure the core remains covered and natural circulation is not affected.

V.1.B.iii Condensate Storage Tank Inventory

The Condensate Storage Tank provides adequate inventory for decay heat removal following a SBO event at MUR power uprate conditions. The SBO analysis assumes an analytical value for core power of 3658.3 MWt (102% of 3586.6 MWt).

V.1.B.iv Class 1E Battery Capacity

The Byron and Braidwood Class 1E batteries have sufficient capacity to provide adequate power for safe shutdown loads. The MUR power uprate does not affect any DC powered indication, control, or protection equipment. Therefore, the Class 1E batteries are acceptable at MUR power uprate conditions.

V.1.B.v Ventilation

Evaluations have been performed for the following areas containing SBO equipment:

(1) Control Room and Auxiliary Electric Equipment Rooms;

(2) Component Cooling Water Pump and Motor-Driven Auxiliary Feedwater Pump Area;

(3) Diesel-Driven Auxiliary Feedwater Pump Area;

(4) Essential Service Water (SX) Pump Room;

(5) Residual Heat Removal Pump Room and Charging Pump Room;

(6) Diesel Generator Rooms;

(7) Battery Room; and

(8) Miscellaneous Electric Equipment and Switchgear Rooms.

There will be an available power source for the HVAC of the dominant areas (except for the main steam tunnel, which contains equipment qualified to the unventilated area temperature) and the heat load in those areas during SBO is not power level dependent. Therefore, the temperatures in the above areas are unaffected by the MUR power uprate.

V.1.B.v Compre ssed Air

No equipment that needs compressed air for operability has been identified for station blackout. Therefore, compressed air is not needed for station blackout.

V.1.B.vi Containment Isolation

The power uprate does not add or remove any containment isolation valves. The ability to close or operate containment isolation valves and position indication capability is not related to power level. The evaluation for containment isolation at current plant conditions remains applicable at MUR power uprate conditions.

V.1.C Environmental Qualification of Electrical Equipment

The Byron and Braidwood Environmental Qualification (EQ) Programs demonstrate that Class 1E electrical equipment will function, as required, under normal, abnormal, and/or accident environmental conditions. No such equipment will be added, removed, or modified as a result of the MUR power uprate. In addition, there is no change in the function of the equipment within the scope of the program. Finally, the MUR power uprate does not cause any zones to be modified and has no effect on the qualification process.

The evaluation of the environmental qualification of electrical equipment, therefore, considered the effects of MUR power uprate on the environmental parameters used in qualifying the Class 1E equipment. The environmental parameters of interest are: temperature, pressure, humidity, caustic spray, submergence, and radiation.

V.1.C.i Normal Operating Conditions

All the existing values of environmental parameters under normal operating conditions remain bounding for the MUR power uprate.

In containment, the MUR power uprate results in slight increases in full-power feedwater and reactor coolant hot leg temperatures, and slight decreases in full-power main steam and reactor coolant cold leg temperatures. The MUR power uprate causes no additional heat load to containment from the control rod drive system. In addition, the reactor vessel upper head follows the reactor coolant cold leg temperature, which is decreasing slightly at full power with the MUR power uprate. The cavity ventilation system was determined to be capable of maintaining the required temperatures at the uprate power level.

Therefore, the containment ventilation systems (i.e., the reactor containment fan coolers, the control rod drive ventilation system, and the reactor cavity ventilation system) are capable of maintaining current normal operating temperatures in containment following MUR power uprate implementation.

Likewise, it was determined that normal ambient temperatures in the auxiliary building and in the main steam pipe tunnels and safety valve enclosures would not be affected by MUR power uprate.

An evaluation of the normal radiation doses concluded that the conservatism in the current analyses was such that those analyses would remain bounding for the slight increase in normal radiation doses expected under MUR power uprate conditions. Therefore, the normal dose contribution to the total integrated doses used for determining equipment qualification parameters remains bounding for the MUR power uprate.

V.1.C.ii Abnormal Conditions

The abnormal condition of relevance for environmental qualification is a two-hour loss of ventilation to various auxiliary building areas following a high energy line break and accompanying loss of offsite power.

As discussed in the previous section, the normal environmental conditions in the auxiliary building, which represent the conditions that would be in effect at the time the high energy line break occurs, are not affected by the MUR power uprate. Additional evaluations determined that following a high energy line break, accompanying loss of offsite power, and subsequent room heatup, the peak room temperatures and pressure used for environmental qualification would not be significantly affected by the MUR power uprate. The evaluations included a consideration of the increased temperatures and pressures in certain high energy lines in the turbine building due to the uprate.

Therefore, with respect to the environmental qualification of equipment, the effects of a two-hour delay in restoring auxiliary building ventilation following a high energy line break are acceptable under MUR power uprate conditions.

V.1.C.iii Accident Conditions

The temperature and pressure values for the containment under accident environmental conditions were revised for the MUR power uprate conditions. An evaluation determined that all equipment in containment within the scope of the EQ program remains qualified, although, in one case, a slight reduction in qualified life (from 36.2 to 35.64 years) was required for the Byron Unit 2 GEMS containment level transmitters (these model level transmitters have not yet been installed on the other Units).

In general, high energy line breaks in the auxiliary building do not affect safe shutdown capability because safety equipment is compartmentalized to limit the consequences of a high energy line break to a single equipment train. As such, no equipment must be qualified for the harsh environments which result from high energy line breaks in auxiliary building compartments. The MUR power uprate has no effect on this situation. In those instances where the effects of a high energy line break in the auxiliary building may not be limited to a single train, the MUR power uprate has no effect on the operating conditions used

in determining the maximum values of temperature, pressure, or relative humidity. Therefore, the current temperature and pressure values for such auxiliary building general areas under abnormal or accident environmental conditions remain bounding.

The existing values of relative humidity, caustic spray, and submergence values under accident environmental conditions remain bounding for the MUR power uprate.

The evaluation of the radiological environmental parameters found that the total integrated doses used for determining equipment qualification parameters remain bounding for MUR PU, as discussed in Section II.5 of this report.

V.1.D Grid Stability

Byron Station

Two grid studies have been completed to support the proposed uprate. The studies were performed using a 1295 (1265) MWe output for Byron Unit 1(2) main generator. This value was chosen for the studies to bound the highest expected electrical output of the main generator under uprated conditions. Using this bounding value provides conservative results for the two studies performed.

PJM Interconnection (PJM), the grid operator, completed a system stability analysis to assess the impact of the uprate on the rotor angle stability of generating plants in the Commonwealth Edison (ComEd) and neighboring control areas. The analysis assumed a 1295 (1265) MWe for Byron Unit 1(2) main generator and a light load flow base case based on 2013 projections. The results of the analysis are as follows:

1. All of the primary-clearing scenarios were found to be stable.
2. All of the maintenance outage (prior outage) scenarios considered in this study were found to be stable.
3. All of the breaker failure scenarios considered in this study were found to be stable.

ComEd Transmission Planning completed an assessment of the capability of the grid to ensure adequate post-trip and LOCA voltage levels. The analysis assumed a 1295 (1265) MWe output for Byron Unit 1(2) main generator. The scenarios studied in these grid assessments are consistent with the transmission service provider requirements and include a single unit trip at the station under study, loss of the largest unit on the grid, loss of the most critical transmission circuit, and loss of load.

Power flow simulations were performed using 2012 transmission grid models for four system load conditions. The assessment concluded that with one exception, the lowest post-contingency voltage for Byron station is 349.1 kV, which remains above the minimum required switchyard voltage of 339.8 kV. The scenario that analyzes a unit trip with the other unit in shutdown and with a system load level equal to 75% of the 50/50 load forecast results in a post contingency voltage of 331.9 kV, which is lower than the minimum required voltage of 339.8 kV. This low post contingency voltage for this scenario is an existing (pre MUR) condition and is not related to the MUR uprate. PJM real-time state estimator continuously monitors and predicts grid voltages under various contingencies (e.g., unit trips). If the state estimator

predicts an inadequate voltage at Byron's switchyard, the station is notified and appropriate abnormal operating procedure is entered. Further details regarding this study are provided in Attachment 10b.

Braidwood Station

Two grid studies have been completed to support the proposed uprate. The studies were performed using a 1295 (1265) MWe output for Braidwood Unit 1(2) main generator. These values were chosen for the studies to bound the highest expected electrical output of the main generator under uprated conditions. Using these bounding values provides conservative results for the two studies performed.

PJM Interconnection (PJM), the grid operator, completed a system stability analysis to assess the impact of the uprate on the rotor angle stability of generating plants in the Commonwealth Edison (ComEd) and neighboring control areas. The analysis assumed a 1295 (1265) MWe for Braidwood Unit 1(2) main generator and a light load base case based on 2013 projections. The results of the analysis are as follows:

1. All of the scenarios considered for baseline instability were found to be stable.
2. All of the primary-clearing scenarios were found to be stable.
3. All of the prior outage scenarios considered in this study were found to be stable.
4. Of all breaker failure scenarios studied, three are unstable. The study provided remediation measures for these three scenarios involving adjustment of the critical clearing time. Exelon Generation Corporation will ensure that any modifications required by PJM are completed prior to uprate implementation. Further details regarding this study are provided in Attachment 10a

ComEd Transmission Planning completed an assessment of the capability of the grid to ensure adequate post-trip and LOCA voltage levels. The analysis assumed a 1295 (1265) MWe output for Braidwood Unit 1(2) main generator. The scenarios studied in these grid assessments are consistent with the transmission service provider requirements and include a single unit trip at the station under study, loss of the largest unit on the grid, loss of the most critical transmission circuit, and loss of load.

Power flow simulations were performed using 2012 transmission grid models for four system load conditions. The assessment concluded that the lowest post-contingency voltage is 349.5 kV, which remains above the minimum required switchyard voltage of 349.2 kV. Further details regarding this study are provided in Attachment 10a.

V.1.E Onsite Power Systems

The AC Distribution System is the source of power for the non safety-related buses and the safety related emergency buses. It consists of the 6.9kV, 4.16kV, 480V, and 120V systems (excluding the EDGs). The electrical changes resulting from the MUR power uprate occur in equipment primarily at the 6.9kV voltage level. The following loads were affected by the uprate: Condensate Pump/Condensate Booster Pump Motor, Heater Drain Pump Motor and Reactor Coolant Pump Motor. None of these revised brake horsepower values exceeded the motor nameplate rating, although the operating points changed. The Condensate Pump/Condensate Booster Pump (nameplate rating of 3500 hp) will increase by a maximum of 36 hp, the Heater Drain Pump Motor (nameplate rating of 2250 hp) will decrease by a minimum of 2

hp, and the Reactor Coolant Pump Motor (nameplate rating of 7000 hp) will increase by a maximum of 5 hp.

An evaluation also determined that current loading levels under MUR power uprate conditions have no impact on the 4.16 kV buses existing capability. There were no load increases on the 480V buses. The LEFM CheckPlus System is being installed as an MUR power uprate device however, no changes to the 120V design loading will occur. No changes as a result of MUR power uprate have been identified that would result in a change in the 120 V design load analysis calculations. Therefore, there is an insignificant change in the margin of the on-site electrical power systems. The 125Vdc system loads are not related to the power generation process and are therefore independent of the MUR power uprate. The 6.9kV, 4.16 kV, 480V, 120V and DC 125V electrical distribution systems are acceptable at power uprate conditions.

V.1.F Power Conversion Systems

V.1.F.i Main Generator

The nameplate rating is 1361 MVA (based on 75 psig hydrogen pressure), 0.90 power factor, and 25 kV. The generator is operated within the generator Capability Curve which provides corner points at 593 MVARs out and 424 MVARs in, and maintain generator load and hydrogen pressure within the limits of the Generator Capability Curve with a generator rating of 1361 MVA. The analyzed main generator output at the current NSSS power level of 3600.6 MWt is shown in Table V.1.F.i-1.

Table V.1.F.i-1 Current MWe Output Based on 3600.6 MWt NSSS Power Level

Unit	Summer (MWe)	Winter (MWe)
Braidwood 1	1239.0	1264.5
Braidwood 2	1213.7	1240.3
Byron 1	1225.7	1268.3
Byron 2	1203.0	1240.2

The analyzed main generator output based on the heat balance at MUR uprate conditions of 3672.3 MWt is shown in Table V.1.F.i -2.

Table V.1.F.i-2 Projected MWe Output Based on 3672.3 MWt NSSS Power Level

Unit	Summer (MWe)	Winter (MWe)
Braidwood 1	1264.2	1291.1
Braidwood 2	1238.0	1265.4
Byron 1	1250.4	1294.4
Byron 2	1227.8	1265.0

For the higher winter generator output, the associated analyzed MVAR and lagging power factor is shown in Table V.1.F.i -3.

Table V.1.F.i-3 MVAR Output and Power Factor Unit Winter (MWe) MVAR Power Factor

Unit	Winter (MWe)	MVAR	Power Factor
Braidwood 1	1291.1	430.6	0.95
Braidwood 2	1265.4	501.1	0.93
Byron 1	1294.4	420.5	0.95
Byron 2	1265.0	502.1	0.93

The exciter has the capability to support main generator operation within the capability curve for a leading power factor.

V.1.F.ii Isolated Phase Bus

The iso-phase bus duct is rated for 33,000 amperes. The rated generator output is 31,431 amperes at 1361 MVA and 25kV. Therefore, the increase from the MUR power uprate remains below the iso-phase bus maximum rated capability.

V.1.F.iii Main (Step-Up) Transformers

The main transformers increase the main generator 25 kV output voltage to the 345 kV transmission voltage. The transformers consist of 2-700 MVA transformers in parallel. The 1400 MVA capacity of the transformers is above the main generator 1361 MVA output capability. The transformers have sufficient capacity and design margin (approximately 1.3% based on conservative assumptions regarding load sharing difference between the two main transformers) to handle the electrical power requirements under the MUR power uprate conditions.

V.1.F.iv Unit Station Service (Auxiliary) Transformers

The Unit auxiliary transformers (UATs) are supplied by the 25 kV isolated phase bus and power the 6.9 kV and the 4.16 kV switchgear. Evaluation of the loading summaries has determined that the existing UATs have sufficient capacity with a minimum margin of approximately 32% at Braidwood and 25% at Byron to support operation at power uprated conditions without modification. The balance of plant (BOP) electrical loads affected by the uprate result in a small increase (< 0.5%) in the loading on the UATs. Even with the increased load, the UATs remain within their current rating with margin.

V.1.F.v Reserve Station Service (System Auxiliary) Transformers

The System Auxiliary Transformers (SATs) are supplied by the 345 kV switchyard. Evaluation of the connection loading has determined that the existing SATs have sufficient capacity with a minimum margin of approximately 32% at Braidwood and 25% at Byron to support operation at power uprated conditions without modification.

All of the auxiliary transformers have two secondary windings (6.9 and 4.16kV). The windings are considered to be independent of each other as long as they are within their rating. Due to the impedance values of the auxiliary transformer windings, the smaller the load on the 6.9kV winding, the greater the voltage drop across the transformer. Therefore to remain conservative with respect to the safety related 4.16kV voltage calculations, zero load is considered on the 6.9kV winding. Evaluation of the running voltage summaries also confirm that bus voltages are essentially unchanged at power uprate loading conditions. Accordingly, plant operation at power uprate conditions has no effect on loss of voltage or degraded grid voltage protection schemes, and motor starting scenarios.

In addition, evaluation of the short circuit duty confirms that short circuit values are essentially unchanged at power uprate loading conditions.

V.1.G Switchyard

The current to the switchyard is bounded by the generator capability. The transmission lead from the main power transformers to the switchyard is capable of carrying the full generator output. Therefore, the overhead lines are acceptable at the MUR conditions. An evaluation determined that the small increase in power output does not significantly impact the switchyard equipment. The switchyard system analyses bound the MUR power uprate conditions.

VI SYSTEM DESIGN

1. *A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems:*

   A. *NSSS interface systems for pressurized-water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling-water reactors (BWRs) (e.g., suppression pool cooling), as applicable*

   B. *containment systems*

   C. *safety-related cooling water systems*

   D. *spent fuel pool storage and cooling systems*

   E. *radioactive waste systems*

   F. *Engineered safety features (ESF) heating, ventilation, and air conditioning systems*

RESPONSE TO VI SYSTEM DESIGN

VI.1.A NSSS Interface Systems

VI.1.A.i Main Steam

The main steam system is described in UFSAR Section 10.3. This system was evaluated to determine the impact of the MUR power uprate and was found to be acceptable. System parameters are bounded by the original design equipment temperature and pressure ratings. Therefore, the main steam system is acceptable at power uprate conditions, with respect to temperature and pressure.

VI.1.A.i.a Main Steam Piping

See Section IV.1.A.v

VI.1.A.i.b Main Steam Safety Valves

A total of five Main Steam Safety Valves (MSSVs) are located on each main steam line outside reactor containment and upstream of the main steam isolation valves (MSIVs). MSSV lift setpoints are determined by steam generator design pressure and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The steam generator design pressure has not changed with the MUR power uprate, so the existing MSSV setpoints were evaluated and do not need to be changed. Capacities were evaluated and determined to be acceptable relative to the sizing criteria.

VI.1.A.i.c Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs provide a means to rapidly isolate a SG in the event of a downstream steam line rupture. The rapid closure of the MSIVs following the postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flowrate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since MUR power uprate does not impact these variables, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, MUR power uprate has no significant impact on the NSSS/BOP interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low power conditions where MUR power uprate has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, the MUR power uprate has no significant impact on the NSSS/BOP interface requirements for the MSIV bypass valves.

VI.1.A.i.d Moisture Separator Reheaters

The Moisture Separator Reheaters (MSR) shell and tube bundle are designed and manufactured per ASME Section-VIII Division 1. The MUR uprated operating conditions are within the accepted limits of the original design.

The total relief valve capacity of MSR safety relief valves is 12,267,000 lb/hr at 275 psia where as the maximum flow during the MUR uprate is 11,244,391 lb/hr or less; therefore, the safety relief valve capacities for the revised steam conditions are within the requirements of ASME Section-VIII Division 1.

The study of the MSR indicates that the flow rate changes are less than 2% of the previous operating condition; therefore, the impact due to the new MUR steam conditions on MSR performance would be negligible. Flow induced vibration calculations indicate there is no concern due to the MUR power uprate condition.

VI.1.A.ii Steam Dump

The steam dump function is accomplished by the SG PORVs (atmospheric relief valves) and the steam dump system (turbine bypass valves). The SG PORVs are described in UFSAR Section 10.3. The steam dump system is described in UFSAR Section 10.4.4.

VI.1.A.ii.a Steam Generator PORVs

There are four steam generator PORVs per unit, one on each main steam line. There is no change in function associated with the power uprate. The steam generator PORVs automatically modulate open and exhaust to the atmosphere whenever the steam line pressure exceeds a predetermined setpoint. This minimizes safety valve lifting during steam pressure transients. The steam generator PORV set pressure is between no-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these

pressures change for the proposed range of NSSS operating parameters, the steam generator PORV setpoint is unchanged.

The primary function of the PORVs is to provide a means for decay heat removal and plant cooldown when the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. The PORVs are set to automatically maintain the steam pressure below approximately 1175 psig under emergency shutdown or when the plant is being maintained on hot standby and the turbine bypass steam dump valves are unavailable. The PORVs are sized to have a capacity equal to approximately 10% of rated steam flow at no-load pressure. The steam generator PORVs in Byron and Braidwood Unit 1 meet this capacity, but the PORVs in Byron and Braidwood Unit 2 have a capacity of 9.3% at uprated conditions. An evaluation of the installed capacity concluded that the original design bases, in terms of plant cooldown capability, can still be achieved for the range of power uprate NSSS design parameters. Therefore, the steam generator PORVs are acceptable for operation at uprate conditions. Note that the Unit 1 PORV trim will be modified to address steam generator margin to overfill concerns as noted in Attachment 5a of this LAR. This modification will increase the PORV steam relief capacity.

VI.1.A.ii.b Steam Dump System

The steam dump system creates an artificial steam load by dumping steam to the main condenser. Each unit is provided with 12 condenser steam dump valves.

Steam dump in conjunction with the reactor control system permits the NSSS to withstand an external load reduction of up to 50% of plant rated electrical load without a reactor trip. The evaluation of the NSSS control systems margin to trip analysis confirms the steam dump system capability at MUR power uprate conditions. There is acceptable margin to the relevant reactor trip setpoints during and following the 50% load rejection transient for the MUR power uprate program.

VI.1.A.iii Extraction Steam

The extraction steam system heats the condensate and feedwater at various stages prior to the SGs, and provides the normal steam supply to the auxiliary steam system. Based on evaluation results, the extraction steam system operating parameters (pressure, temperature, flow, velocity) are not significantly impacted at MUR power uprate conditions. Therefore, the extraction steam system is acceptable at power uprate conditions.

VI.1.A.iv Condensate and Main Feedwater System

VI.1.A.iv.a Condensate System

There are four condensate and condensate booster pumps. Normally three of the four pumps are operating at full load, delivering water to the feedwater pumps suction header. Three heater drain pumps, with one in standby, deliver heater drain flow to the condensate booster system upstream of the 5th stage feedwater heater.

The power uprate results in increased condensate flow of approximately 1.9%. Adequate condensate and condensate booster pump net positive suction head is available at uprate conditions.

The condensate/condensate booster piping is discussed in Section IV.1.A.v. These lines are acceptable for operation at the MUR power uprate power level.

Relevant parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the condensate system is acceptable at power uprate conditions.

VI.1.A.iv.b Main Feedwater System

The main feedwater system employs two parallel 50% capacity turbine driven main feedwater pumps and one 50% capacity motor driven feedwater pump. Normal alignment is two turbine driven pumps in operation at full load conditions. The turbine driven feedwater pumps are variable speed, so the feedwater flow is controlled by the turbine driver speed and the feedwater regulating valves at the inlet to the steam generators.

The power uprate results in increased feedwater flow of approximately 1.9%. Adequate main feedwater pump net positive suction head is available at uprate conditions. Adjustment of the feedwater pump speed control program to accommodate the minor increase in flow due to the MUR will help maintain the feedwater control valves (FCVs) near their current full power stroke positions without significantly affecting system performance. The slight increase in extraction steam flow through the feedwater heaters results in a small increase in feedwater temperature entering the steam generators.

The ability of the feedwater isolation valves to isolate within 5 seconds of an isolation signal is unaffected by MUR power uprate conditions. The quick-closure requirements imposed on the FCVs and the bypass FCVs and the backup feedwater isolation valves causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam line break from no load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the MUR power uprate, the design loads and associated stresses resulting from rapid closure of these valves will not change. Operating conditions at MUR power uprate for the main feedwater valves remain bounded by the valve pressure and temperature ratings.

The feedwater piping is discussed in Section IV.1.A.v. These lines are acceptable for operation at the MUR power uprate power level.

Therefore, the main feedwater system is acceptable at power uprate conditions.

VI.1.A.iv.c Abnormal/Transient Operating Conditions

The following transients that impact feedwater flow were evaluated at power uprate conditions: loss of heater drain pump, loss of a main turbine driven feed pump, and a 105% feedwater flow increase transient.

The feedwater system was shown to support plant response to any of these postulated transients.

VI.1.A.v Feedwater Heaters

There are three parallel trains of 1st through 4th point low pressure feedwater heaters. There are two parallel trains of 5th and 6th point low pressure feedwater heaters and two parallel trains of 7th point high pressure feedwater heaters. All of the low pressure feedwater heaters (1st through 6th points) are located on the suction side of the main feedwater pumps. The 1st point feedwater heaters are located in the main condenser neck. The 7th point high pressure feedwater heaters are located on the discharge side of the main feedwater pumps.

Relevant feedwater heater parameter changes resulting from the power uprate do not exceed component design specifications. Feedwater heater shell side nozzles have been recommended for inclusion in the FAC program.

The feedwater heaters are acceptable at power uprate conditions.

VI.1.A.vi Feedwater Heater and Moisture Separator Reheater Vents and Drains

Feedwater heater and moisture separator reheater drains were evaluated at MUR power uprate conditions. Operating parameters (flow rate, pressure, and temperature) for MUR power uprate conditions were evaluated against design parameters. Evaluated components include system piping, level control valves, air-operated valve actuator sizing, heater drain pumps, drain tanks, and drain tank level control system scaling and setpoints.

Operating parameters (flow, pressure, temperature, and velocity) at uprate conditions do not significantly impact components and equipment design parameters. Evaluations concluded that all components were acceptable for MUR operating conditions.

Level control valve capacities were evaluated. The following valves were determined to have inadequate operating margins, particularly for plant transients:

Feedwater Heater Control Valves	Heater 2 Normal	Heater 2 Emergency	Heater 3 Emergency
Byron Unit 1:	HD026A/B	HD054A/B/C	HD051A/B/C
Byron Unit 2:	HD026B/C	HD054A/B/C	HD051A/B/C
Braidwood Units 1 and 2:	HD026A/B/C	No Change	No Change

All valves listed will be acceptable after the necessary valve trim upgrades or replacements are performed.

Air-operated valve actuator sizing was evaluated. The maximum expected differential pressures were evaluated against the maximum allowable differential pressures. All evaluated air-operated valves have maximum expected differential pressures which are acceptable with respect to the valve ratings to ensure proper actuation.

The heater drain pumps and drivers were evaluated and determined to be acceptable for MUR power uprate operation with respect to operating region, driver horsepower, and NPSH margin.

In addition to pressure and temperature evaluations, system drain tanks were evaluated for nozzle velocities and internal Q/A ratios (i.e., index of controllability). All parameters were determined to be acceptable for MUR power uprate.

The heater drain system level control setpoints and scaling were reviewed for MUR power uprate operation, and were determined to be acceptable.

Therefore, feedwater heater and moisture separator reheater drains piping (as discussed in Section IV.1.A.v) has been evaluated and is acceptable for MUR power uprate operating conditions.

VI.1.A.vii Auxiliary Feedwater System

The Auxiliary Feedwater (AFW) system design basis of record is described in UFSAR Section 10.4.9. The AFW system supplies feedwater to the secondary side of the steam generator when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The minimum flow requirements of the AFW system are dictated by safety analyses, and the results of the MUR power uprate safety analyses confirm that the current AFW system performance is acceptable for the MUR power uprate.

Each unit's system includes one motor driven pump and one diesel driven pump configured into two trains. Each pump takes suction through independent lines from the condensate storage tank (CST) or from the safety related service water system in case of emergency. The AFW system analyses are based on a core thermal power level of 3658.3 MWt, which is 102% of 3586.6 MWt. The analyzed core power level of 3658.3 MWt remains conservative and appropriately bounds the MUR power level.

The AFW system maximum operating pressure and temperatures remain essentially unchanged as a result of the MUR power uprate. Piping and component pressure and temperature design parameters bound power uprate operating pressure and temperature conditions.

AFW system flow requirements associated with the analyses are bounding for the power uprate. The AFW system has the capacity to provide adequate flow under transient and accident conditions. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design/performance or operation. Since no changes are being made to the pump design, the brake horse-power requirements are unaffected. No AFW system modifications are required to support the MUR power uprate. However, a modification to the AFW valves is being made as a result of the SG tube rupture accident analysis as described in LAR Attachment 5a. This modification adds air accumulators to provide a back-up air supply to the AFW flow control valves (AF-005).

The Byron and Braidwood Stations Units 1 and 2 licensing basis dictates that in the event of a LOOP, sufficient CST useable inventory must be available to maintain the “Reactor Coolant System in hot standby (MODE 3) at normal operating pressure and temperature for 2 hours, followed by a cooldown to the Residual Heat Removal (RHR) system entry conditions at 50ºF /hour, followed by a period not longer than one-hour to allow warm-up of the RHR pumps prior to placing the RHR System into service in

shutdown cooling mode". In light of these design bases requirements the Byron/Braidwood Technical Specifications ensures that CST contains a minimum useable inventory of 212,000 gallons. The minimum required useable inventory is based on reactor trip from 3658.3 MWt (i.e. 102% of current rated core power 3586.6 MWt). The power level is bounding for the power uprate. The Technical Specification minimum CST volume requirement of 212,000 gallons ensures that the usable volume bounds the minimum CST volume requirement. Therefore, the auxiliary feedwater system is acceptable at power uprate conditions.

VI.1.B Containmen t Systems

The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the environmental qualification acceptance limits.

VI.1.B.i Containment Spray System

The containment spray system is designed to reduce the pressure in the containment atmosphere at a rate which will ensure that the design leakage is not exceeded and to remove sufficient iodine from the containment atmosphere to limit the offsite and site boundary doses to values below those set by 10 CFR 50.67.

The containment response analyses are performed to power levels which bound the power uprate. The containment spray system operating and design parameters in the analyses bound the power uprate parameters. There are no new operating requirements imposed on the system as a result of power uprate. Therefore, the containment spray system is acceptable for operation at MUR power uprate conditions.

VI.1.B.ii Containmen t Ventilation

The containment ventilation systems are described in UFSAR Section 9.4.8. The containment ventilation system provides general area cooling and direct cooling to critical components. It also provides the means to purge containment atmosphere prior to personnel entry during maintenance periods. Containment ventilation consists of the following sub-systems:

- Reactor Containment Fan Cooler (RCFC),
- Containment charcoal filter units,
- Control rod drive mechanism ventilation, and
- Reactor cavity ventilation subsystem.

NSSS equipment heat load changes were analyzed at MUR power uprate conditions. The heat changes are insignificant and will not affect the containment bulk air temperature. Therefore, the MUR power uprate will have no significant impact on the containment atmosphere and the RCFC performance under both normal and accident conditions.

The operation of the containment charcoal filter units is not impacted by the MUR power uprate. No changes to flows or conditions are proposed which could affect the capacity of the system to remove airborne contaminants.

Containment ventilation subsystems, including reactor cavity ventilation and containment charcoal filter units as well as the containment purge system are not impacted by the MUR power uprate.

The CRDM equipment was analyzed at MUR power uprate conditions. The operating temperatures of the CRDM coils will remain below their design temperature under the Byron and Braidwood MUR program. There is no additional heat load to containment from the CRDMs as a result of the MUR power uprate.

The Reactor Coolant pump (RCP) has a motor oil cooler in addition to cooling from the RCFC's via containment atmosphere cooling. The RCFCs and CRDM cooling system provide air cooling that, in combination with the RCP motor oil cooler, maintain containment bulk air temperature within the Technical Specification limits.

VI.1.C Safety Related Cooling Water Systems

VI.1.C.i Component Cooling Water System

The Component Cooling Water (CC) System is described in UFSAR Section 9.2.2. The CC system is a closed loop piping system shared between Units 1 and 2, and rejects heat to the Essential Service Water (SX) system. Two CC pumps and one CC heat exchanger and one surge tank serve each unit.

One additional pump and heat exchanger are available as backup for either unit. Normally, two heat exchangers and two pumps (one per unit) are required to support the normal heat loads of both units. The CC system is designed to provide the cooling requirements for normal plant operation, plant shutdown, and following an accident.

The CC system was evaluated to confirm that the heat removal capabilities are sufficient to satisfy the MUR power uprate heat removal requirements during normal plant operation, plant shutdown, and following an accident. The analysis confirms that at MUR power uprate conditions, normal plant operation and required cooldown continue to be met.

VI.1.C.ii Essential Service Water System

The Essential Service Water (SX) system is described in UFSAR Section 9.2.1.2 and is divided into two redundant loops for each unit with an opposite unit crosstie. There are four SX pumps (two per unit). Each pump takes suction from the SX cooling tower (Byron) / pond (Braidwood). The SX system is designed to support a LOCA coincident with a LOOP in one unit and the concurrent orderly shutdown and cooldown from maximum power of the other unit to cold shutdown.

The normal and accident heat loads used in the design basis analyses for the SX system bound the targeted MUR power uprate power level with margin for calorimetric uncertainty. The evaluations determined that the existing SX flows will continue to support the heat removal requirements at power uprate conditions. The SX system and component design parameters remain bounding for power uprate

operation. No system modifications are required to support the power uprate. Therefore, the SX system is acceptable for operation at MUR power uprate conditions.

VI.1.C.iii Ultimate Heat Sink

Byron Station

The ultimate heat sink is described in UFSAR Section 9.2.5 and is common to both units. The Technical Specification required ultimate heat sink is the Essential Service Water (SX) Cooling Towers.

The SX system inlet temperature for normal, shutdown, and accident conditions is bounded for the power uprate. The ultimate heat sink is capable of cooling the SX system to prevent SX temperature from exceeding the inlet temperature limits during operating conditions. No system modifications are required to support the power uprate. The analyses of record assessing the UHS capability remain bounding for MUR PU. Therefore, the ultimate heat sink is acceptable for operation at power uprate conditions.

Braidwood Station

The ultimate heat sink is described in UFSAR Section 9.2.5 and is common to both units. The Technical Specification required ultimate heat sink is the Essential Service Cooling Pond (ESCP).

The SX system inlet temperature for normal, shutdown, and accident conditions is bounded for the power uprate. The ultimate heat sink is capable of cooling the SX system to prevent SX temperature from exceeding the inlet temperature limits during operating conditions. No system modifications are required to support the power uprate. The analyses of record assessing the UHS capability remain bounding for MUR PU. Therefore, the ultimate heat sink is acceptable for operation at power uprate conditions.

VI.1.C.iv Residual Heat Removal System/Shutdown Cooling

The plant cooldown performance licensing basis is documented in Table 5.4-7, and Figures 5.4-6 and 5.4-7 of the UFSAR. Generally, cooldown times increase due to the higher MUR power uprate decay heat load. For the normal (2-train) cooldown cases, the time increases from 39.9 hours (current UFSAR) to 42.3 hours after shutdown (No Spent Fuel Pool heat load), and from 43.6 hours (current UFSAR) to 46.7 hours (Minimum Spent Fuel Pool heat load), to cool from 350°F to 140°F. For the single-train case, the time increases from 47.6 hours to 50.3 hours after shutdown (No Spent Fuel Pool heat load), to cool from 350°F to 200°F. The single-train acceptance criterion for single-train cooldown of 72 hours (reference UFSAR section 5.4.7.2.7) continues to be met at MUR power uprate conditions.

Section II.2.18, "Safe Shutdown Fire Analysis," discusses the residual heat removal system cooldown requirements for the Safe Shutdown Fire Analysis.

VI.1.D Spent Fuel Pool Storage and Cooling Water

VI.1.D.i Spent Fuel Pool Criticality

The spent fuel pool provides for storage of various Westinghouse Optimized Fuel Assembly (OFA) types of different initial fuel enrichments and exposure histories in two distinct regions. (For this discussion,

the term OFA is intended to refer to the specific reduced fuel rodlet diameter, and includes all analyzed fuel types with this diameter, such as Vantage 5.) The Region 1 racks may contain an initial nominal enrichment of ≤ 5.0 weight percent U-235. Region 2 racks are analyzed for storing fuel assemblies which may contain an initial nominal enrichment of ≤ 5.0 weight percent U-235 with credit for burn-up.

The objective of the criticality calculation is to show that the effective neutron multiplication factor (k_{eff}) is ≤ 0.95 with the racks fully loaded with the highest anticipated reactivity. Of the 6 assumptions identified in the SFP criticality calculation, only the SFP temperature which impacts the moderator reactivity coefficient and the depletion of the fuel during core operation might be impacted by MUR power uprate conditions. An evaluation was conducted to verify that these assumptions remain valid.

The temperature of the spent fuel pool may be affected by the MUR power uprate The criticality analysis uses a pool temperature of 4°C (39.2°F) and a negative reactivity coefficient. Using the temperature of maximum possible water density (i.e., 4°C); therefore, assures that the true SFP reactivity will always be lower than the calculated value regardless of temperature. Therefore, changes to the actual spent fuel pool temperature as a result of the MUR power uprate will have no impact the spent fuel pool criticality analysis.

Since the criticality analysis for the Region 1 spent fuel pool fuel racks assumes unburned fuel assemblies, the MUR power uprate will have no impact on the criticality analysis as long as the maximum nominal enrichment of post-MUR power uprate fuel assemblies is ≤5.0 weight % U-235 as required by Technical Specifications 3.7.16, “Spent Fuel Assembly Storage” and 4.3.1, “Criticality.”.

Reactor core operating conditions as a result of the MUR power uprate; e.g., highest fuel and moderator temperature and the soluble boron concentrations, may impact the criticality calculation with regards to the spent fuel stored in Region 2 of the SFP. If the post-MUR power uprate fuel assemblies have an initial fuel enrichment of ≤ 5.0 weight % U-235 and a fuel burn-up > 40,000 MWD/MTU as required by Technical Specification Figure 3.7.16-1, then the post-MUR power uprate fuel assemblies will be allowed to be stored in Region 2 of the spent fuel pool. Therefore, MUR power uprate has no impact on the criticality analysis for the Region 2 of the spent fuel pool.

Additionally, Areva Lead Use Assemblies (LUAs) have been used at Braidwood Station. The criticality analysis for the Braidwood Station Region 1 spent fuel pool fuel racks is for unburned fuel assemblies. Since the reactivity of the unburned LUA fuel is less than the unburned OFA fuel, the MUR power uprate will have no impact on the criticality analysis as long as the maximum nominal enrichment of post-MUR power uprate fuel assemblies (LUA or OFA) ≤ 5.0 weight % U-235. Region 2 of the spent fuel pool can accommodate the storage of fuel assemblies (LUA or OFA) in any cell for fuel assemblies with a nominal initial enrichment of ≤ 5.0 weight % U-235 with minimum discharge burnups. Therefore, the MUR power uprate has no impact on the criticality analysis.

Based on the evaluation, the current criticality calculation remains valid and will not be impacted by the MUR power uprate.

VI.1.D.ii Spent Fuel Pool Cooling System

The Spent Fuel Pool Cooling (FC) system was evaluated for MUR power uprate conditions. The FC system is designed to remove the amount of decay heat that is produced by the number of spent fuel assemblies that are stored in the pool following a refueling and the accumulated assemblies resulting from previous refuelings that are in the pool.

The FC system design basis addresses three scenarios. Each scenario assumes a 100 hour in-core decay time and no heat loss to the environment or Spent Fuel Pool structures. The three scenarios include:

1) a normal 1/3 core refueling discharge assuming one train of FC cooling operating;
2) a full core discharge assuming one train of FC cooling operating;
3) a normal 1/3 core refueling discharge followed 17 days later by a full core offload from the opposite unit assuming both FC trains operating.

Under MUR power uprate the decay heat load in the spent fuel pool increases slightly, resulting in an increase of approximately 3.5°F in the expected peak spent fuel pool water temperature for each of the three scenarios. The peak spent fuel pool water temperature, for each scenario, remains well below the FC system design temperature of 200 °F.

The FC system capacity for make-up to the spent fuel pool bounds the required make-up to the pool under MUR conditions for all three refueling scenarios, with significant margin. There are no required modifications to the FC system and all existing components, including associated pressures and flow rates, have been evaluated as acceptable for operation at MUR power uprate conditions.

VI.1.E Radioactive Waste Systems

VI.1.E.i Gaseous Waste

The gaseous waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units at Braidwood and both units at Byron and is sized to treat the radioactive gases released during simultaneous operation of both units. Gaseous waste system functions are unaffected by the MUR power uprate and there is an insignificant impact on the gaseous waste volume. No system or component design parameters were exceeded at uprate conditions. The gaseous waste system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.ii Liquid Waste

The liquid waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units at Braidwood and both units at Byron and is sized to treat the radioactive liquid waste produced during simultaneous operation of both units. Liquid waste system functions are unaffected by the MUR power uprate and there is an insignificant impact on the liquid waste volume. No system or component design parameters were exceeded at uprate conditions. The liquid

waste system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.iii Solid Waste

The solid waste system and its various subsystems and components were evaluated for the power uprate. The system is common to both units at Braidwood and both units at Byron and is sized to treat the radioactive solid waste produced during simultaneous operation of both units. Solid waste system functions are unaffected by the MUR power uprate and there is an insignificant impact on the solid waste volume. No system or component design parameters were exceeded at uprate conditions. The solid waste system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

VI.1.E.iv Steam Generator Blowdown System

The Steam Generator Blowdown System (SGBS) controls the chemical composition of the steam generator secondary-side water within the specified limits. The SGBS also controls the buildup of solids in the steam generator secondary. The blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary to secondary leakage. Since these variables are not impacted by the MUR power uprate, the blowdown required to control secondary chemistry and steam generator solids will not be impacted by the MUR power uprate. The SG blowdown system will continue to be operated per the plant chemistry program following the MUR PU with no significant changes in blowdown flow rate. Therefore, the MUR PU will not challenge the design flowrate of 360 gpm for the SGBS system.

Blowdown system operating temperatures and pressures will decrease and remain bounded by the existing parameters under uprate conditions.

The uprate will not significantly increase the potential for flow accelerated corrosion on the blowdown system piping and components, as the blowdown flowrate is not significantly impacted. Therefore, the SG blowdown system will continue to meet system design requirements at MUR power uprate conditions.

VI.1.F Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning Systems

VI.1.F.i Control Room Ventilation System

Byron and Braidwood UFSAR Section 9.4.1 describes the main control room heating, cooling and ventilation (HVAC) system. The control room HVAC system is common to both Units 1 and 2 and serves the main control room (Units 1 and 2), auxiliary electric equipment rooms, upper cable spreading rooms, HVAC equipment room, security control center, Shift Manager's office/records room and miscellaneous locker room, toilets, kitchen (Braidwood only), and storage rooms. The control room HVAC system is comprised of two full capacity, redundant equipment trains, each located in separate HVAC equipment rooms. The control room HVAC is designed to provide a controlled temperature of 75°F ± 2°F and a relative humidity of 20% to 60% in the control room, auxiliary electric equipment rooms, kitchen

(Braidwood only), record room, storage room, and security control center. The control room HVAC system maintains the control room environment for personnel comfort and ensures that a temperature of 90°F is not exceeded for equipment concerns. The upper cable spreading room ambient conditions are expected to fluctuate between 65°F and 90°F, 20% and 70% relative humidity depending on outside temperatures.

The control room and auxiliary electric equipment rooms do not contain piping that is expected to see an increase in fluid temperature as a result of MUR power uprate implementation. In addition, the electrical equipment load demand and transmission loads are also not expected to be increased as a result of MUR power uprate implementation. As such, the area heat loads will not be impacted by MUR power uprate.

VI.1.F.ii ESF Ventilation System

Byron and Braidwood UFSAR Section 9.4.5 describes the Engineered Safety Features (ESF) Ventilation system. The ESF ventilation system is comprised of the Auxiliary Building HVAC system, Diesel-Generator Facilities Ventilation System, Miscellaneous Electric Equipment Room Ventilation System, and the ESF Switchgear Ventilation System.

The auxiliary building HVAC system serves all plant areas of the auxiliary building including the engineered safety features cubicles and the fuel handling building, but excludes the solid radwaste facilities control room, computer rooms, auxiliary electric equipment rooms, the control room, miscellaneous offices, and laboratories within auxiliary building, which are served by separate independent HVAC systems.

Each of the four diesel-generator rooms and day tank rooms is provided with an independent ventilation system which provides: (1) continuous ventilation for the day tank room during normal plant operation, (2) ventilation for the diesel generator when it operates, and (3) a source of combustion air for the diesel-generator.

Each of the four diesel oil storage rooms is provided with an independent ventilation system which provides continuous ventilation of each diesel oil storage room. The power generation design basis for the diesel oil storage rooms is to prevent the accumulation of oil fumes.

The ESF portion of the miscellaneous electric equipment room ventilation system serves the miscellaneous electric equipment and battery rooms for Units 1 and 2. Each Unit 1and 2 room is provided with an independent ventilation system. Supplemental, non-ESF cooling is provided to the inverters (Byron only) and rod drive cabinets.

The ESF switchgear ventilation system serves the ESF switchgear rooms. The system removes equipment heat to maintain the switchgear room temperatures in accordance with equipment requirements. Independent switchgear ventilation systems are provided for each of Units 1 and 2 ESF switchgear Divisions (11, 12, 21, and 22).

The auxiliary building heat load under normal operation will not increase in most areas under MUR power uprate conditions. For those areas with no increase in heat load, there are no adverse operational or equipment affects. Heat loads in a limited number of areas did increase under MUR power uprate

conditions. The heat load increase in these areas was minimal and was evaluated to be acceptable. It is noted that the ESF cubicle coolers only operate during operation of the corresponding pump. These unit coolers are actively cooled by Essential Service Water (ESW) during accident conditions. It is noted that the sump temperature under MUR power uprate conditions will not exceed the value used in the existing analyses. Therefore, the auxiliary building HVAC system is acceptable for the MUR power uprate.

The diesel-generator room, miscellaneous electric equipment room, and switchgear room do not contain piping that is expected to see an increase in fluid temperature as a result of MUR power uprate implementation. In addition, the electrical equipment load demand and transmission loads are also not expected to increase as a result of MUR power uprate implementation. As such, the area heat loads in these rooms will not be impacted by MUR power uprate.

VI.1.F.iii Fuel Handling Area Ventilation System

The fuel handling area is served by the Auxiliary Building Ventilation system, is described under Section VI.1.F.ii – ESF Ventilation System.

VII OTHER

1. *A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.*

2. *A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:*

   A. *emergency and abnormal operating procedures*

   B. *control room controls, displays (including the safety parameter display system) and alarms*

   C. *the control room plant reference simulator*

   D. *the operator training program*

3. *A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate.*

4. *A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2% to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.*

5. *A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:*

   A. *A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.*

   B. *A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.*

RESPONSE TO VII OTHER

VII.1 Operator Actions

Operator actions included in the safety analyses were reviewed for potential MUR power uprate impact. The following design basis events were reviewed:

▪ Appendix R	Fire Protection Report
▪ Boron Dilution	UFSAR Section 15.4.6
▪ Small Break LOCA	UFSAR Section 15.6.5.2.2
▪ Radioactive Release from a Subsystem or Component	UFSAR Section 15.7
▪ Large Break LOCA	UFSAR Section 15.6.5.2.1
▪ Main Steamline Break	UFSAR Sections 15.1.5 and 15.1.6
▪ Main Feedwater Line Break	UFSAR Section 15.2.8
▪ Steam Generator Tube Rupture	UFSAR Section 15.6.3
▪ Fuel Handling Accident	UFSAR Section 15.7.4

The safety analysis reviews have determined that the existing required operator actions are not affected by the MUR power uprate. There is no reduction in time for required operator actions. No new manual operator actions were created and no existing manual actions were automated. Note that required operator actions were modified by the re-analysis of the Steam Generator Tube Rupture Margin to Overfill event (Reference 5a); however, the MUR had no impact the required operator actions.

The power uprate is being implemented under the administrative controls of the design change process. Other potential impacts on operator actions and action times in plant procedures may be identified and evaluated during the design change impacts review. The design change process ensures that impacted procedures will be revised prior to the power uprate implementation.

VII.2 Modifications That Change Operator Actions

VII.2.A Emergency and Abnormal Operating Procedures

Emergency and abnormal operating procedures were reviewed to determine any MUR power uprate impact. No changes are required to operator mitigation actions as a result of the MUR power uprate with the exception of the operator response times noted for mitigation of the Steam Generator Tube Rupture as discussed in Attachment 5a. The review identified a subset of emergency operating procedure (EOP) setpoints that require revision. These EOP setpoints and associated operator procedures will be revised to reflect a total core power that bounds the MUR power uprate in conformance with the Westinghouse EOP Setpoint Methodology. The procedure changes and any associated operator training will be completed during the power uprate implementation and prior to operation above 3586.6 MWt.

VII.2.B Control Room Controls, Displays and Alarms

The MUR power uprate is being implemented under the plant modification process administrative controls. The MUR power uprate modification will implement the changes that are required to certain non-safety related systems, including Control Room displays and alarms. Various Balance of Plant (BOP) instrument rescaling, setpoint and alarm point changes in the plant will be made, but will not result in any control or instrumentation changes in the Control Room. These changes will be made in accordance with

the requirements of 10 CFR 50.59, "Changes, tests, and experiments," and will be implemented prior to operating above the current licensed thermal power of 3586.6 MWt.

If any one of the four LEFM instruments becomes inoperable the Control Room will receive an annunciator alarm and a Plant Process Computer (PPC) alarm. A control room annunciator response procedure will be developed providing guidance to the operators for initial alarm diagnosis and response. Control Room Operators will conservatively respond to a LEFM single path or single plane failure in the same manner as a complete system failure. The Byron and Braidwood Station Technical Requirement Manuals (TRM) will be revised as discussed in Attachment 1 to the LAR to address contingencies for inoperable LEFM instrumentation.

As described in Section I.1.E, the Braidwood and Byron calorimetric application on the PPC will execute three simultaneous calculations of reactor power. The results of the calculations will be made available to the Control Room Operators on the PPC.

Sections VII.3 "Intent to Complete Modifications," I.1.D "Disposition of NRC SER Criteria During Installation" and LAR Attachment 1 Section 3.4.8 "Operator Training, Human Factors, and Procedures" provide additional information.

VII.2.C Control Room Plant Reference Simulator

The MUR power uprate is being implemented under the plant modification process administrative controls. As part of this process, simulator modifications will be implemented. Simulator required changes resulting from the MUR power uprate will be evaluated, implemented and tested per Byron and Braidwood Station approved procedures. Simulator fidelity will be revalidated per Byron and Braidwood Station approved procedures. Necessary simulator modifications will be completed in time to support operator training.

VII.2.D Operator Training Program

The operator training program requires revision as a result of the MUR power uprate. Operator training will be developed and the operations staff will be trained on the plant modifications, Technical Specification and TRM changes, and procedure changes will be implemented per controlled plant procedures prior to operating above the current licensed thermal power of 3586.6 MWt.

VII.3 Intent to Complete Modifications

The MUR power uprate is being implemented under the plant modification process administrative controls. As discussed in Attachment 1, Section 3.4.5, "Plant Modifications," changes to certain non-safety related systems, including minor equipment changes, replacements, and setpoint / alarm changes necessary to support the MUR power uprate will be implemented. These changes will be made in accordance with the requirements of 10 CFR 50.59, "Changes, tests, and experiments," and will be implemented prior to operating above the current licensed thermal power of 3586.6 MWt.

VII.4 Temporary Operation Above Licensed Power Level

Consistent with the Nuclear Energy Institutes (NEI) Position Statement (Reference VII.4-1) as endorsed by the NRC (References VII.4-2 and VII.4-3), Exelon's procedures prohibit temporary operation above full steady-state licensed power level. Byron and Braidwood Stations General Operating Procedures (BGP100-3 and BwGP 100-3, respectively) proactively prevent operation above full steady-state licensed power levels during planned evolutions and direct operators to promptly reduce power levels in the event that full steady-state licensed power level is exceeded during unplanned events and transients. Guidance provided is to monitor and take conservative actions to maintain the 10-minute calorimetric power level below 100% such that the 1-hour calorimetric will not exceed 100%. No procedure revisions are required to prevent operation above the licensed power level.

References

VII.4-1 NEI Position Statement for Guidance to Licensees on Complying with the Licensed Power Limit (ADAMS ML081750537)

VII.4-2 NRC “Safety Evaluation Regarding Endorsement of the NEI Guidance for Adhering to the Licensed Thermal Power Limit” on October 8, 2008 (ADAMS ML082690105)

VII.4-2 NRC Regulatory Issue Summary 2007-21, Rev. 1, “Adherence to Licensed Power Limits,” dated February 9, 2009 (ML090220365)

VII.5 Environmental Analysis (10 CFR 51.22)

A discussion on the environmental analysis is presented in LAR Attachment 1, Section 5.0, “Environmental Consideration.” As noted in Section 5.0, 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusions or otherwise not requiring environmental review," addresses requirements for submitting environmental assessments as part of licensing actions. 10 CFR 51.22, paragraph (c)(9) states that a categorical exclusion applies for Part 50 license amendments that meet the following criteria:

i. No significant hazards consideration (as defined in 10 CFR 50.92(c)).

ii. No significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

iii. No significant increase in individual or cumulative occupational radiation exposure.

As discussed in LAR Attachment 1, Section 5.0, “No Significant Hazards Consideration,” the proposed changes in this amendment request do not involve a significant hazards consideration.

VII.5.A Effluents

There is no significant change in the types or significant increase in the amounts of gaseous, liquid or solid effluents. Evaluations of the effects of the proposed changes related to the increase in reactor power

on effluent sources concluded that, at most, the increase in radiological effluents is proportional to or slightly greater than the requested power increase. The radiological effluent calculations in the revised SGTR dose analysis show more than a minimal increase in the accident dose, as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000. This "more than minimal increase" is not considered a significant increase as the revised SGTR accident dose values remain within the limits specified in the Standard Review Plan (SRP), Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)."

Non-radiological effluent releases are either unaffected (i.e., not power dependent) or insignificantly affected (i.e., increase by approximately 2% or less) by the proposed changes and continue to be bounded by those described in the Final Environmental Statement for Byron Station, Units 1 and 2; and Braidwood Station, Units 1 and 2.

VII.5.B Occupational Radiation Exposure

There is no significant increase in individual or cumulative occupational radiation exposure. Evaluations of projected radiation exposure due to liquid, gaseous and solid radwaste concluded that normal operation radiation levels increase slightly, (approximately 2.0%) for the proposed uprate. The occupational exposure is controlled by the plant radiation protection program and is maintained within values required by regulations.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

VII.6 Programs and Generic Issues

VII.6.A Fire Protection Program

The Fire Protection Program is described in the Fire Protection Report. The program is in conformance with the requirements of Branch Technical Position CMEB 9.5-1, as described in the report.

The installation of the LEFM CheckPlus flow meter and associated cables will not have any significant effect on the Fire Hazards Analysis. An analysis of the change in combustible loading determined that the overall increase in fire loading is small and does not change the fire load classification of each affected fire zone. The increase in combustible loading in any affected zone is a fraction of a percent. Furthermore, the existing fire barriers, fire detection, and fire suppression equipment are adequate for the fire hazards.

The MUR power uprate will not require any new operator actions, and the procedures and resources necessary for systems required to achieve and maintain safe shutdown will not change and are adequate for the MUR power uprate.

A review of the impact of the MUR power uprate on the plant ventilation systems determined that any effects from additional heat in the plant environment due to the increased power will not prevent required

post fire operator manual actions, as identified in the fire protection program, from being performed at and within their designated time.

VII.6.A.i Fire Protection Systems

The Fire Protection Systems are not impacted by the MUR power uprate.

In addition to its fire protection functions, the Fire Protection System can also be utilized to provide a supply of water to the spent fuel pool. The UFSAR Section 9.1.3.3 states:

> "The results of the unlikely event of a failure of the return line to the spent fuel pool downstream of the two spent fuel pool heat exchangers would be a rise in pool water temperature followed by an increase in evaporative losses. These losses could be made up indefinitely from the Safety Category I Fire Protection System."

With MUR power uprate, the heat load on the spent fuel pool is expected to increase slightly. However, because the maximum evaporation rate from the spent fuel pool under current conditions (75,340 lb/hr) is much less than the capacity of the Fire Protection System, and because this spent fuel pool makeup function would not be required when the Fire Protection System is called upon for fire protection functions, the supply of water from the Fire Protection System for this makeup function will continue to be adequate.

The Fire Protection System can also be utilized to provide cooling water to the centrifugal charging pumps in the unlikely event that essential service water is not available. UFSAR Table 9.2-11, Note 6 states:

> "An alternate cooling source is available to the centrifugal charging pumps by use of temporary hoses from the Fire Protection System (not credited in any design basis accident)."

The MUR power uprate does not affect the centrifugal charging pump flow rate or fluid temperature. Thus, no effect on the capability of the Fire Protection System to provide cooling water to the centrifugal charging pumps is expected.

VII.6.A.ii Responsibilities

Plant management, supervisory, and station personnel responsibilities in support of the Fire Protection Program are not impacted by the MUR power uprate.

VII.6.A.iii Administrative Controls

The administrative controls outlined in the Fire Protection Report were reviewed. MUR power uprate does not affect the established administrative controls.

VII.6.A.iv Fire Brigade

There are no changes in the fire brigade structure, responsibilities, reporting relationships, equipment, or qualifications resulting from the MUR power uprate.

VII.6.A.v Evaluation of Inadvertent Operation of Fire Protection Systems

The MUR power uprate does not affect the existing evaluation conclusions for the inadvertent or spurious operation of systems credited in the Fire Protection Report.

VII.6.B High Energy Line Break Program

The high and moderate energy break program ensures that systems or components required for safe shutdown or important to safety are not susceptible to the consequences of high and/or moderate energy pipe breaks. The effects of high energy line breaks inside containment have been assessed in Section 3.6 of UFSAR. The effects of high energy line breaks in the turbine building have been evaluated with respect to potential impact on safety-related equipment located in adjoining auxiliary building rooms. The results of this evaluation are described in Section 3.11 of UFSAR. The description of the design approach is detailed in Section 3.6.1.2 of UFSAR. High-energy pipe breaks are analyzed for piping for which the maximum operating pressure exceeds 275 psig and the maximum operating temperature equals or exceeds 200°F. High-energy pipe cracks are postulated in piping for which either the operating pressure exceeds 275 psig or the operating temperature equals or exceeds 200°F.

The evaluation concluded that the MUR power uprate does not result in any new or revised high or moderate energy line break locations. The high and moderate energy line break analysis is not affected. Area temperature and pressure resulting from high energy line breaks and internal flooding conditions resulting from moderate energy line breaks remain valid at power uprate conditions.

VII.6.C Appendix J Testing Program

UFSAR Section 6.2.6, “Containment Leakage Testing,” states that the containment leakage testing program includes Type A tests to measure the containment overall integrated leakage rate, Type B tests to detect and measure local leakage at containment penetrations, and Type C tests to measure containment isolation valve leakage rates. The containment leakage tests are performed as required by 10 CFR 50, Appendix J, Option B.

The LOCA containment response was reanalyzed for the MUR power uprate and it was confirmed that the peak calculated containment internal pressure (P_a) specified in Technical Specification 5.5.16 remains bounding for the MUR power uprate. No changes or modifications are required to the existing Appendix J Program or procedures. Therefore, Technical Specification 5.5.16 and the applicable Appendix J Program procedures are acceptable at MUR PU conditions.

VII.6.D Coatings Program

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and radionuclide contamination. Coatings also provide wear protection during plant operation

and maintenance activities. These coatings are subject to 10CFR 50, Appendix B quality assurance requirements, because their degradation could adversely impact safety related equipment. The Service Level I coatings in Byron and Braidwood Unit 1 and 2 are currently qualified to withstand a LOCA environment.

The current Service Level I coatings bound the maximum accident primary containment conditions during a DBA LOCA under MUR power uprate. Coating acceptability was based on the acceptance criteria of ANSI N101.2-1972 and NRC Regulatory Guide 1.54 Rev. 0.

VII.6.E NRC Generic Letters

Safety-related valves and other safety-related SSCs were evaluated against the requirements of GL 89-10, GL 95-07, and GL 96-06 to determine if any changes were required as a result of the MUR power uprate. No required changes were identified.

VII.6.E.i GL 89-10 Motor-Operated Valve (MOV) Program

The NRC issued GL 89-10 requiring licensees to develop a comprehensive program to ensure MOVs in safety-related systems would operate under design basis conditions. GL 96-05 provides more complete guidance regarding periodic verification of safety-related MOVs and supersedes GL 89-10 and its supplements with respect to MOV periodic verification.

The review determined that the design basis maximum differential pressures, line pressures, and flow rates for the GL 89-10/GL 96-05 MOVs were not affected by the MUR power uprate. In addition, it was found that MUR power uprate will not affect the maximum ambient temperatures currently used in determining MOV motor capability torque values. Therefore, the MOVs within the scope of GL 89-10 and GL 96-05 are not affected by the MUR power uprate.

VII.6.E.ii GL 95-07 Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves

The NRC issued GL 95-07 to address potential pressure locking and thermal binding of safety-related power-operated gate valves.

The review determined that the MUR power uprate does not affect the pressure locking and thermal binding evaluations previously completed. The power uprate does not affect valve design or valve function. Although there are slight changes in operating conditions in certain systems, these do not affect valve susceptibility to pressure locking or thermal binding. Therefore, the conclusions previously documented in a letter from the NRC to Commonwealth Edison (Reference VII.6-4) for valve pressure locking and thermal binding acceptability are not impacted by the MUR power uprate.

VII.6.E.iii GL 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

The NRC issued GL 96-06 to address (1) the potential for water hammer and two-phase flow conditions during design-basis accidents and (2) the potential for thermally induced overpressurization of piping

sections during design-basis accident conditions. The design-basis conditions of interest are LOCA and MSLB.

As discussed in Sections III.15 and III.16, respectively, both the LOCA and MSLB containment responses were re-analyzed for the MUR power uprate.

The potential for thermally induced overpressurization of isolated piping sections in containment was re-evaluated for the revised profiles and bounding calculated values for internal pressures remain within allowable values. Only slight differences in the post-accident containment temperature profiles resulted from these re-analyses. These slight differences are not expected to materially affect the size of the voids formed in the cooling water system serving the containment air coolers during design-basis accidents. Therefore, the potential for waterhammer and two-phase flow conditions in the cooling water system serving the containment air coolers during design-basis accidents is not affected by the MUR power uprate.

In the course of the review of the analysis of record for computing GL 96-06 waterhammer loads in the cooling water system piping, a discrepant condition was discovered. This condition was entered into the Exelon corrective action program. The condition relates to the rate of collapse of the void created in the containment air coolers under LOCA or MSLB conditions. The void collapse rate is not sensitive to initial void size for the void sizes of interest, and, as discussed in the previous paragraph, these initial void sizes are not materially changed by MUR power uprate conditions. Similarly, the MUR power uprate does not change the containment air cooler or containment air cooler cooling water system component or system design. Therefore, boundary conditions and equipment response are unchanged by the MUR power uprate. Therefore, this issue is unrelated to the MUR power uprate. Final resolution of this discrepant issue is being tracked under the corrective action program.

The potential for thermally induced overpressurization of isolated piping sections in containment was also re-evaluated for the revised containment temperature profiles and it was determined that the internal pressures of isolated sections remain within allowable values.

VII.6.E.iv NRC Generic Safety Issue GSI-191

The Nuclear Regulatory Commission (NRC) identified its concern regarding maintaining adequate long-term core cooling (LTCC) in Generic Safety Issue (GSI) 191 (Reference VII.6-1). The scope of GSI-191 addresses a variety of concerns associated with the operation of the emergency core cooling system (ECCS) and the containment spray system (CSS) in the recirculation mode. These concerns include debris generation associated with a postulated high energy line break, debris transport to the containment sump when the ECCS is realigned to operate in the recirculation mode, and the effects of debris that might pass through the sump screens on downstream components and fuel. Specifically, the debris has been postulated to either form blockages or adhere to the cladding, thereby reducing the ability of the coolant to remove decay heat from the core.

After a LOCA, the chemical makeup of the containment sump and core provides the potential for chemical interactions that may lead to precipitate formation and plate-out on the fuel rods. The LOCA deposition model (LOCADM, Reference VII.6-3) is a calculation tool that can be used to conservatively predict the build-up of chemical deposits on fuel cladding after a LOCA. LOCADM predicts both the

deposit thickness and cladding surface temperature as a function of time that results from coolant impurities entering the core following a LOCA.

Generic Letter (GL) 2004-02 (Reference VII.6-2), issued in September 2004, identified actions that utilities must take to address the sump blockage issue. It is the position of the NRC that plants must be able to demonstrate that debris transported to the sump screen after a loss-of-coolant accident (LOCA) will not adversely affect the long-term operation of either the ECCS or the CSS. The NRC expects utilities to use LOCADM to demonstrate the maximum clad temperature will not exceed 800°F and the thickness of the cladding oxide and fuel deposit does not exceed 0.050 inches.

Byron and Braidwood previously completed a LOCADM evaluation that demonstrated the plants were within the acceptance criteria. The LOCADM evaluation requires plant-specific inputs including core power. The Margin Uncertainty Recovery (MUR) power uprate affects the core power of Byron and Braidwood; therefore, the LOCADM evaluation has been revised to include the updated core power. The LOCADM evaluation conducted with the revised core power value still demonstrates Byron and Braidwood are within the acceptance criteria noted above. The revised LOCADM evaluation indicated that the existing fuel parameters are bounded by the LOCADM acceptance criterion and remain valid at MUR power uprate conditions.

VII.6.F Air Operated Valve Program

The Air Operated Valve (AOV) Program includes the following categories of AOVs:

Category 1 – active valves that are high risk / safety significant, and

Category 2 – active low safety significant, safety-related valves.

A review of component level calculations for Category 1 valves and an evaluation of systems and components for Category 2 valves indicate that the MUR power uprate does not affect the design basis conditions for the Category 1 and Category 2 air-operated valves.

References

VII.6-1 NRC Generic Safety Issue GSI-191, “Assessment of Debris Accumulation on PWR Sumps Performance,” footnotes 1691 and 1692 to NUREG-0933.

VII.6-2 Nuclear Regulatory Commission Generic Letter GL 2004-02, “Potential Impact of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors,” September 2004. (Note: this document is readily retrievable as a PDF file from the NRC website.)

VII.6-3 WCAP-16793-NP, Revision 1, “Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid,” April 2009.

VII.6-4 George F. Dick, Jr., Nuclear Regulatory Commission to Oliver D. Kingsley, Commonwealth Edison, “Response to Generic Letter 95-07 – Braidwood Station, Units 1 and 2; and Byron Station, Units 1 and 2,” dated 12/02/1999 (ML993430056)

VIII CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

1. A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:

A. a description of the change

B. identification of analyses affected by and/or supporting the change

C. justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by change

RESPONSE TO VIII CHANGES TO TECHNICAL SPECIFICATIONS, PROTECTION SYSTEM SETTINGS, AND EMERGENCY SYSTEM SETTINGS

VIII.1 Technical Specification Changes

VIII.1.A Description of Change

The changes to the Operating License, Technical Specifications (TS), TS Bases and Technical Requirements Manual (TRM) proposed in this License Amendment Request are presented in the Attachment 1, Section 2.0, "Detailed Description." In addition to these changes, a revised Steam Generator Tube Rupture and Margin to Overfill Analysis is being presented for NRC approval. This revised analysis is summarized in Attachment 1, Section 3.4.4, "Steam Generator Tube Rupture Analysis and Margin to Overfill Analysis Summary," and described in detail in Attachment 5a, "Steam Generator Tube Rupture Analysis and Margin to Overfill Analysis Report."

VIII.1.B Supporting Analysis

The current Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, rated thermal power (RTP) is 3586.6 MWt. A comprehensive evaluation has been completed, addressing all four units, to confirm that the requested increase in licensed RTP is acceptable. The evaluations/analyses were performed assuming to bound the requested increase in Rated Thermal Power (RTP) to 3645 MWt (i.e., an increase of 1.63%). These evaluations addressed design transients, accidents, nuclear fuel, NSSS systems and Balance of Plant (BOP) systems.

A summary of the supporting analysis is presented in Attachment 1, Section 3.0, "Technical Evaluation." This section summarizes the following major topics:

Section 3.1	Background and General Approach
Section 3.2	Evaluation of Changes to License and Technical Specifications
Section 3.3	LEFM Ultrasonic Flow Measurement and core Thermal Power Uncertainty Calculation Summary
Section 3.4	Analysis Summary

Section 3.4.1	MUR uprate Evaluation Approach
Section 3.4.2	NSSS Design Parameters
Section 3.4.3	Subchanned Analysis Code (VIPRE) and DNB Correlations (ABB-NV and WLOP)
Section 3.4.4	Steam Generator Tube Rupture Analysis and Margin to Overfill Analysis Summary
Section 3.4.5	Plant Modifications
Section 3.4.6	Technical Specification Instrument Setpoint Changes
Section 3.4.7	Grid Stability
Section 3.4.8	Operator Training, Human Factors, and Procedures
Section 3.4.9	NRC Requested Information During the May 18, 2011 Pre-Application Teleconference

The detailed descriptions of these topics are presented earlier in this Attachment 5. The SGTR and MTO analysis is detailed in Attachment 5a. The results of all analyses and evaluations performed were found to be acceptable and will adequately support MUR uprated power conditions.

VIII.1.C Justification for Changes

As stated earlier in Attachment 1, Section 1.0, Summary Description," the proposed Measurement Uncertainty Recapture (MUR) power uprate is based on a change in instrumentation error assumptions specified in 10 CFR 50, Appendix K, "ECCS Evaluation Models." Prior to the subject change, Appendix K required the following: "…it must be assumed that the reactor is operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications." The NRC approved a change to the Appendix K requirements on June 1, 2000 (effective July 31, 2000), that allowed licensees the option that states: "An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation errors."

The reduction in the ECCS evaluation model assumed power level is justified by increased feedwater flow measurement accuracy, which will be achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation.

The justifications for the proposed changes to the Operating License and Technical Specifications associated with the proposed increase in power level are specifically discussed in Section 3.2, "Evaluation of Changes to License and Technical Specifications." The detailed evaluations and analyses performed in support of the proposed changes are discussed above in Section VIII.1.B, "Supporting Analysis."

VIII.2 Protection System Settings Changes

VIII.2.A Description of Change

There are no Reactor Protection System setpoint changes being proposed as part of this license amendment request; however, minor scaling changes such as normalizing the Delta T/T_{ave} and Turbine Impulse Pressure channels will be required to support the new MUR power level.

Note that the pressure coefficient constant, K_3, in the overtemperature delta-T (OTΔT) setpoint equation is being revised from 0.00181 to 0.00135. To support operation at MUR power uprate conditions, new core thermal limits were generated as discussed in, Section III.I.A.5.1, "Core Thermal Limits." The revision to K_3 was required to ensure that the revised core thermal limits were fully protected and to ensure that necessary DNB margin was maintained. The K_3 constant is maintained in the Byron Station and Braidwood Station Core Operating Limits Report (COLR), and does not require a change to Technical Specifications.

VIII.2.B Affect ed Analyses

Reanalyzed events that require the OTΔT trip function to be available for a primary trip function are the Excessive Increase in Secondary Steam Flow (Section III.3), Loss of External Electrical Load/Turbine Trip (Section III.6), Uncontrolled RCCA Bank Withdrawal at Power (Section III.10) and Accidental Depressurization of the Reactor Coolant System (Section III.12). The Chemical and Volume Control System Malfunction that results in a Decrease in Boron Concentration in the Reactor Coolant (Section II.2.8) was also evaluated. The results of all events were determined to remain acceptable.

VIII.2.C Justification for Changes

The results of the reanalyzed events show that the revised OTΔT function continues to perform its intended protective function (Reactor Trip) given the specified revision of the pressure coefficient constant, K_3. Detailed information and results for each affected transient are present in Section III as noted above.

VIII.3 Emergency System Settings Changes

There are no emergency system setpoint changes resulting from the MUR power uprate.