Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion

April 16, 2007

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

Serial No. 06-578B
NL&OS/CDS: R0
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
REQUEST FOR APPROVAL OF TOPICAL REPORT DOM-NAF-5, "APPLICATION OF DOMINION NUCLEAR CORE DESIGN AND SAFETY ANALYSIS METHODS TO THE KEWAUNEE POWER STATION (KPS)"

In a January 31, 2006, public meeting with NRC staff, Dominion Energy Kewaunee (DEK) presented a conceptual approach and implementation strategy for application of approved nuclear core design and safety analysis methods to Kewaunee Power Station (KPS) (reference 1). Fundamental to the proposed approach was creation of a composite topical report (DOM-NAF-5) that would document the application of the relevant methodologies to KPS.

On August 16, 2006, DEK submitted DOM-NAF-5 without attachments A and B (reference 2). On December 6, 2006, Attachment A to DOM-NAF-5, containing CMS benchmark analysis results, was submitted (reference 3).

Attachment B to DOM-NAF-5, containing RETRAN benchmark analysis results, is completed and is attached. This submittal, in conjunction with References 2 and 3, provides the complete contents of topical report DOM-NAF-5. DEK will issue a consolidated version of the topical report, DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," including the supplemental material of Attachments A and B, following NRC review and approval.

DEK is requesting approval of DOM-NAF-5 with the intent of subsequently implementing the Dominion Statistical DNBR Evaluation Methodology with VIPRE-D at KPS. The plant specific application of this methodology will be submitted separately and will employ the VIPRE-D code with the Westinghouse WRB-1 Critical Heat Flux correlation for the thermal-hydraulic analysis of Westinghouse 422V+ fuel assemblies at KPS.

In order to support application of these methods to KPS Cycle 29, DEK requests NRC staff review and approval of DOM-NAF-5 by September 30, 2007. A subsequent administrative license amendment request (LAR) to include DOM-NAF-5 among the reference methodology reports in the KPS Technical Specifications is scheduled to be submitted in June 2007. The requested date for NRC staff approval of that LAR will be January 31, 2008. The LAR approval date supports application of DOM-NAF-5 to KPS Cycle 29, which is scheduled to begin in April 2008.

Should you have any questions, please contact Mr. Craig D. Sly at 804-273-2784.

Very truly yours,

G. T. Bischof
Vice President - Nuclear Engineering

References:

1. Summary of Meeting on January 31, 2006, "To Discuss the Applicability of Dominion Safety and Core Design Methods to Kewaunee Power Station," (TAC No. MC 9566), (ADAMS Accession Number ML 060400098).
2. Letter from G. T. Bischof (DEK) to NRC, "Request for Approval of Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated August 16, 2006 (ADAMS Accession Number ML 062370351).
3. Letter from G. T. Bischof (DEK) to NRC, "Attachment A to Topical Report DOM-NAF-5, 'Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),'" dated December 6, 2006 (ADAMS Accession Number ML 0063410177).

Attachment:

1. DOM-NAF-5, Attachment B, "RETRAN Benchmarking Information," dated March 2007.

Commitments made in this letter: None

cc: Regional Administrator
U. S. Nuclear Regulatory Commission
Region III
2443 Warrenville Road
Suite 210
Lisle, Illinois 60532-4352

Mr. R. F. Kuntz
Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O-7D1A
Washington, D. C. 20555

Mr. S. C. Burton
NRC Senior Resident Inspector
Kewaunee Power Station

Attachment 1

TOPICAL REPORT DOM-NAF-5

ATTACHMENT B

RETRAN BENCHMARKING INFORMATION

DOM-NAF-5, APPLICATION OF DOMINION NUCLEAR CORE DESIGN AND SAFETY ANALYSIS METHODS TO THE KEWAUNEE POWER STATION (KPS), DATED MARCH 2007

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

DOM-NAF-5

Attachment B RETRAN Benchmarking Information

Prepared by:

John C. Lautzenheiser

Mark C. Handrick Mark C. Handrick

Approved:

Cary Laroe
Supervisor, Nuclear Safety Analysis

Table of Contents

TABLE OF CONTENTS 2
1.0 Introduction and Summary 3
1.1 Introduction 3
1.2 Summary 3
2.0 KPS RETRAN Model 4
3.0 Method of Analysis 8
4.0 Demonstration Analysis Results 9
4.1 Loss of Load 9
4.2 Locked Rotor 15
4.3 Loss of Normal Feedwater 24
4.4 Main Steam Line Break 30
4.5 Control Rod Bank Withdrawal at Power 37
4.6 Loss of Flow 44
5.0 Conclusions 49
6.0 References 49

1.0 Introduction and Summary

1.1 Introduction

Topical report VEP-FRD-41, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," (Reference 1) details the Dominion methodology for Nuclear Steam Supply System (NSSS) non-LOCA transient analyses. This methodology encompasses the non-LOCA licensing analyses required for the Condition I, II, III, and IV transients and accidents addressed in the Updated Safety Analysis Report (USAR). The VEP-FRD-41 methods are also used in support of reload core analysis. In addition, this capability is used to perform best-estimate analyses for plant operational support applications. The material herein supplements the applicability assessment of RETRAN methods for Kewaunee Power Station (KPS) that is presented in Section 3.4 of DOM-NAF-5, demonstrating that the VEP-FRD-41 methods are acceptable for the stated applications.

1.2 Summary

This report provides a description of the RETRAN base model for KPS and results of demonstration analyses using this model. The KPS model was developed in accordance with the methods in VEP-FRD-41, with certain noding changes noted below. This assessment reaffirms the conclusion in Section 3.4 of DOM-NAF-5, that the Dominion RETRAN methods, as documented in topical report VEP-FRD-41, are applicable to KPS and can be applied to KPS licensing analysis for reload core design and safety analysis. Dominion analyses of KPS will employ the modeling in VEP-FRD-41, as augmented with the noding changes listed below. Thus, VEP-FRD-41, as augmented, is the Dominion methodology for analyses of non-LOCA NSSS transients for KPS.

The KPS RETRAN base model contains the following alterations in noding with respect to the modeling that is documented in VEP-FRD-41.

a) The KPS model explicitly models the safety injection (SI) accumulators.
b) The KPS model has separate volumes for the steam generator inlet and outlet plenums.
c) The KPS model includes cooling paths between downcomer and upper head (Main Steam Line Break overlay).

2.0 KPS RETRAN Model

A KPS RETRAN-02 Base Model and associated model overlays are developed using Dominion analysis methods described in the Dominion RETRAN topical report (Reference 1). The Dominion analysis methods are applied consistent with the conditions and limitations described in the Dominion topical report and in the applicable NRC Safety Evaluation Reports (SERs).

The KPS Base Model noding diagram is shown on Figure 2-1. Volume numbers are circled, junctions are represented by arrows, and the heat conductors are shaded. This model simulates both reactor coolant system (RCS) loops and has a single-node steam generator (SG) secondary side, consistent with Dominion methodology. The SG primary nodalization includes 10 steam generator tube volumes and conductors. There is a multi-node SG secondary overlay that can be added to the Base Model for sensitivity studies although none of the analysis results presented herein utilize this overlay.

In addition to the base KPS model, an overlay deck is used to create a split reactor vessel model to use when analyzing Main Steam Line Break (MSLB) events, consistent with Dominion methodology. This overlay adds volumes to create a second, parallel flow path through the active core from the lower plenum to the upper plenum such that RCS loop temperature asymmetries can be represented. This overlay also includes flow paths between the downcomer and the upper head to model the small amount of cooling flow to the upper head. These flowpaths may also be added for other events when flashing in the upper head is expected to occur. A noding diagram of the split reactor vessel is shown on Figure 2-2. This figure shows the hot leg volumes (101, 201) and cold leg volumes (116, 216) so the reactor vessel can be seen in context of the RCS interface. Otherwise, the noding for all other regions of the model are unchanged from Figure 2-1.

The base KPS model noding is virtually identical to the Surry (SPS) and North Anna (NAPS) models with the exception of some minor noding differences listed as follows, which are updated from the original list provided in DOM-NAF-5 Section 3.4.1.4.

a) The KPS model explicitly models the SI accumulators.
b) The KPS model has separate volumes for the SG inlet and outlet plenums.
c) The KPS model includes cooling paths between downcomer and upper head (MSLB overlay).

The SI accumulators are part of the KPS model because injection from the accumulators is more likely to occur during a MSLB cooldown event for a two-loop plant. The cooling paths are included in the MSLB overlay to appropriately model the effects of flashing in the head, as noted above. The use of separate volumes for the inlet and outlet should have little effect on transient response since the fluid temperature in these volumes is generally the same as the connecting RCS piping.

The Dominion models, including the KPS model, have some differences compared to the vendor RETRAN model that was used to perform the current USAR analyses. Table 2-1 and the subsequent text discussion provides an overview of these differences. Additional details concerning differences between the Dominion KPS and USAR RETRAN models are discussed in the demonstration analyses in Section 4.

A description of the Dominion RETRAN methodology is provided in Reference 1, where specific model details are discussed in Sections 4 and 5 of that reference.

Table 2-1 Dominion USAR RETRAN Model Comparison

Parameter	**Dominion**	**USAR**
Noding:		
Reactor Vessel	Single flow path - 3 axial nodes for active core (special split core overlay for MSLB only)	Split (two parallel flow path) - 4 axial nodes for active core. Increased nodalization in other vessel regions.
Steam Generator	Single node secondary. Five axial levels (10 nodes) for SG tubes primary side. Local Conditions Heat Transfer model available for loss of heat sink events.	Multi-node secondary. Four axial levels for SG tubes (primary and secondary).
Reactivity Model		
Doppler Feedback	Doppler temperature coefficient that is a function of T_{FUEL}.	Doppler-only power coefficient and a Doppler temperature coefficient effect driven by moderator temperature.
Moderator Feedback	Moderator temperature coefficient	Moderator density coefficient
Decay Heat	ANS-5.1 1979 Standard U-235 with 1500 day burn. Q = 190 MeV/fission. Bounds additional 2σ uncertainty	ANS-5.1 1979 Standard Equilibrium decay heat, Bounds additional 2σ uncertainty
Other:		
Heat Transfer Option	Forced	Forced + Free Convection HT Map
Gap Expansion Model	Used for HZP events only.	Used for all events.

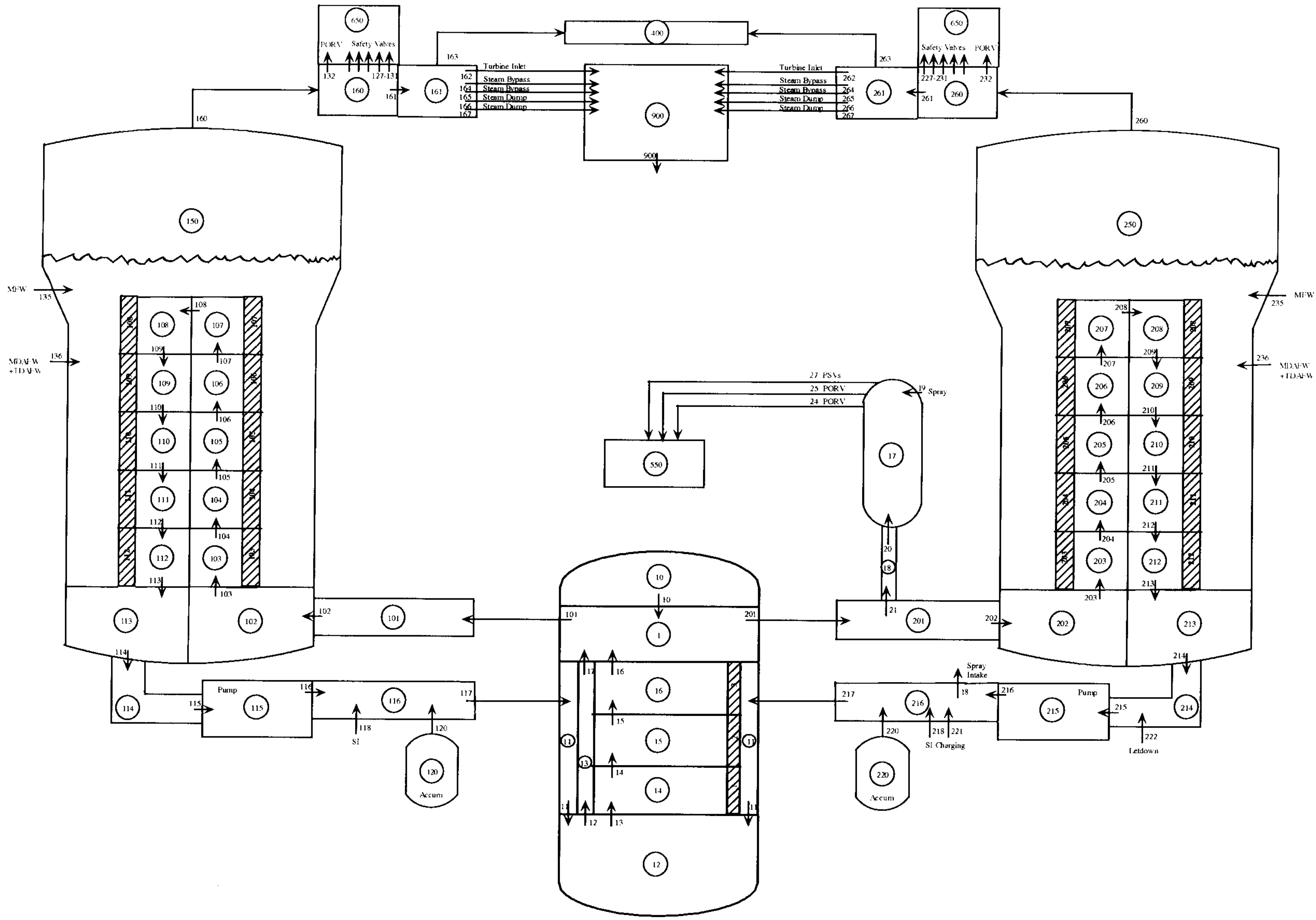


Figure 2-1. KPS Base Model Nodalization Diagram

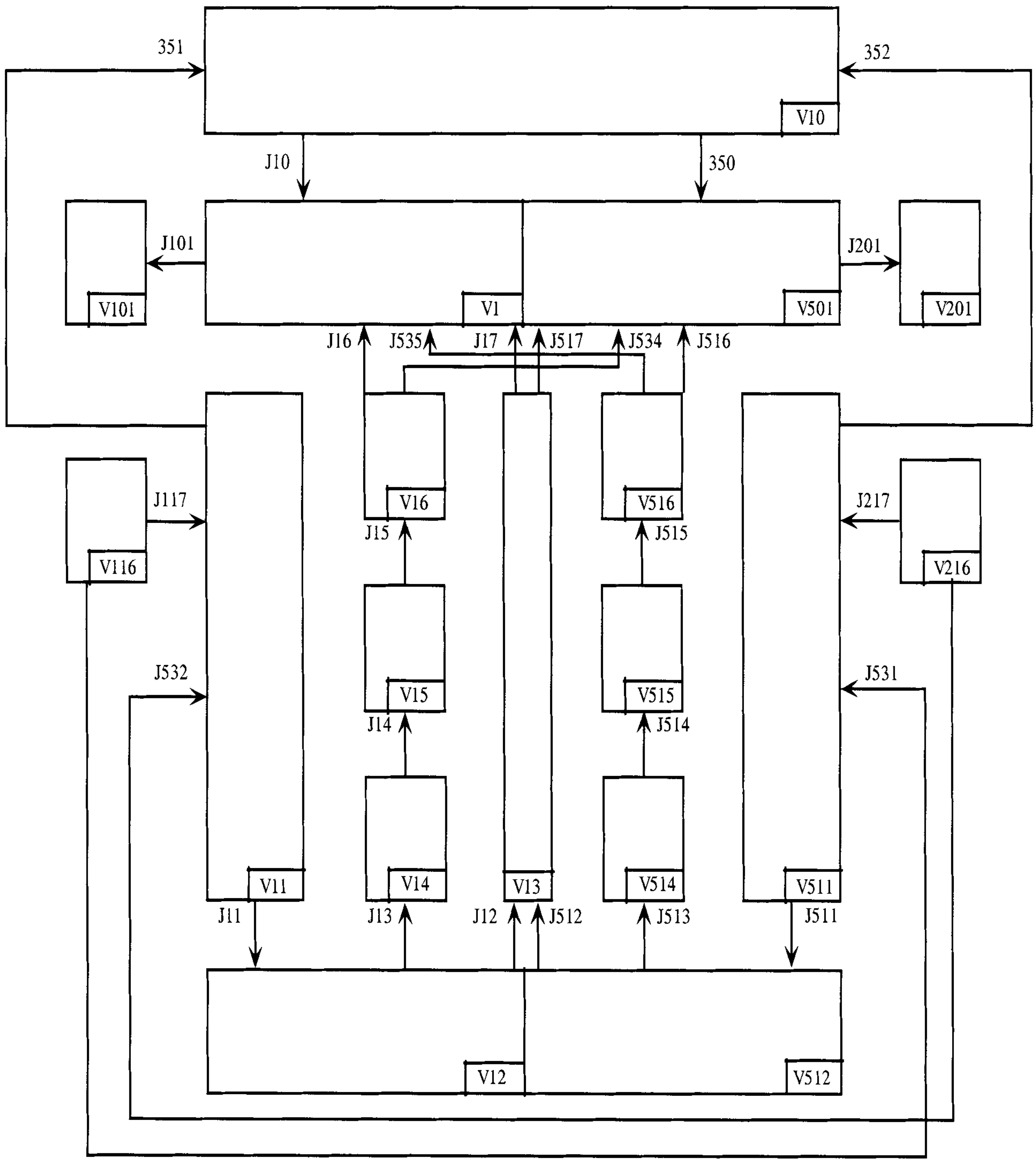


Figure 2-2. KPS Split Vessel Nodalization

3.0 Method of Analysis

As discussed in Section 3.4.3 of DOM-NAF-5, validation of the Dominion KPS RETRAN method involves comparison of RETRAN analyses to the KPS USAR analysis of record (AOR) for select events. These events represent a broad variation in behavior (e.g. RCS heatup, RCS cooldown/depressurization, reactivity excursion, loss of heat sink, etc.), and demonstrate the ability to appropriately model key phenomena for a range of transient responses. The transients selected for comparison with their corresponding KPS USAR section are provided in Table 3-1. For each transient, an analysis is performed using the Dominion KPS RETRAN model and analysis methods. Initial conditions are established to be consistent with the input used in USAR analyses.

Table 3-1 Transients Analyzed for USAR Comparison

Transient	KPS USAR Section
Control Rod Withdrawal at Power	14.1.2
Loss of Flow	14.1.8
Locked Rotor	14.1.8
Loss of Load/Turbine Trip	14.1.9
Loss of Normal Feedwater	14.1.10
Main Steam Line Break	14.2.5

4.0 Demonstration Analysis Results

A summary for each transient comparison is presented in the following sections. Included in each section is an input summary identifying key inputs and assumptions along with differences from USAR assumptions. A comparison of the results for key parameters is provided with an explanation of key differences between the Dominion and USAR cases.

4.1 Loss of Load

The Loss of Load/Turbine Trip (LOL) event is defined as a complete loss-of-steam load and turbine trip from full power without a direct reactor trip, resulting in a primary fluid temperature rise and a corresponding pressure increase in the primary system. This transient results in degraded steam generator heat transfer, reactor coolant heatup and pressure increase following a manual turbine trip.

The LOL transient scenario presented here was developed to analyze primary RCS overpressurization. It is initiated by decreasing both the steam flow and feedwater flow to zero immediately after a manual turbine trip. The input summary is provided in Table 4.1-1. Where differences from USAR inputs exist, they are indicated in the Notes column.

Table 4.1-1 LOL Input Summary

Parameter	**Value**	**Notes**
Initial Conditions		
NSSS Power (MW)	1815.6	Includes 2% uncertainty and pump heat
RCS Flow (gpm/loop)	89,000	Thermal Design
Vessel T_{AVG} (F)	579	Includes +6 F uncertainty
RCS Pressure (psia)	2200	Includes -50 psia uncertainty (delays trip)
Pressurizer Level (%)	53	Includes +5 % uncertainty
SG Level (%)	44	
SG Pressure (psia)	836.8	USAR = 797.98
Assumptions/Configuration		
Reactor trip	-	only Hi Pzr Pressure is active
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
AFW flow	-	Not credited
SG tube plugging (%)	10	Max value
Reactivity Parameters		
Doppler Temp. Coefficient (pcm/F)	-2.5, Most negative	USAR uses least negative Doppler-only power coefficient, and a least negative DTC (driven by moderator temp).
Moderator Temp. Coefficient (pcm/F)	0.0	USAR uses 0.0 $\Delta k/(gm/cc)$ for MDC

Results - LOL

Pressure in the RCS increases during a LOL due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open. The pressurizer pressure response is shown on Figure 4.1-1 and the peak RCS pressure values are listed in Table 4.1-1. Pressure for the Dominion case increases slightly faster initially but peaks at about the same time and same value as the USAR data.

Figure 4.1-2 shows the power response which is nearly constant until a reactor trip on high pressurizer pressure occurs. The Dominion case trips slightly earlier than the USAR data because of the higher RCS pressurization rate.

The core average temperature is shown on Figures 4.1-3. The Dominion and USAR temperature are virtually identical until after the reactor trip and peak pressure occurs, at which time they diverge somewhat but trend together. Because of the RCS heat up and coolant expansion, there is a liquid insurge to the pressurizer as shown by the pressurizer liquid volume increase on Figure 4.1-4. The pressurizer liquid volume increases faster for the Dominion case early in the event, yielding the slightly faster pressure increase discussed earlier. By the time the peak pressure is reached at approximately 11 seconds, the liquid volumes for both cases compare well. The difference after that time is consistent with the temperature response.

The steam generator pressure is shown on Figure 4.1-5. The steam generator heat transfer degradation is strongly related to the secondary pressure increase (saturation temperature increase) since the dominant secondary heat transfer mode is boiling heat transfer. The Dominion case starts from a higher initial pressure, which is the result of model initialization to match the design heat transfer surface area for the single node steam generator, and peaks at a higher pressure than the USAR case.

Table 4.1-2 LOL Primary Overpressure Results

Parameter	DOM	USAR
Sequence of Events:		
Reactor Trip (sec) (High Pzr Pres)	8.4	8.9
Peak RCS Pressure (psia)	2697	2697
Peak MSS Pressure (psia)	1192	1182

Summary - LOL

The Dominion Kewaunee analysis provides results that are similar to the USAR analysis for the LOL event. The RCS temperatures agree well early in the event and although they diverge somewhat later in the event, this does not occur until after the RCS pressure peak occurs and pressure relief begins. There are small differences in pressurization rates early in the event; however, the peak RCS pressure values are the same for both cases. In addition, the peak SG pressure is slightly higher for the Dominion case. There is adequate margin to the RCS pressure acceptance criterion of 2750 psia.

Figure 4.1-1 LOL - Pressurizer Pressure

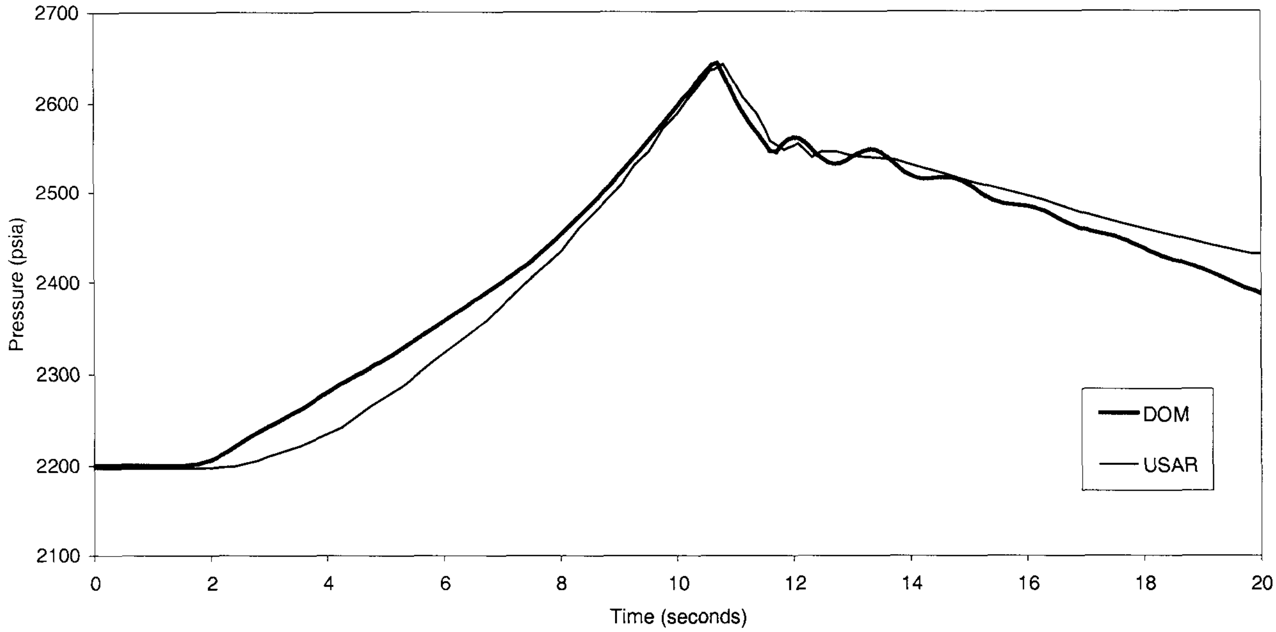


Figure 4.1-2 LOL - Reactor Power

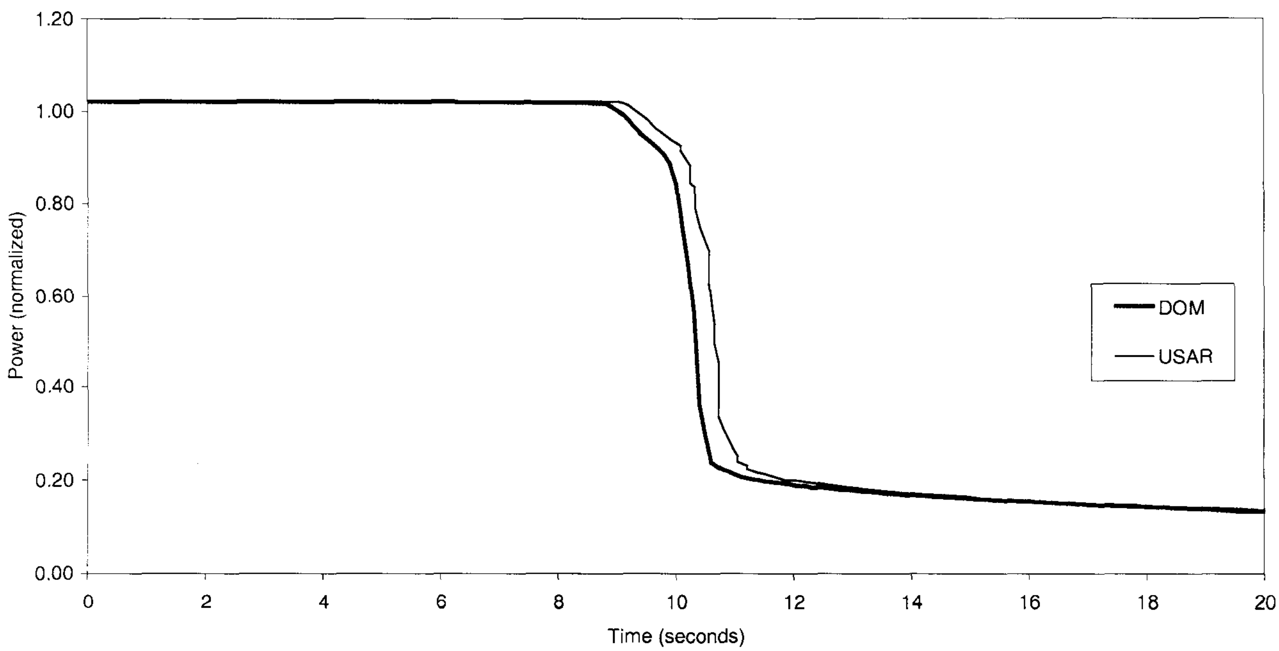


Figure 4.1-3 LOL - RCS Average Temperature

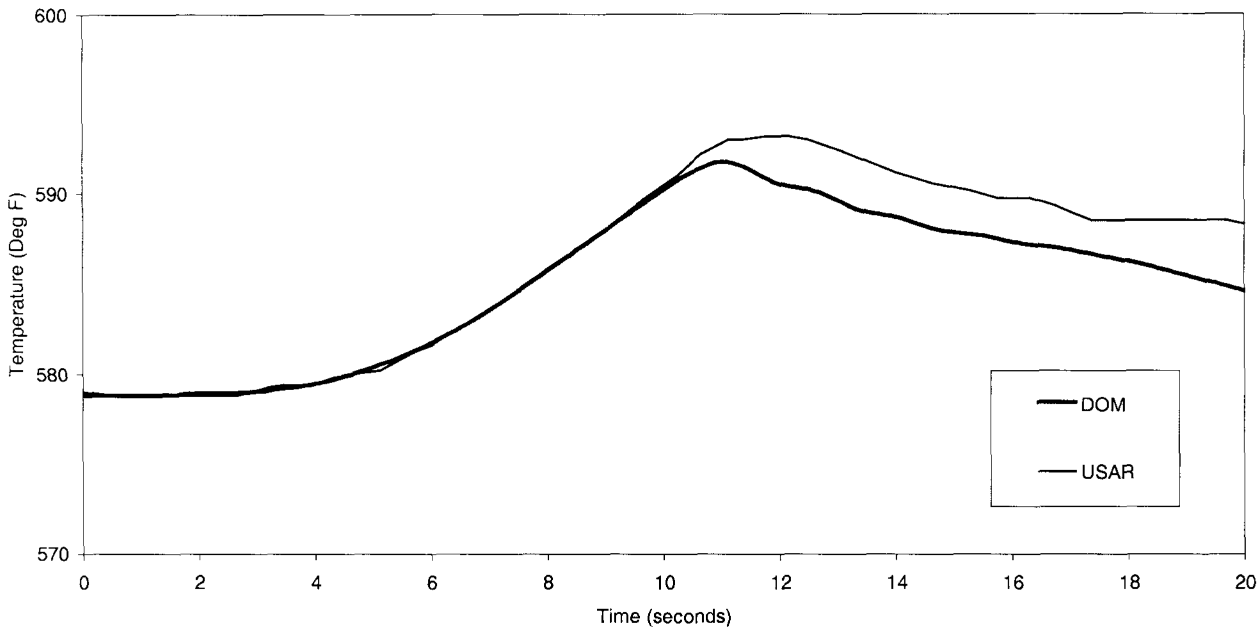


Figure 4.1-4 LOL - Pressurizer Liquid Volume

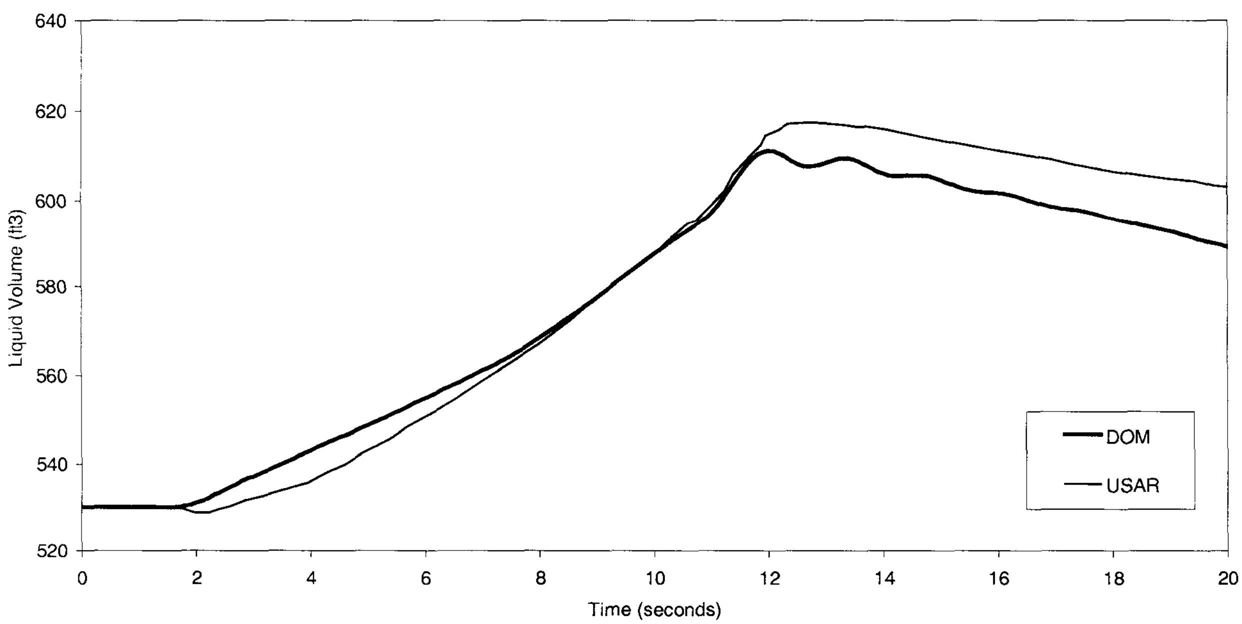
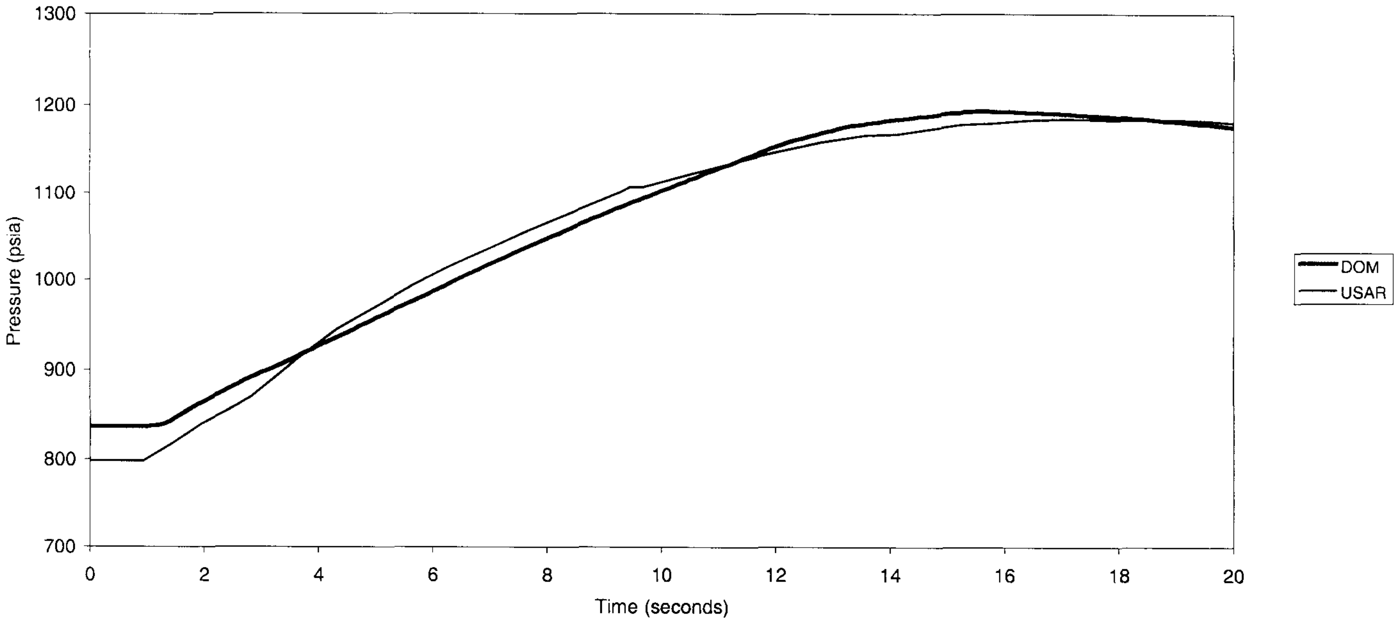


Figure 4.1-5 LOL - Steam Generator A Pressure



4.2 Locked Rotor

The Locked Rotor / Shaft Break (LR) event is defined as an instantaneous seizure of a Reactor Coolant Pump (RCP) rotor, rapidly reducing flow in the affected reactor coolant loop leading to a reactor trip on a low-flow signal from the Reactor Protection System. The event creates a rapid expansion of the reactor coolant and reduced heat transfer in the steam generators, causing an insurge to the pressurizer and pressure increase throughout the reactor coolant system (RCS).

The LR transient scenario presented here was developed to analyze primary RCS overpressurization. It is initiated by setting one RCP speed to zero as the system is operating at full power. The reactor coolant low loop flow reactor trip is credited, with a setpoint of 86.5% of the initial flow. The input summary is provided in Table 4.2-1. Most of the input parameters are the same as those used in the USAR Chapter 14 analyses. Where differences from the USAR inputs exist, they are indicated in the Notes column.

Table 4.2-1 LR Input Summary

Parameter	**Value**	**Notes**
Initial Conditions		
NSSS Power (MW)	1815.6	Includes pump heat and 2% uncertainty
RCS Flow (gpm/loop)	89,000	Thermal Design Flow
Vessel T_{AVG} (F)	579	Includes +6 F uncertainty
Initial Fuel Average Temperature (F)	1332	
RCS Pressure (psia)	2300	Includes +50 psia uncertainty
Pressurizer Level (%)	48	Nominal
SG Level (%)	44	Nominal
SG Pressure (psia)	836.8 (Note 1)	
Assumptions/Configuration		
Reactor trip	-	Only Low RCS Loop Flow is credited
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
AFW flow	-	Not credited
SG tube plugging (%)	10	Max value
RCP/motor moment of inertia (lbm/ft^2)	72,000	90% of nominal
Reactivity Parameters		
Doppler Temp. Coefficient (pcm/F)	-1.2, Least Negative	USAR uses most negative Doppler-only power coefficient, and a least negative DTC (driven by moderator temp).
Moderator Temp. Coefficient (pcm/F)	0.0	USAR uses 0.0 Δk/(gm/cc) for Moderator Density Coefficient

1 – SG pressure adjusted for minimal change to the SG tube surface areas by steady-state initialization

Results – LR RCS Overpressure Case

Pressure in the RCS increases during a LR event due to degraded heat transfer in the steam generator and is alleviated only when the pressurizer safety valves (PSV) open. The pressurizer pressure response is shown on Figure 4.2-1 and the peak RCS pressure values are listed in Table 4.2-1. Pressure for the Dominion case increases slightly faster initially but peaks at about the same time and approximately the same value as the USAR data. Figure 4.2-2 shows the pressure response in the reactor vessel lower plenum, which compares well to the USAR data, particularly until the point of peak pressure.

Figures 4.2-3 and 4.2-4 show the total core inlet volumetric flow and faulted loop volumetric flow (fraction of nominal), respectively. The predicted volumetric core inlet volumetric flow rate decreases more rapidly than the USAR data. This behavior is also present for the faulted loop volumetric flow, where the loop flow reverses earlier than the USAR data. This is conservative behavior in comparison to the USAR data. Each analysis assumes 90% of nominal RCP inertia to limit the coastdown.

Figure 4.2-5 shows the core thermal power response, which matches the USAR analysis well, except for small differences prior to the reactor trip. The USAR Doppler feedback model is a function of (1) Doppler-only power coefficient (DPC), and (2) Doppler temperature coefficient (DTC). The DTC modeled in the USAR analysis is actually a function of moderator temperature rather than fuel temperature. The Dominion reactivity model uses a Doppler Temperature Coefficient, dependent only on changes in fuel temperature, which provides the prompt feedback component. The Dominion DTC model is described in Section 5.13 of Reference 1. The Dominion model predicts core power to decrease prior to reactor trip, which is expected due to negative Doppler feedback as the fuel temperature increases.

The computed core average heat flux shown in Figure 4.2-6 compares well with the USAR data. The small difference in the core heat flux response during the first second of the transient is probably due to the difference in modeling of the core heat transfer coefficients (HTC). In the USAR analysis, the core HTCs are held fixed at their initial values. In the Dominion model, the forced convection HTCs are allowed to decrease with the decaying RCS flow rate, effectively reducing the core heat flux during the first second of the event.

The faulted loop hot leg and cold leg temperatures are shown on Figure 4.2-7. The Dominion and USAR temperature are virtually identical until time when the peak pressure occurs, at which time they diverge somewhat but trend together.

A summary of the LR transient analysis comparison is provided in Table 4.2-2.

Table 4.2-2 LR RCS Overpressure Results

Parameter	DOM	USAR
Sequence of Events:		
Reactor Trip on Low RCS Flow (sec)	0.80	0.80
Peak RCS Pressure (sec)	4.2	4.5
Peak RCS Pressure (psia)	2681	2683

Summary - LR RCS Overpressure Case

The Dominion Kewaunee analysis provides results that are similar to the USAR analysis for the LR event. There are small differences in RCS coastdown flow and core power response during the early portion of the transient. However these differences occur prior to the time of peak RCS pressure (at approximately 4 seconds), and are relatively insignificant for this transient. The peak RCS pressure values are essentially the same for both cases. The predicted peak RCS pressure for the Dominion model (2681 psia) is just slightly below the USAR peak pressure (2683 psia). In each case, there is adequate margin to the RCS peak pressure acceptance criterion of 2750 psia.

Figure 4.2-1 LR - Pressurizer Pressure

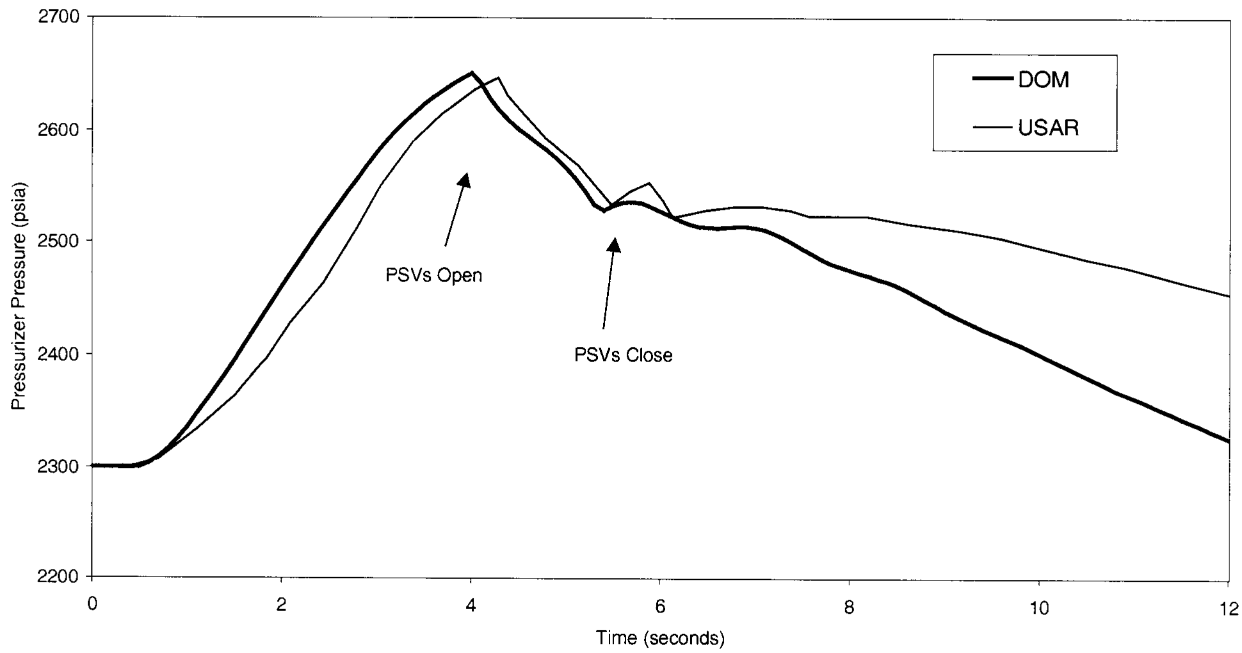


Figure 4.2-2 LR - Lower Plenum Pressure

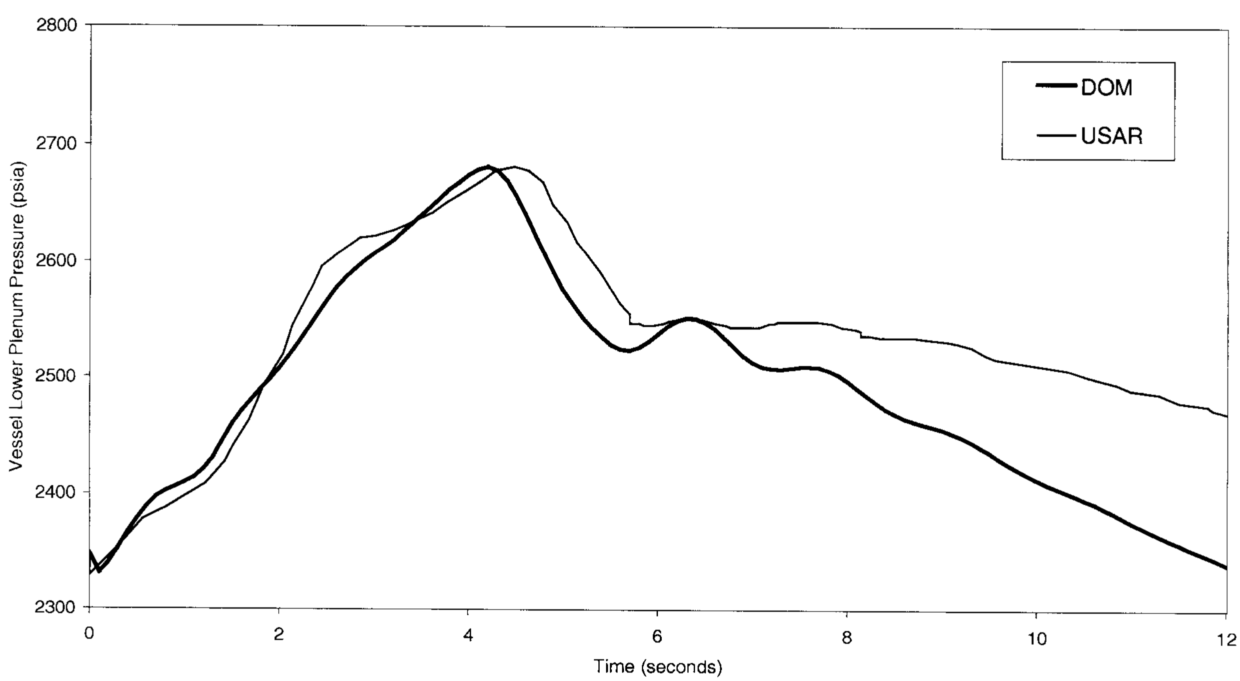


Figure 4.2-3 LR - Core Inlet Volumetric Flow

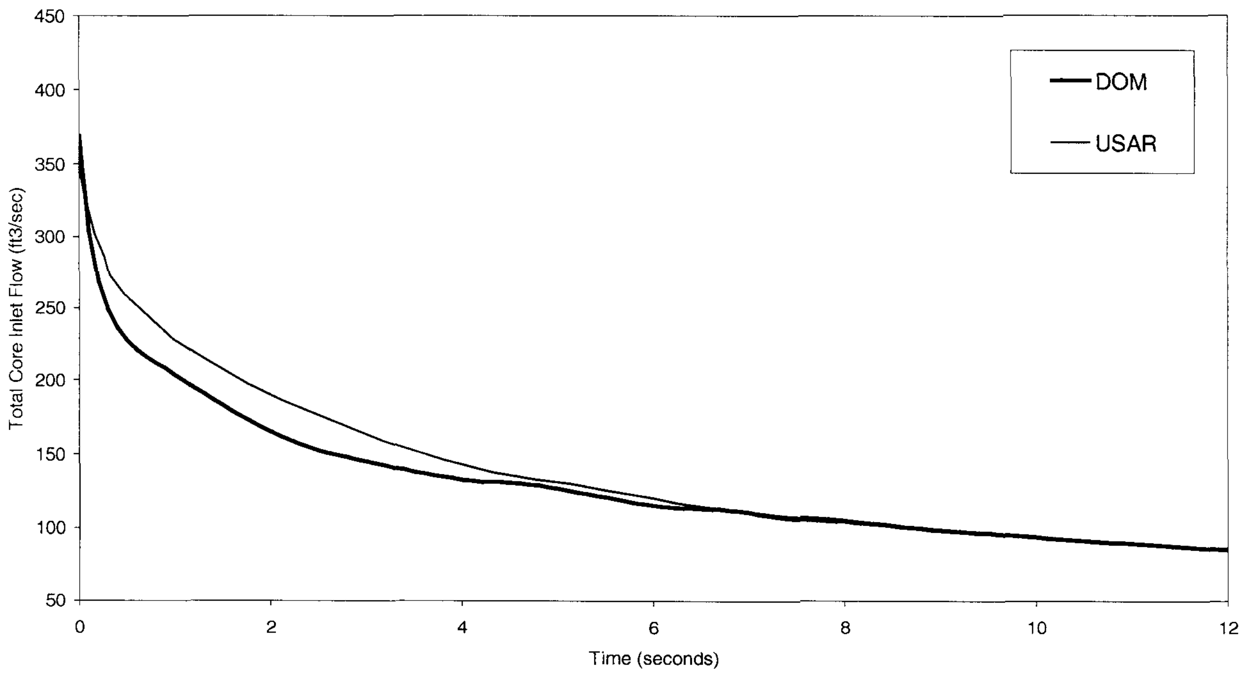


Figure 4.2-4 LR - Faulted Loop Volumetric Flow

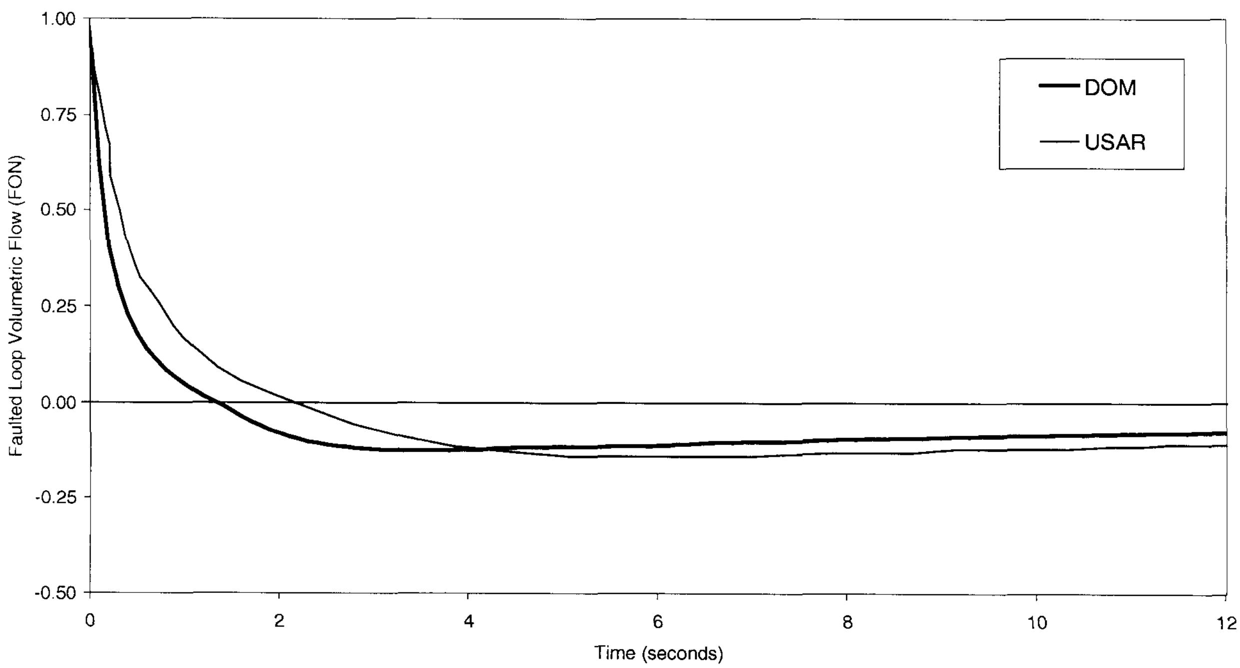


Figure 4.2-5 LR - Core Power

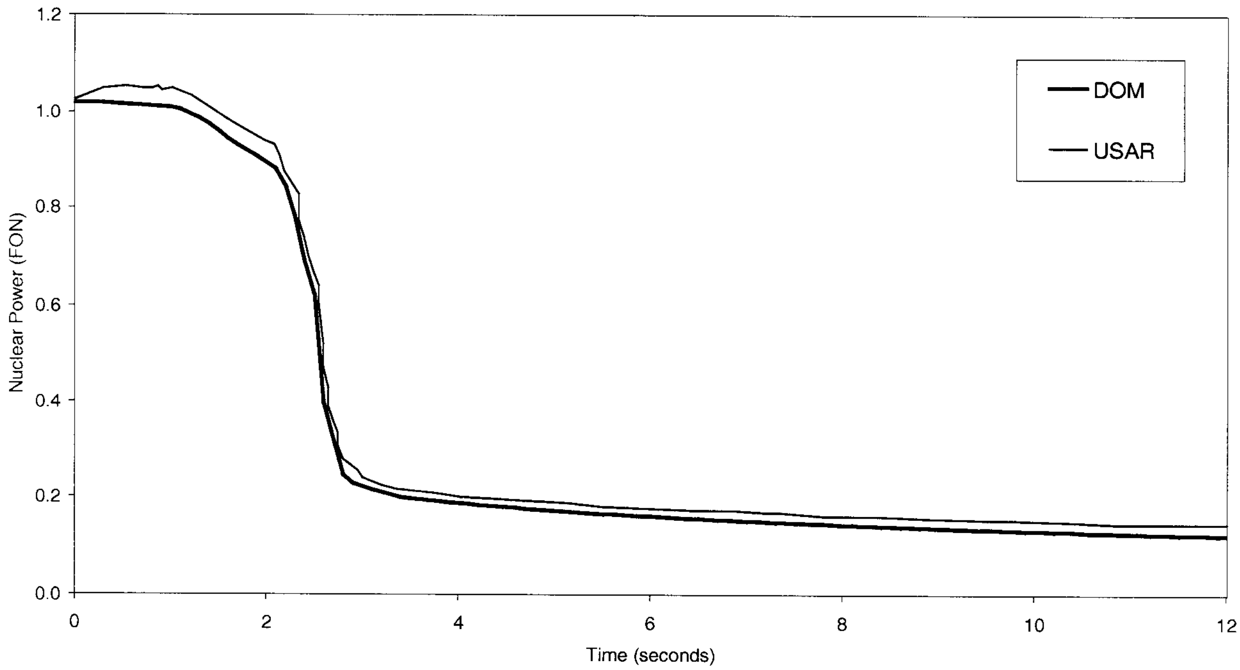


Figure 4.2-6 LR - Core Heat FLux

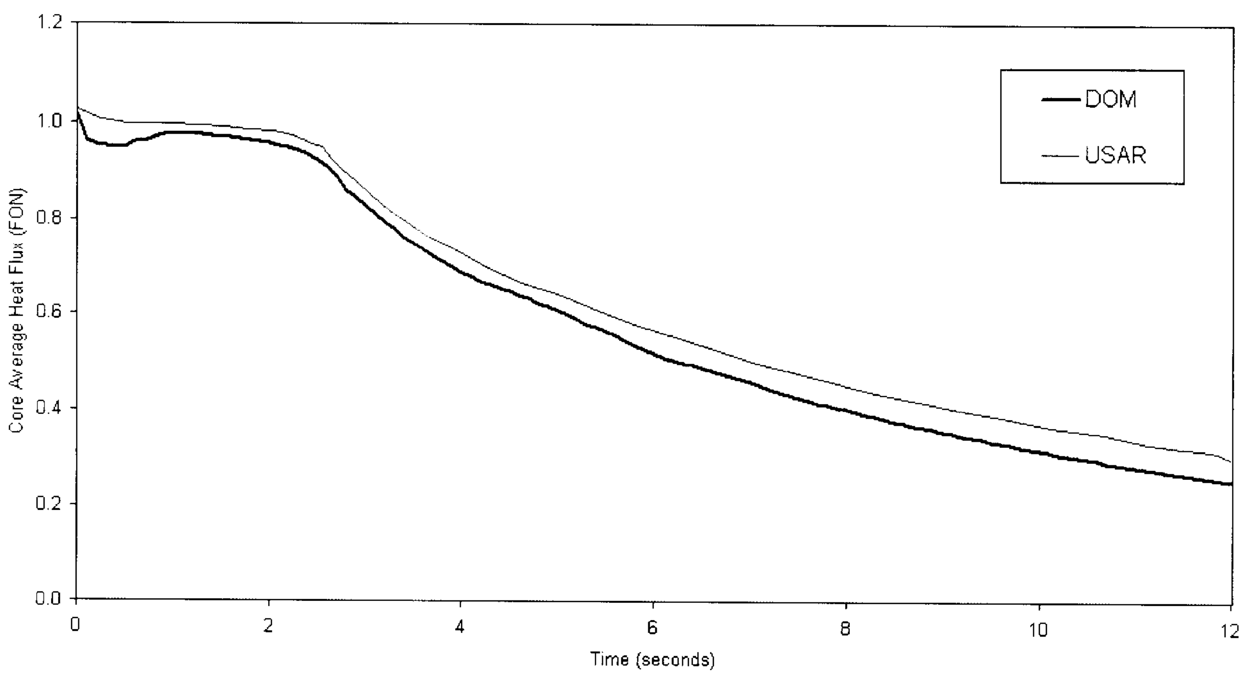
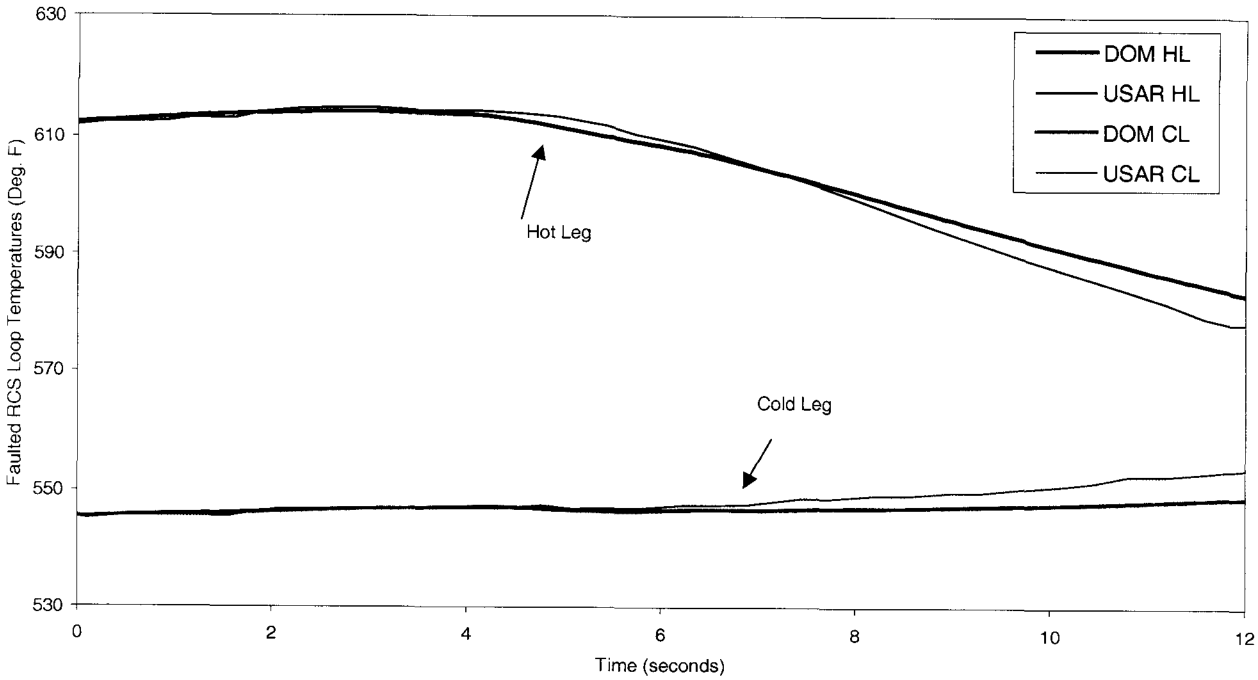


Figure 4.2-7 LR - Faulted Loop Temperatures



LR Peak Cladding Temperature

The Locked Rotor event is also analyzed to demonstrate that a coolable core geometry is maintained. Acceptance criteria for this analyses are met by showing that the peak cladding temperature (PCT) remains below 2700 °F, and that the oxidation level is below 16.0 percent by weight. A hot spot evaluation is performed to calculate the peak cladding temperature and oxidation level. The Dominion Hot Spot model is described in Topical Report VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient." (Reference 2) The Dominion Hot Spot model was used to evaluate the Kewaunee PCT and oxidation level for the LR event.

The Dominion hot spot model is used to predict the thermal-hydraulic response of the fuel for a hypothetical core hot spot during a transient. The hot spot model describes a one-foot segment of a single fuel rod assumed to be at the location of the peak core power location during a transient. The hot spot model uses boundary conditions from the LR system transient analysis to define inlet flow and core average power conditions. The hot spot model uses Kewaunee-specific values for fuel dimensions, fuel material properties, fluid volume, and junction flow areas.

The hot spot model is run to 0.001 seconds and a restart file is saved. Upon restart, the fuel/cladding gap conductance (thermal conductivity) is modified to simulate gap closure by setting the gap heat transfer coefficient to 10,000 Btu/ft^2-hr-°F for a gap conductance of 3.125 Btu/ft-hr-°F. The hot spot model input summary is provided in Table 4.2-3. Most of the input parameters are the same as those used in the USAR Chapter 14 analyses. Where differences from the USAR inputs exist, they are indicated in the Notes column.

Table 4.2-3 Hot Spot Model Input Summary

Parameter	**Value**	**Notes**
Computer Code Used	RETRAN-02	USAR uses FACTRAN
Initial Conditions		
Ratio of Initial to Nominal Power	1.02	
RCS Thermal Design Flow (gpm/loop)	89,000	
Hot Spot Peaking Factor	2.50	
Assumptions/Configuration		
Pre-DNB Film Heat Transfer Coefficient	Thom	USAR uses Dittus-Boelter or Jens-Lottes
Time of DNB (sec)	0.001	
Post DNB Film Boiling Heat Transfer Coefficient	Bishop-Sandberg-Tong	
Fuel Pin Model		
Post DNB Gap Heat Transfer Coefficient	10,000	

(Btu/hr-ft2-°F)		
Gap Thermal Expansion Model activated?	Yes	
Zircaloy-Water Reaction activated?	Yes	

LR Peak Cladding Temperature Results

The peak cladding inner surface temperature obtained from the Dominion Kewaunee hot spot model is 1633 °F. The maximum zircaloy-water reaction depth into the cladding is 2.04E-06 feet, which corresponds to approximately 0.10% by weight based on the nominal cladding thickness of 2.025E-03 feet. A summary of the LR Peak Cladding Temperature Hot Spot analysis comparison is provided in Table 4.2-4.

Table 4.2-4 LR Hot Spot Results

Parameter	DOM	USAR
Peak Cladding Temperature	1633 °F	1900 °F
Maximum Zr-water reaction (w/o)	0.10	0.61

The Dominion peak cladding temperature and maximum oxidation values are less than the USAR values, however both cases demonstrate considerable margin to the acceptance criterion of 2700 °F and 16.0% by weight, respectively.

The difference in zirconium-water reaction results is understood by examination of the Baker-Just parabolic rate equation, which shows that the zirconium-water reaction becomes significant above cladding temperatures of 1800 °F. Since the Dominion hot spot model results do not predict peak cladding temperatures of this magnitude, it is expected that the corresponding zirconium-water reaction would be less than the USAR analysis. In the RETRAN-02 code, the rate of change in the reacting metal-oxide interface is proportional to $\exp(-41200/T)$ where T is the absolute clad temperature (°R). The ratio of $\exp[-41200/(1633+460)]$ / $\exp[-41200/(1900+460)] \approx 0.1$. This value corresponds approximately to the oxidation ratio of the Dominion vs. USAR result.

4.3 Loss of Normal Feedwater

The Loss of Normal Feedwater (LONF) event causes a reduction in heat removal from the primary side to the secondary system. Following a reactor trip, heat transfer to the steam generators continues to degrade resulting in an increase in RCS fluid temperature and a corresponding insurge of fluid into the pressurizer. There is the possibility of RCS pressure exceeding allowable values or the pressurizer becoming filled and discharging water through the relief valves. The event is mitigated when Auxiliary Feedwater (AFW) flow is initiated and adequate primary to secondary side heat removal is restored. This analysis shows that the AFW system is able to remove core decay heat, pump heat and stored energy such that there is no loss of water from the RCS and pressure limits are not exceeded. The LONF input summary is provided in Table 4.3-1. Where differences from USAR inputs exist, they are indicated in the Notes column.

Table 4.3-1 LONF Input Summary

Parameter	**Value**	**Notes**
Initial Conditions		
NSSS Power (MW)	1790.62	Includes 0.6% unc. and pump heat
RCS Flow (gpm/loop)	89,000	Thermal Design
Vessel T_{AVG} (F)	579	HFP nominal + 6 F
RCS Pressure (psia)	2300	Nominal + 50 psi
Pressurizer Level (%)	53	Nominal + 5%
SG Pressure (psia)	863.4	USAR= 829.0
SG Level (%)	51	Nominal + 7%
Assumptions/Configuration		
Low-Low Level Reactor Trip Setpoint	0%	Percent of narrow range span
Pressurizer: sprays, heaters, PORVs	-	assumed operable
AFW Temperature (F)	120	max value
AFW Pump configuration	-	one motor-driven pump per SG
Auxiliary feedwater flow rate (gpm)	-	variable as function of SG press.
AFW Delay after Low-Low-SG level (sec)	800	Max delay
Local Conditions Heat Transfer model	active	SG secondary side USAR= multi-node SG
Steam Generator MSSV blowdown (%)	3	USAR= blowdown not modeled
Reactivity Parameters		
Doppler Temp. Coefficient (pcm/F)	-2.5, Most negative	USAR uses most negative Doppler-only power coefficient, and a least negative DTC (driven by moderator temp).
Moderator Temp. Coefficient (pcm/F)	0.0	USAR uses 0.0 $\Delta k/(gm/cc)$ for Moderator Density Coefficient

Results - LONF

The results for the LONF comparison analysis are presented in Table 4.3-2 and Figures 4.3-1 through 4.3-6. The loss of feedwater flow to the steam generators (SG) results in a reduction in SG level until a reactor trip occurs on Low-Low SG level. Normalized power is shown on Figure 4.3-1. The power response is similar for the Dominion and USAR cases, except that the trip for the Dominion case occurs about 10 seconds later due to slightly higher initial fluid mass in the Dominion SG secondary side. (Higher initial mass has been demonstrated in a sensitivity case to yield conservative results due to delayed reactor trip).

The reduction in SG level results in degraded heat transfer from the primary to secondary systems and an increase in RCS temperature, plotted on Figure 4.3-2. The heatup prior to reactor trip is more pronounced for the Dominion model due in part to the delay in reactor trip. After the reactor trip occurs, the RCS cools somewhat until the loss of SG level and related heat removal is no longer able to remove decay and residual heat. The temperature then increases until AFW flow is actuated and adequate heat removal is restored.

The effect of the temperature change is reflected in the fluid density and associated pressurizer level change, as seen on Figure 4.3-3. The initial pressurizer insurge is higher for the Dominion case, which receives a later reactor trip signal as noted earlier. The maximum pressurizer level, which occurs after the reactor trip, is higher for the USAR case. This appears to be due primarily to higher pressurizer spray flow rates and more conservative decay heat assumptions for the USAR cases. Note, both the Dominion and USAR methods use decay heat profiles that conservatively bound the values for the 1979 ANS-5.1 decay heat model plus 2-sigma uncertainty; however, the USAR method uses a higher value for the decay heat multiplier. Also, the Dominion methodology assumes a conservative value for pressurizer spray flow, although the USAR model appears to apply additional conservatism.

Next, pressurizer pressure responds to the level insurge as shown on Figure 4.3-4. The initial pressure increase for the Dominion case is sufficiently high to cause the pressurizer Power Operated Relief Valves (PORV) to open. The subsequent Dominion pressure response is below the USAR profile but eventually increases above the USAR values during the second insurge. Overall, the pressure response for the USAR case is somewhat flatter due to the higher pressurizer spray flow, which tends to suppress the pressure increases. Also, the delay in reactor trip causes the initial pressure increase for the Dominion case to be more severe.

The secondary side response for SG mass and pressure is shown on Figures 4.3-5 and 4.3-6, respectively. Other than the differences in initial pressure and fluid mass, the responses for the Dominion case and USAR case are similar. Note, the Dominion case models Main Steam Safety Valve (MSSV) blowdown, resulting in slightly lower steady state SG pressures and a more clearly defined valve opening and closing response. Finally, as noted in the plot for SG mass, once AFW flow is initiated, the SG level gradually increases and adequate heat removal is eventually restored.

Table 4.3-2 LONF Results

Parameter	**DOM**	**USAR**
Sequence of Events:		
Loss of Feedwater (sec)	20.0	20.0
Reactor Trip (sec) (Low-Low SG Level)	65.4	54.5
Peak RCS Pressure (psia)	2433	2341
Peak PZP Liquid Volume (ft3)	842	925

Summary - LONF

The Dominion analysis provides results that are similar to the USAR analysis for the LONF event. Differences in the maximum pressurizer level are explained primarily by differences in pressurizer spray assumptions and decay heat modeling. Both analyses are conservative and demonstrate adequate margin to pressurizer overfill acceptance criteria.

Figure 4.3-1 LONF - Core Power

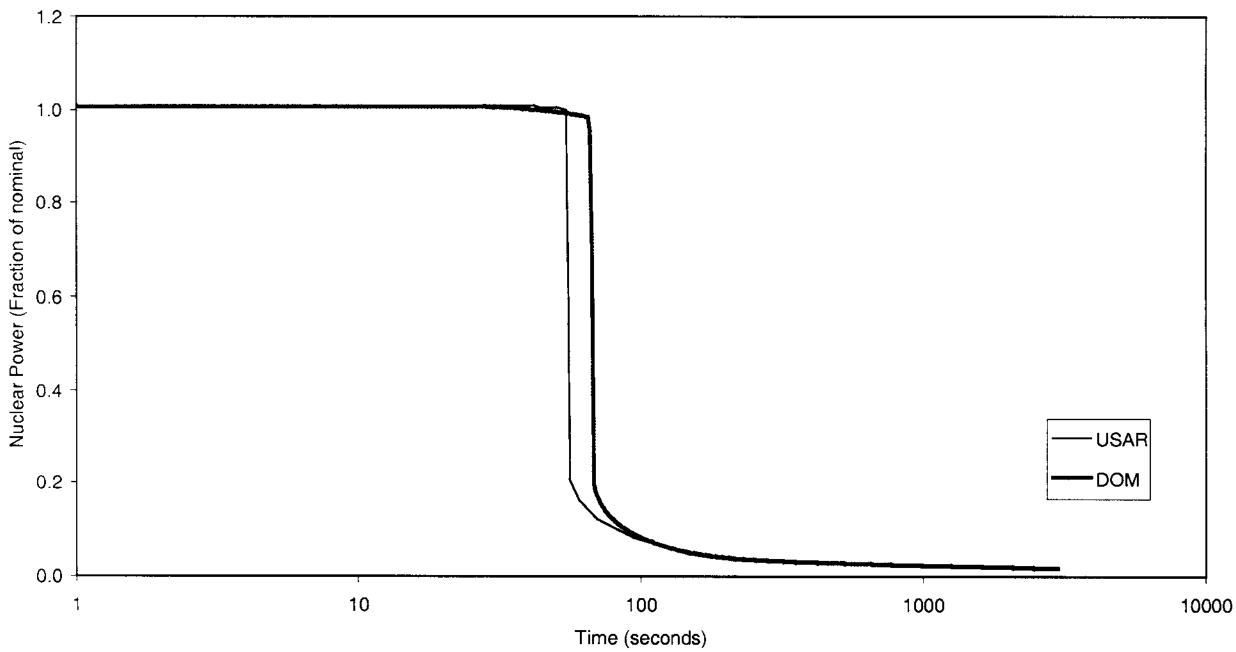


Figure 4.3-2 LONF - RCS Average Temperature

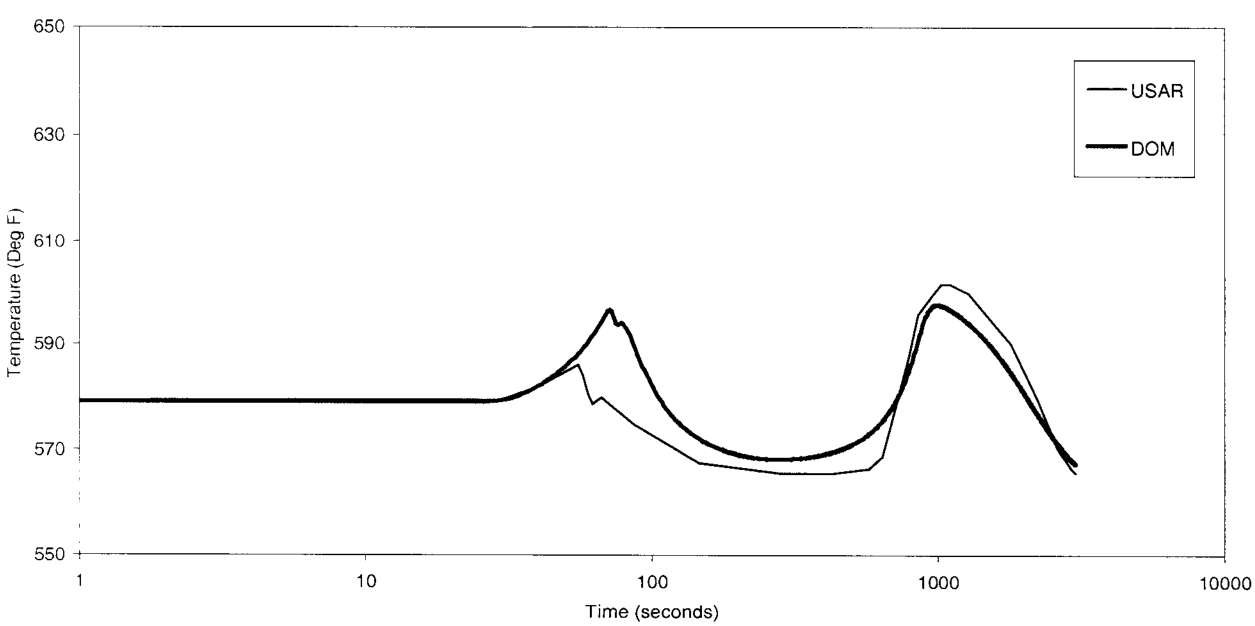


Figure 4.3-3 LONF - Pressurizer Liquid Volume

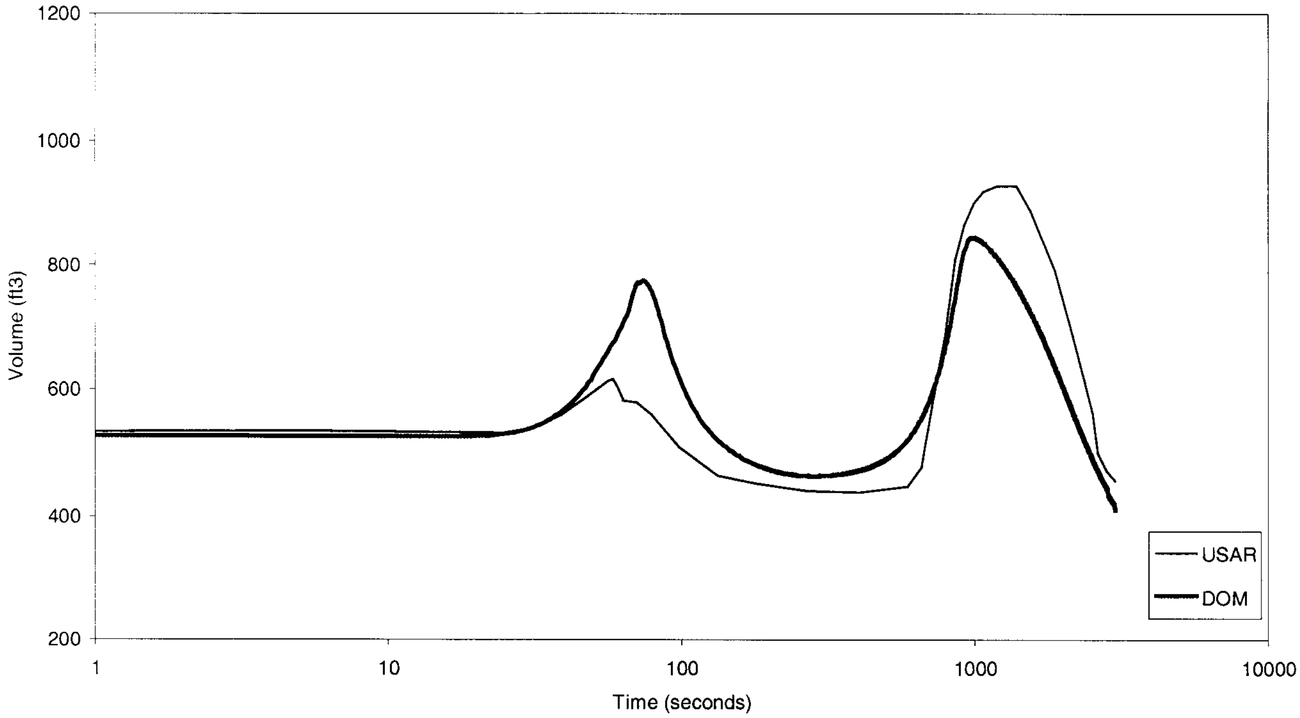


Figure 4.3-4 LONF - Pressurizer Pressure

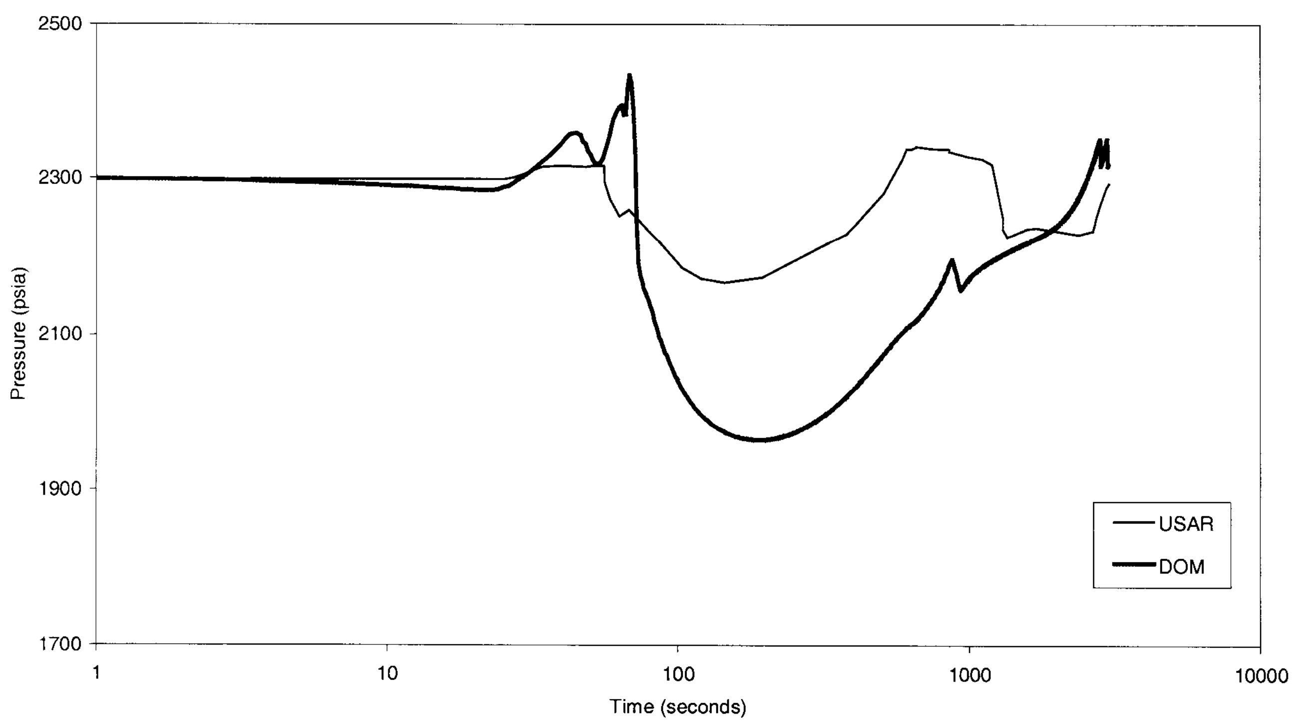


Figure 4.3-5 LONF - Steam Generator Mass

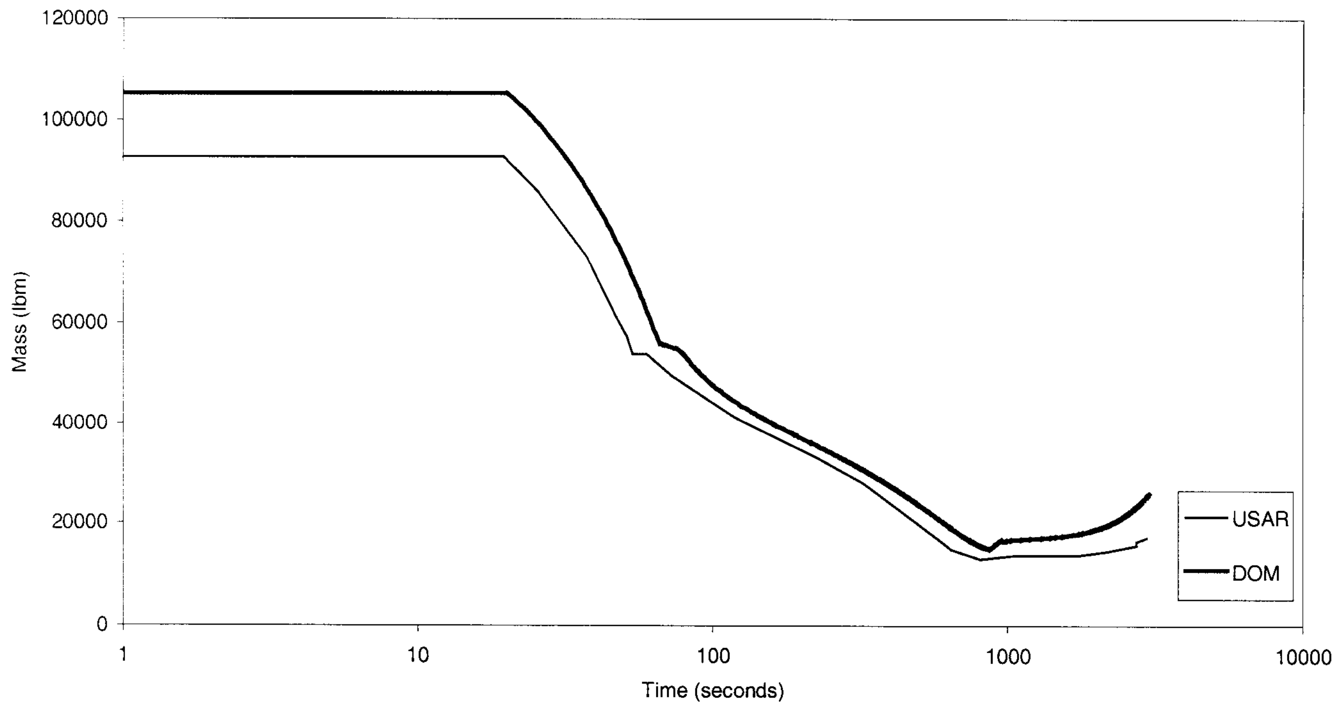
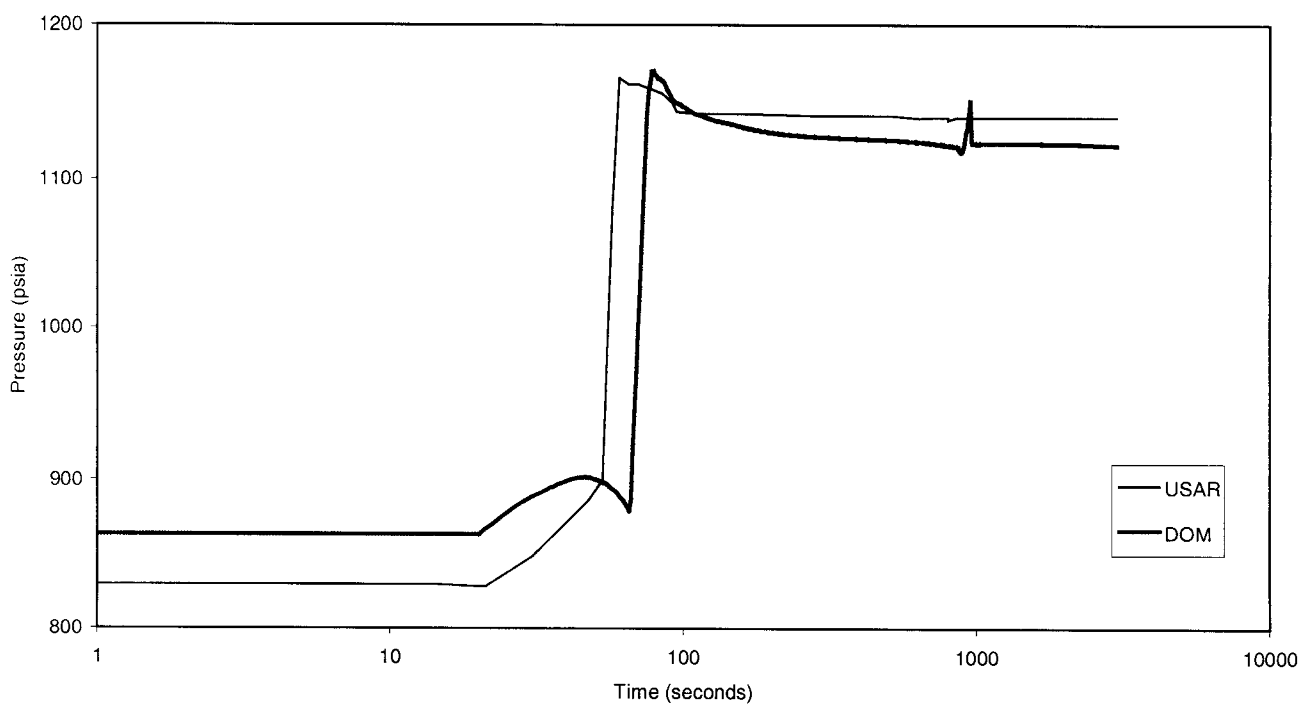


Figure 4.3-6 LONF - Steam Generator A



4.4 Main Steam Line Break

The Main Steam Line Break (MSLB) event is a rupture in the main steam piping resulting in a rapid depressurization of the SG secondary and corresponding cooldown of the primary. The temperature reduction results in an insertion of positive reactivity with the potential for core power increase and DNBR violation.

The MSLB transient scenario presented here is modeled as an instantaneous, double-ended break at the nozzle of one steam generator from hot shutdown conditions with offsite power available. The input summary is provided in Table 4.4-1. Where differences from USAR inputs exist, they are indicated in the Notes column.

Table 4.4-1 MSLB Input Summary

Parameter	**Value**	**Notes**
Initial Conditions		
Core power (MW)	1772.9E-9	HZP; USAR ~ 1.0% RTP
Pump power (MW)	8	Nominal
RCS Flow (gpm/loop)	89,000	Thermal Design
Vessel T_{AVG} (F)	547	HZP nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	21	HZP nominal
SG Level (%)	44	Nominal
Assumptions/Configuration		
Heat transfer option	Forced HT Map (note 1)	USAR = Forced + Free Convection HT Map
Manual Reactor Trip	-	Assumed at time 0 sec
Main feedwater flow (% HFP value)	100	initiated at time 0 sec
Auxiliary feedwater flow rate (gpm)	600	initiated at time 0 sec; per SG
SG tube plugging (%)	0	Minimum value
Reactivity Parameters		
Boron Reactivity (pcm/ppm)	-8.0	USAR= reactivity f(boron concentration).
Doppler Reactivity Feedback	Doppler Power defect, DTC model disabled	USAR - Doppler power defect plus DTC
Moderator Feedback	Moderator density feedback	Moderator density feedback
Decay heat multiplier	1.0	USAR= 1.E-20

1 - Dominion method maximizes heat transfer coefficients for the faulted SG secondary side.

Results – MSLB with Offsite Power Available

The forcing function for the MSLB transient is the break flow. The main steam flow from the faulted and unfaulted SGs is plotted on Figure 4.4-1. Flow from the unfaulted SG stops at approximately 11 seconds due to Main Steam Isolation Valve (MSIV) closure. Flow from the faulted SG continues and there is good agreement between the Dominion and USAR cases for the first 85 seconds. After that time, the break flow predictions from the faulted steam generator begin to diverge slightly. This is primarily due to the differences in the predicted core heat flux response as discussed below.

The SG pressure response (Figure 4.4-2) matches well initially with the USAR data until the MSIV closes. After the MSIV closure, the unfaulted SG pressure increases and the USAR SG pressure remains higher than the Dominion model, most likely due to difference in SG modeling.

The core heat flux response is shown on Figure 4.4-3. The Dominion case core heat flux increases to a higher value compared to the USAR case. This results primarily from differences in the amount and timing of boron reaching the core. The Dominion boron transport method conservatively models the various system delays associating with purging of fluid from the SI lines and transport of borated water from the Refueling Water Storage Tank (RWST) to the core. The initial fluid in the SI piping is assumed to be at a boron concentration of zero ppm. This affects the timing at which boron reaches the core from the RWST and begins to suppress power. In addition, the USAR analysis RCS pressure decreases more quickly than the Dominion pressure. While this difference is relatively small, it allows for greater injection of accumulator flow for the USAR cases, which is further reflected in the power response as well as the RCS temperature and pressure response. Note that since accumulators are included in the KPS RETRAN model, the Dominion boron transport model described in Reference 1 was modified slightly to include the accumulators, however the accumulators are not subject to the system delays associated with the purge time for the Safety Injection (SI) system piping.

The total core reactivity is initially similar to the USAR data as shown in Figure 4.4-4. After 60 seconds, the USAR core reactivity becomes negative, reflecting the sudden increase in core boron concentration. After approximately 100 seconds the negative reactivity for the Dominion model increases as the core boron concentration increases. However, this occurs after the point of peak heat flux.

The core average boron concentration response is shown in Figure 4.4-5. As can be seen, boron starts to reach the core prior to 60 seconds for both the USAR case and the Dominion case. In the USAR case, a higher accumulator flow rate results in a larger increase in core boron concentration. As seen on Figure 4.4-3 for the USAR analysis, the effect on core heat flux is immediate. Once accumulator flow ceases, core boron concentration continues to increase slowly as a result of continued SI flow from the RWST. In the Dominion model, the accumulator flow is less than the USAR case during the first 100 seconds of the transient. Due to the SI piping purge delay times in the Dominion model, borated water from the RWST does not enter the core until approximately 140 seconds. The higher core heat flux predicted by the Dominion case is mainly attributable to the later timing of boron injection.

The pressurizer pressure response is shown in Figure 4.4-6. The pressure initially decreases at a rate comparable to the USAR result. At approximately 20 seconds, the upper head begins to flash and the depressurization rate is decreased. The timing of the upper head flashing and the following depressurization is a contributor to when the accumulators activate. After the accumulator flow stops, the RCS pressure starts to rise slowly.

The reactor vessel inlet temperature response (Figure 4.4-7) shows that the initial cooldown matches well for both the USAR and Dominion cases. After approximately 150 seconds, the temperature differences are attributed to the different core heat flux response. However, this occurs well after the point of peak heat flux, as core power is steadily decreasing.

The sequence of events is compared to the USAR in Table 4.4-2. USAR values are taken from Kewaunee USAR Table 14.2.5-1.

Table 4.4-2 MSLB with Offsite Power Results

Parameter	DOM	USAR
Sequence of Events:		
Initiate Break	0.0	0.01
Unfaulted SG High-High Steam flow setpoint reached	0.79	0.71
Faulted SG Low-Low Steam Pressure Signal	1.02	1.44
Unfaulted SG Low-Low Steam Pressure Signal	1.64	2.01
Safety Injection Actuation Signal	2.79	2.72
Steam line Isolation (MSIV Closure) occurs	10.29	10.22
Peak Heat Flux occurs	100.0	56.5
Feedwater Isolation occurs	86.72	87.82
Peak Heat Flux (fraction of nominal)	0.318	0.288

Figure 4.4-1 MSLB w/ Offsite Power - Steam Flow

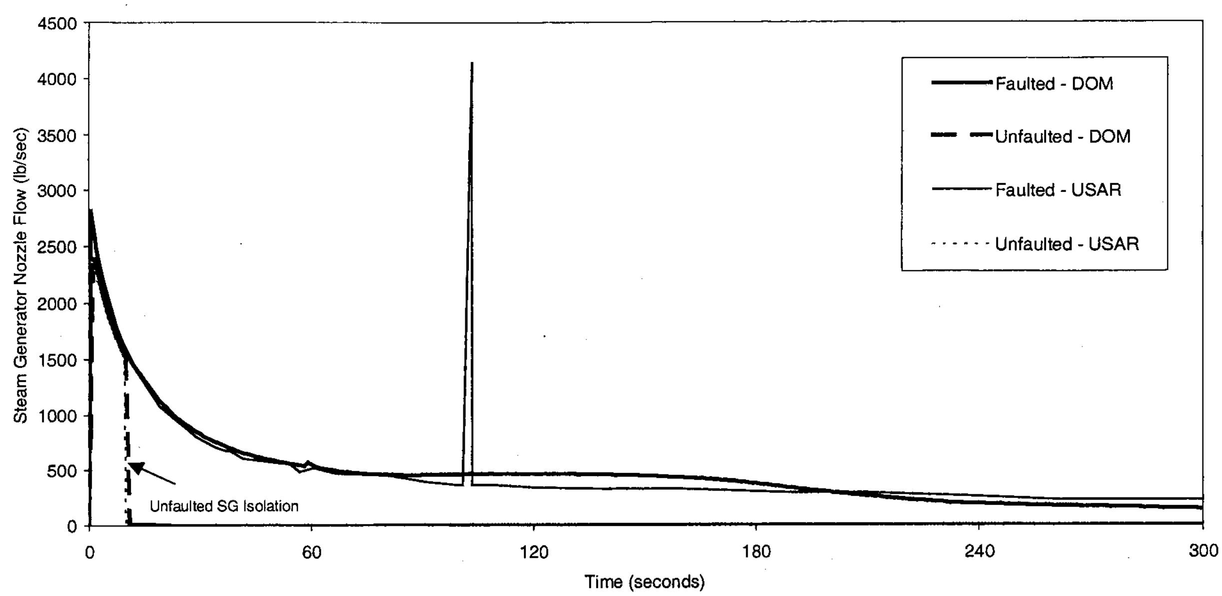


Figure 4.4-2 MSLB w/ Offsite Power - Steam Generator Pressure

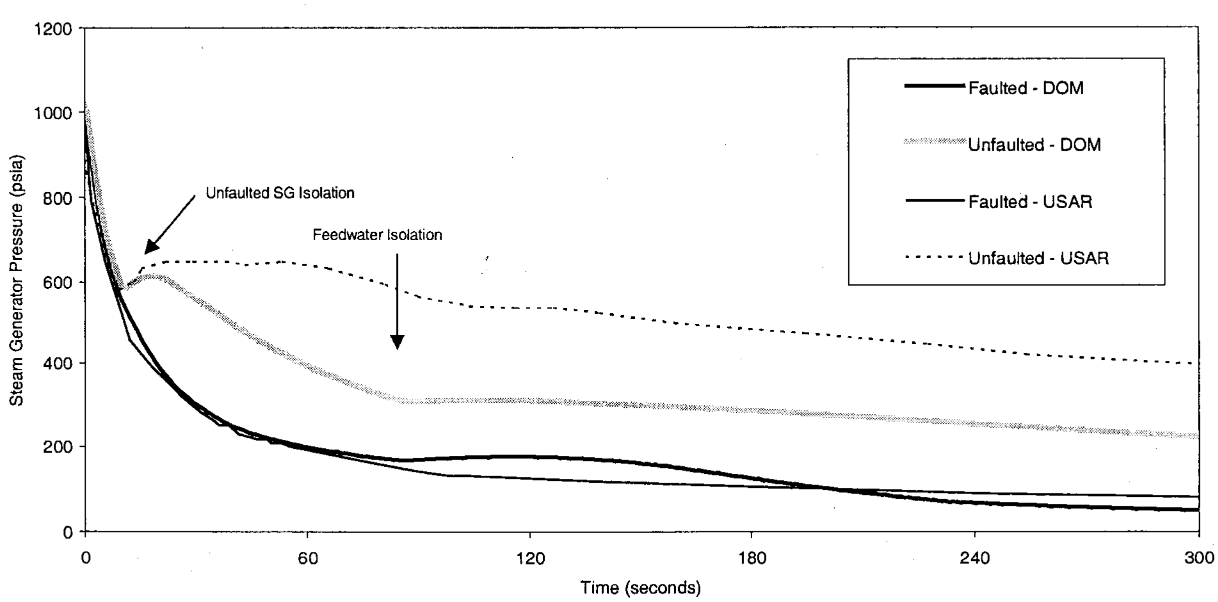


Figure 4.4-3 MSLB w/ Offsite Power - Core Heat Flux

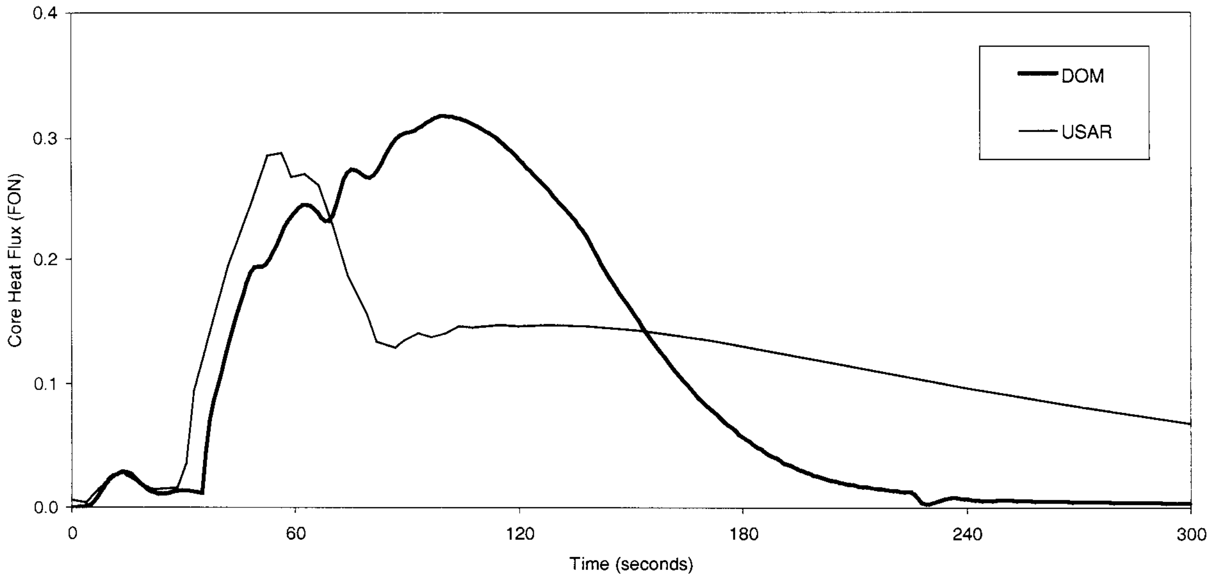


Figure 4.4-4 MSLB w/ Offsite Power - Core Reactivity

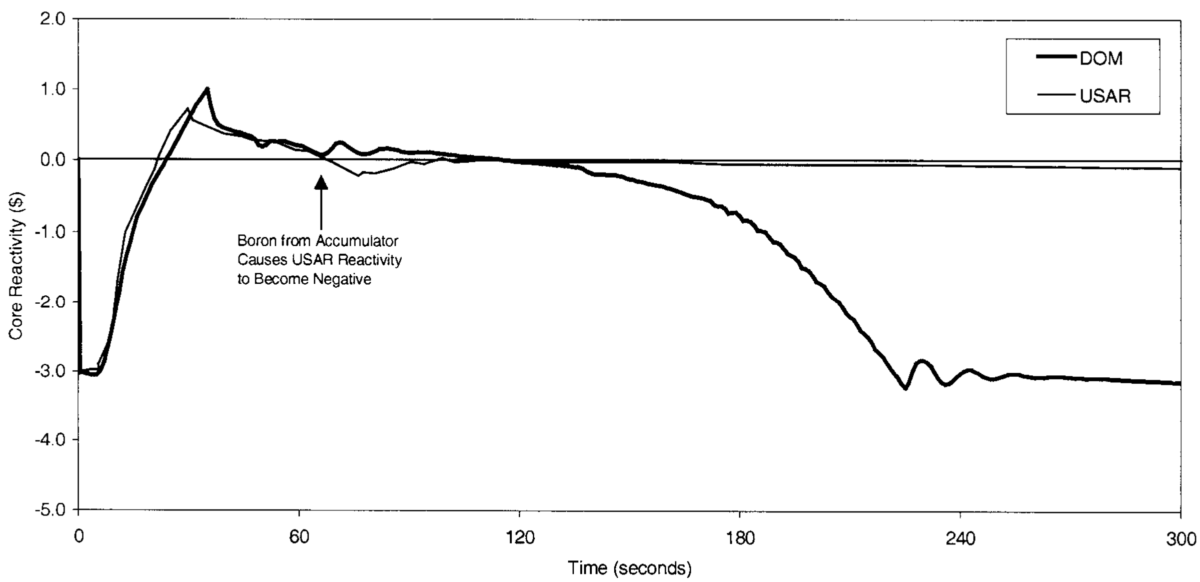


Figure 4.4-5 MSLB w/ Offsite Power - Core Average Boron Concentration

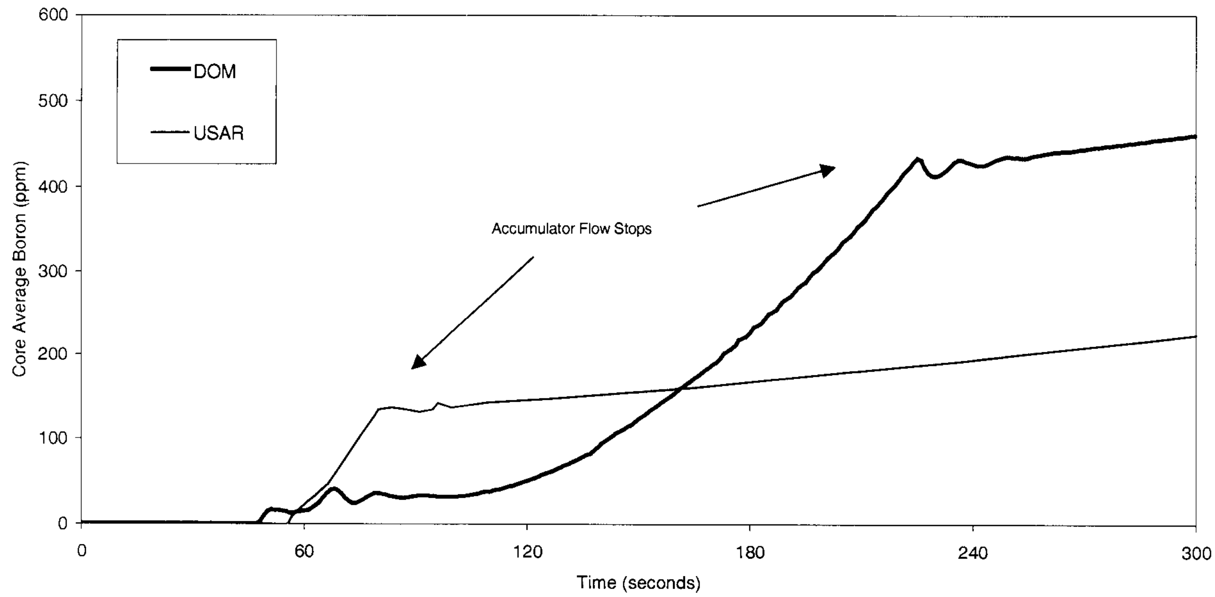


Figure 4.4-6 MSLB w/ Offsite Power - Pressurizer Pressure

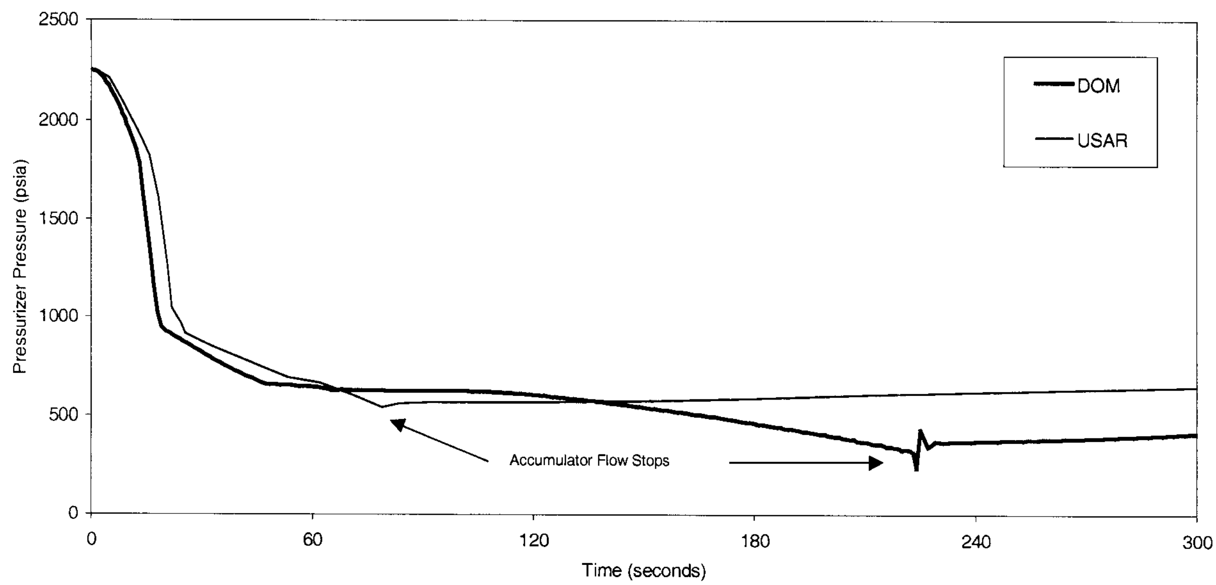
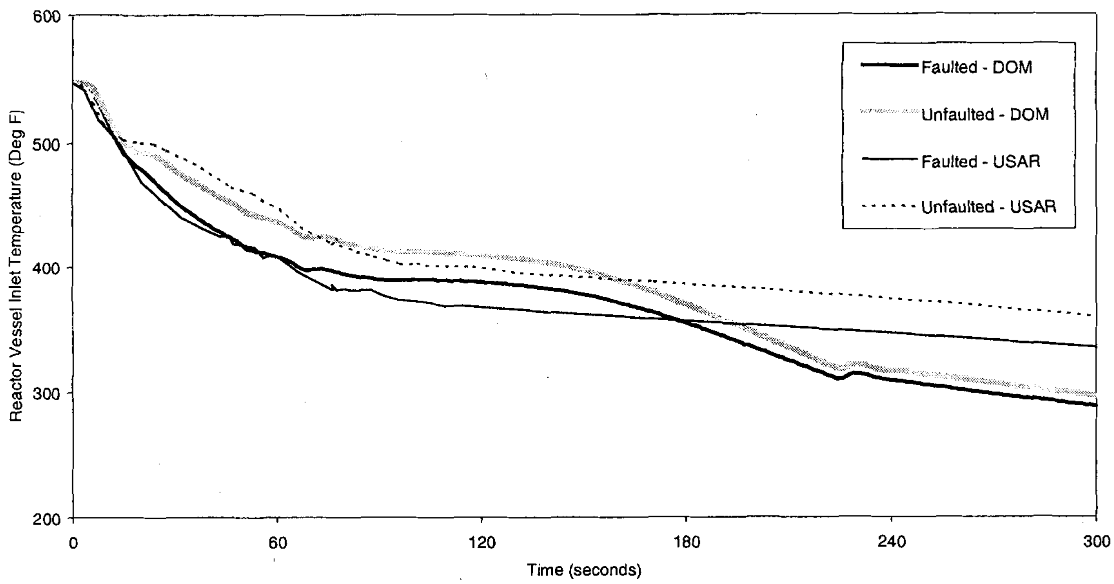


Figure 4.4-7 MSLB w/ Offsite Power - Reactor Vessel Inlet Temperature



Summary - MSLB

This section presents a comparison of a RETRAN-02 Main Steam Line Break transient calculation with the Kewaunee model using the Dominion RETRAN transient analysis methods (Reference 1) compared to the USAR results. The key observations from these comparisons are that:

1) The peak power and heat flux reached with the Dominion methods is higher than the USAR result.

2) The effect of boron is significant in these transients because it can determine the timing and the magnitude of the transient peak power. The core average boron concentration resulting from SI and accumulator flow is different between the USAR and the Dominion model. The amount of flow from the accumulators greatly affects the amount of boron in the system and core power.

4.5 Control Rod Bank Withdrawal at Power

The Control Rod Bank Withdrawal at Power (RWAP) event is defined as the inadvertent addition of core reactivity caused by the withdrawal of rod control cluster assembly (RCCA) banks when the core is above no load conditions. The RCCA bank withdrawal results in positive reactivity insertion, a subsequent increase in core nuclear power, and a corresponding rise in the core heat flux. The RWAP event described here is terminated by the Reactor Protection System on a high neutron flux trip or the overtemperature ΔT trip (OTΔT), consistent with the USAR analyses.

The RWAP event is simulated by modeling a constant rate of reactivity insertion starting at time zero and continuing until a reactor trip occurs. The Dominion analysis involves two different reactivity insertion rates, 3 pcm/sec and 100 pcm/sec that match the reactivity insertion rates described in the USAR. Both cases assume that the reactor is initially operating at 100% power, with minimum core reactivity feedback. Most of the input parameters are the same as those used in the USAR Chapter 14 analyses. Where differences from the USAR inputs exist, they are indicated in the Notes column.

Table 4.5-1 RWAP Input Summary

Parameter	Value	Notes
Initial Conditions		
NSSS Power (MW)	1780	Nominal plus pump heat
RCS Flow (gpm/loop)	93,000	Minimum Measured Flow
Vessel T_{AVG} (F)	573	Nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	48	Nominal
SG Level (%)	44	Nominal
Assumptions/Configuration		
Reactor trip	-	High neutron flux or OTΔT
Automatic rod control	-	Not credited
Pressurizer level control	-	Not credited
Pressurizer heaters	-	Not credited
Pressurizer sprays, PORVs	-	Active
SG tube plugging (%)	10	Max value
Reactivity Parameters		
Doppler Temp. Coefficient (pcm/F)	-1.2	Dominion least negative DTC model. USAR uses least negative Doppler-only power coefficient, and a least negative DTC (driven by moderator temperature).
Moderator Temp. Coefficient (pcm/F)	0.0	USAR uses a value of 0.0 pcm/°F for MTC
Moderator Density Coefficient	N/A	USAR uses 0.0 Δk/(gm/cc)
Fuel Heat Conduction Model		
Initial Fuel Average Temperature	Minimum	USAR targets a minimum value

Results – RWAP 3 pcm/sec Case

Figure 4.5-1 shows the core power response, which is slowly changing until a reactor trip occurs. The core power rate of increase for the Dominion case is somewhat greater than the USAR data. The Dominion case trips on high neutron flux at about 41 seconds, while the USAR case trips on an OTΔT signal at about 45 seconds. It is noted that the USAR core power response very nearly reaches the 118% setpoint for the high flux trip. The difference in reactor trip mechanisms between the Dominion and USAR cases is reasonable, considering the breakpoint for switching between OTΔT and high flux occurs at approximately 4 pcm/sec, as shown in USAR Figure 14.1.2-3. The pressure response also affects the OTΔT setpoint (setpoint will be lower in the USAR case, due to the USAR pressure response discussed below).

The pressurizer pressure response is shown in Figure 4.5-2. For the Dominion case, the pressure rises faster than the USAR result. For the first 5 seconds the results are similar. However, the USAR result shows a flat pressure response from about 5 to almost 40 seconds maintaining the pressure at about 2255 psia. In the Dominion case, the pressure steadily increases rising above the USAR value and continuing until the reactor trips. The RCS pressure response is determined by the modeling assumptions, especially pressurizer spray flow. As noted previously in Section 4.3, the Dominion method uses a conservative value for pressurizer spray flow rate; however, it appears that the USAR model adds additional conservatism, which suppresses the pressure increase associated with the RWAP event.

Figure 4.5-3 shows the RCS Loop A average temperature. There is good agreement between the USAR analysis and the Dominion model, as the peak temperatures are approximately 585 °F for the USAR, and 584 °F for the Dominion model. The time of peak temperature is related to the time of reactor trip as shown in Figure 4.5-1. The sequence of events for the 3 pcm/sec RWAP transient is compared to the USAR in Table 4.5-2.

Table 4.5-2 RWAP 3 pcm/sec Time Sequence of Events

Event	Time (seconds)	
	DOM	**USAR**
Reactivity Insertion at 3 pcm/sec	0.0	0.0
Reactor Trip Signal Initiated	41.27*	45.28**

* *Trip on high neutron flux*

** *Trip on OTΔT*

Results – RWAP 100 pcm/sec Case

Figure 4.5-4 shows the core power response which rises rapidly until a reactor trip on high flux occurs. The Dominion case trips on a high neutron flux signal of 118% at about 1.8 seconds, compared to about 2.03 seconds for the USAR case (each includes a 0.65 second delay). The Dominion case includes decay heat while the USAR analysis neglects decay heat. The effect of decay heat modeling differences is seen post-trip, where the USAR case power drops to the delayed neutron stable period following shutdown, while the Dominion case follows a decay heat curve defined by the ANS-5.1 1979 single isotope (U-235) model.

The 100 pcm/sec transient is a fast transient and the time period before the reactor trip is so brief that the any differences in fuel pin heat transfer modeling assumptions have little impact on Doppler reactivity feedback.

The pressurizer pressure response is shown in Figure 4.5-5. As is the case with the RWAP analysis for a 3 pcm/sec reactivity insertion rate, the Dominion model predicts a higher pressurizer pressure response. This is mainly due to the differences in modeling of the pressurizer spray.

Figure 4.5-6 shows the RCS Loop A average temperature. There is good agreement between the USAR analysis and the Dominion model, as the peak temperatures are approximately 576 °F for both models. The sequence of events for the 100 pcm/sec RWAP transient is compared to the USAR in Table 4.5-3.

Table 4.5-3 RWAP 100 pcm/sec Time Sequence of Events

Event	Time (seconds)	
	DOM	**USAR**
Reactivity Insertion at 3 pcm/sec	0.0	0.0
Reactor Trip Signal Initiated	1.78*	2.03*

* *Trip on high neutron flux*

Figure 4.5-1 RWAP 3 pcm/sec - Core Power

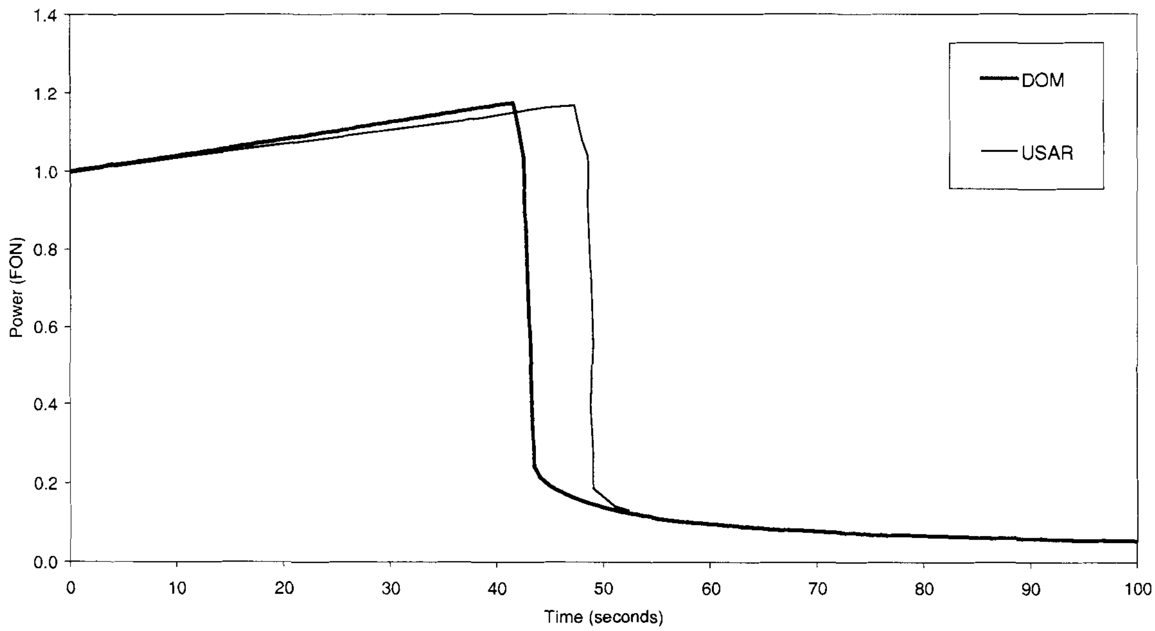


Figure 4.5-2 RWAP 3 pcm/sec - Pressurizer Pressure

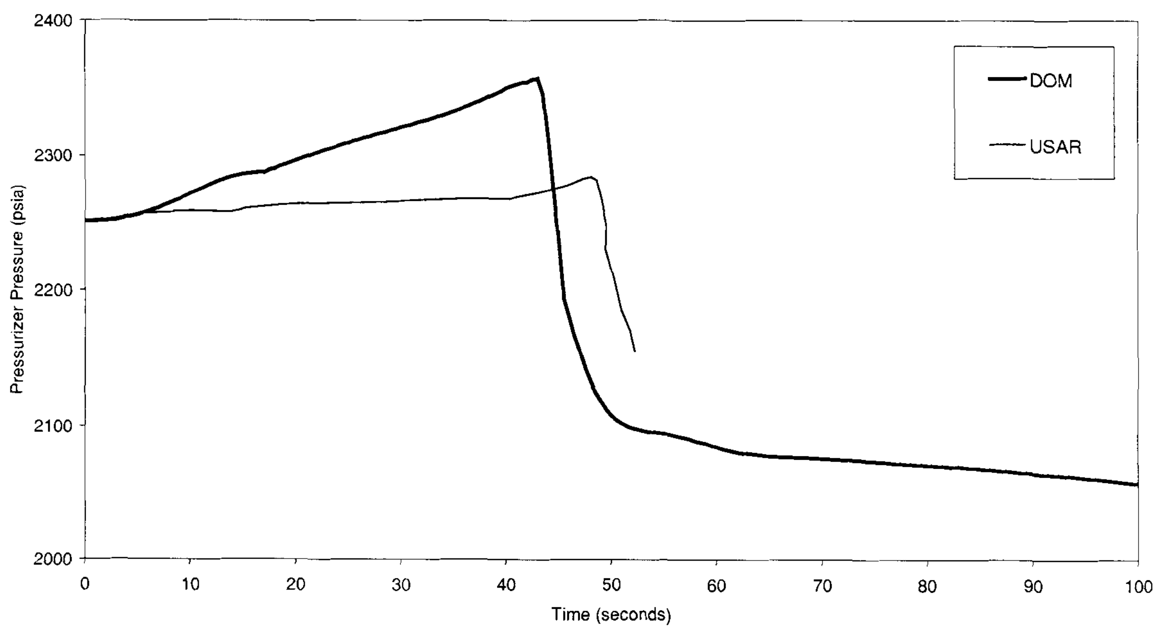


Figure 4.5-3 RWAP 3 pcm/sec - RCS Average Temperature

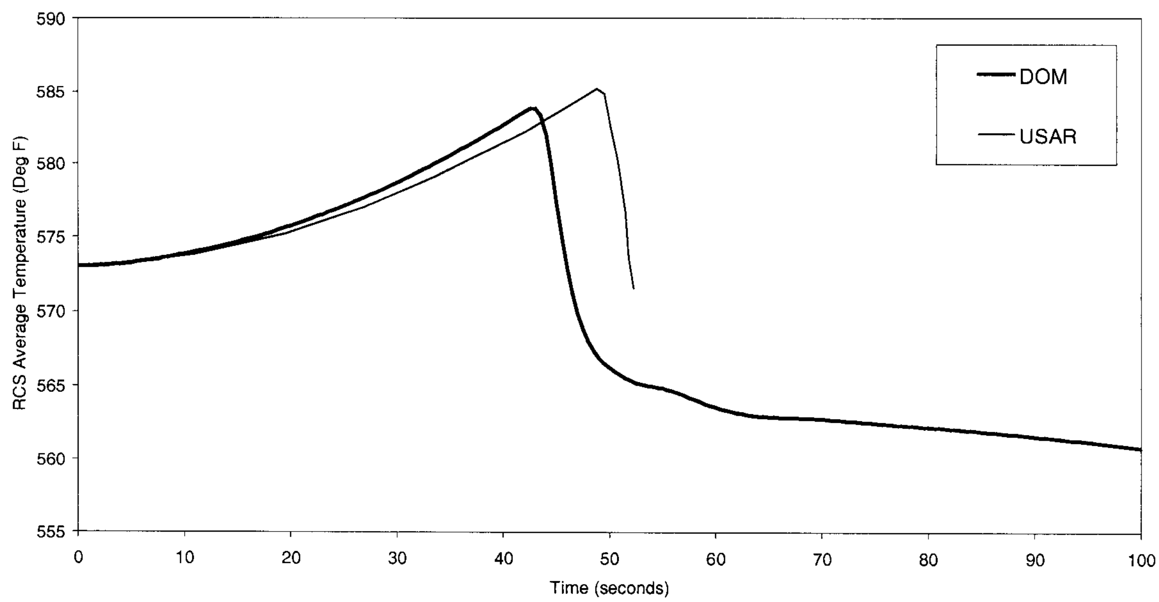


Figure 4.5-4 RWAP 100 pcm/sec - Core Power

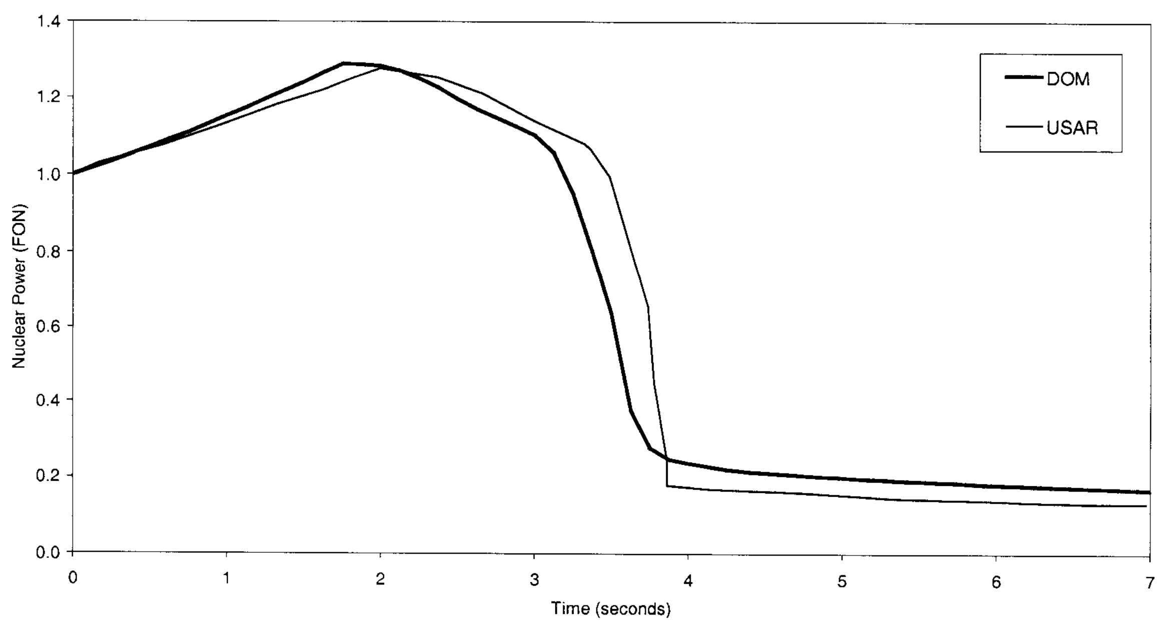


Figure 4.5-5 RWAP 100 pcm/sec - Pressurizer Pressure

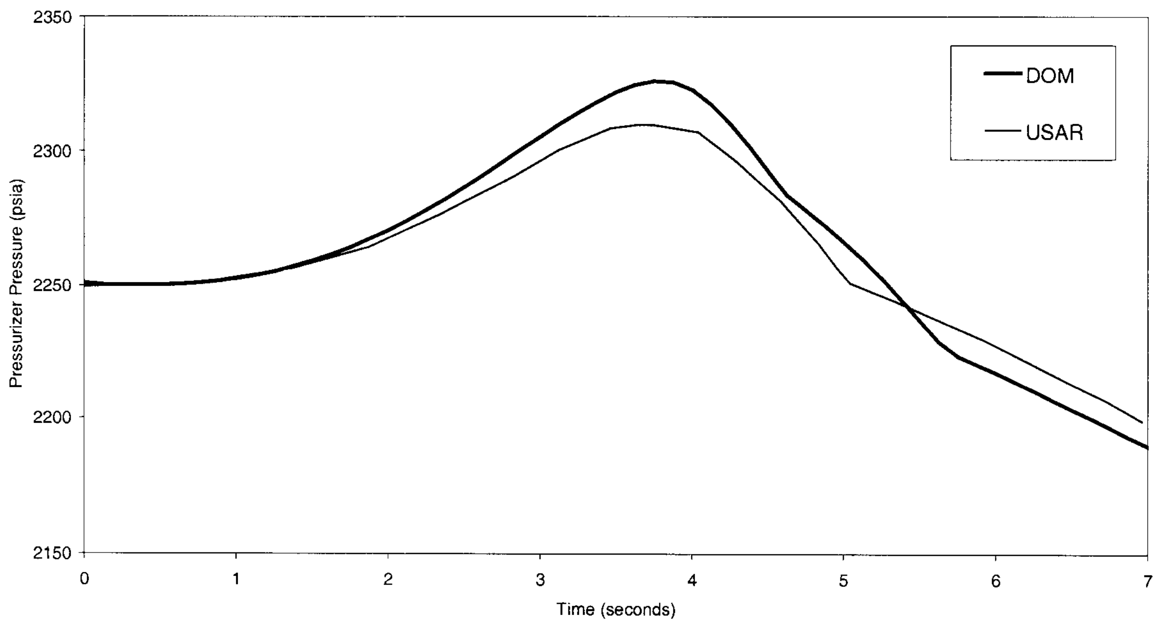
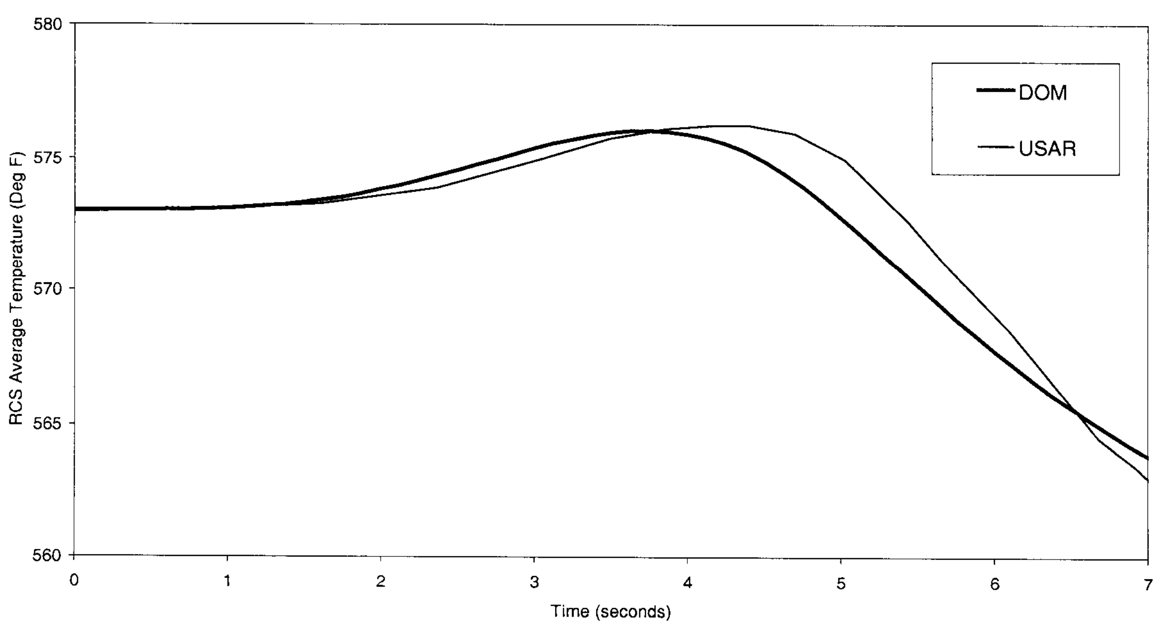


Figure 4.5-6 RWAP 100 pcm/sec - RCS Average Temperature



Summary - RWAP

The Dominion Kewaunee analysis provides results that are similar to the USAR analysis for the RWAP event. There are small differences in core power response during the early portion of the transient. For lower reactivity insertion rates, the fuel pin heat transfer modeling differences can affect the time to reactor trip; however, the peak core powers and peak average coolant temperatures are in close agreement. The USAR spray flow rate appears to include additional conservatism, resulting in a less severe pressurizer pressure response compared to the Dominion analysis.

4.6 Loss of Flow

The Loss of Flow (LOF) event is the loss of one or two reactor coolant pumps (RCP) and an associated coastdown of reactor flow. The reduction in core flow results in a sudden increase in coolant temperature and increased probability of violating a DNBR limit.

The LOF transient scenario presented here is a complete loss of flow resulting from the trip of both RCPs. The input summary is provided in Table 4.6-1. Where differences from USAR inputs exist, they are indicated in the Notes column.

Table 4.6-1 LOF Input Summary

Parameter	Value	Notes
Initial Conditions		
NSSS Power (MW)	1780	Nominal plus pump heat
RCS Flow (gpm/loop)	93,000	Minimum measured flow
Vessel T_{AVG} (F)	573	Nominal
RCS Pressure (psia)	2250	Nominal
Pressurizer Level (%)	48	Nominal
SG Level (%)	44	Nominal
SG Pressure (psia)	804.5	
Assumptions/Configuration		
Pump inertia (lbm-ft2)	72,000	
Reactor trip	-	Low RCS flow is active
Automatic rod control	-	Not credited
Pressurizer sprays, PORVs	-	Not credited
Main steam dumps, SG PORV	-	Not credited
SG tube plugging (%)	10	Max value
Reactivity Parameters		
Doppler Temp. Coef (pcm/F)	-1.2	USAR uses most negative Doppler-only power coefficient, and a least negative DTC (driven by moderator temp).
Moderator Temp. Coef	0	

Results - LOF

RCS flow coasts down following LOF as shown on Figure 4.6-1. The Dominion results compare well with the USAR data demonstrating good agreement between the pump model and friction losses. Since there is minimal reactivity feedback, the core power remains nearly constant prior to the reactor trip on low RCS flow, and the Dominion and USAR power response compares well as seen on Figure 4.6-2. The resulting core average heat flux is provided on Figure 4.6-3. The Dominion case provides a higher, more conservative value than the USAR case. This is primarily due to the fact that the Dominion model is initialized at a higher fuel temperature with higher stored energy. The USAR assumes a higher gap conductance and UO_2 thermal conductivity resulting in lower initial fuel temperature and stored energy. A sensitivity case is also shown on Figure 4.6-3 that modifies the inputs to more closely match the USAR case initial conditions and assumptions, including the higher gap heat transfer conductance and UO_2 thermal conductivity multiplier. In this case, the heat flux closely tracks the USAR value.

The RCS loop temperature response is shown on Figure 4.6-4. Again, the higher stored energy results in higher loop temperatures for the Dominion case compared to the USAR response. This is also reflected in the pressurizer pressure response shown on Figure 4.6-5 where the RCS pressure for the Dominion case peaks above the USAR case and remains higher for the duration of the event. The effect of pressurizer liquid flashing after the pressure peak can be seen by a reduction in the rate of pressure decrease.

Table 4.6-2 LOF Results

Parameter	**DOM**	**USAR**
Sequence of Events:		
Reactor Trip (sec) (Low RCS flow)	2.61	2.57

Summary - LOF

The Dominion Kewaunee LOF analysis provides RCS flow response that is very similar to the USAR results, demonstrating close agreement between the pump model and friction losses. The timing for reactor trip and the core power response is also in close agreement. The core heat flux is higher for the Dominion case due to differences in initial fuel temperature and stored energy. This is also reflected in higher primary side pressure and temperature values. The higher heat flux is conservative for DNBR acceptance criteria.

Figure 4.6-1 LOF - Total Core Inlet Flow

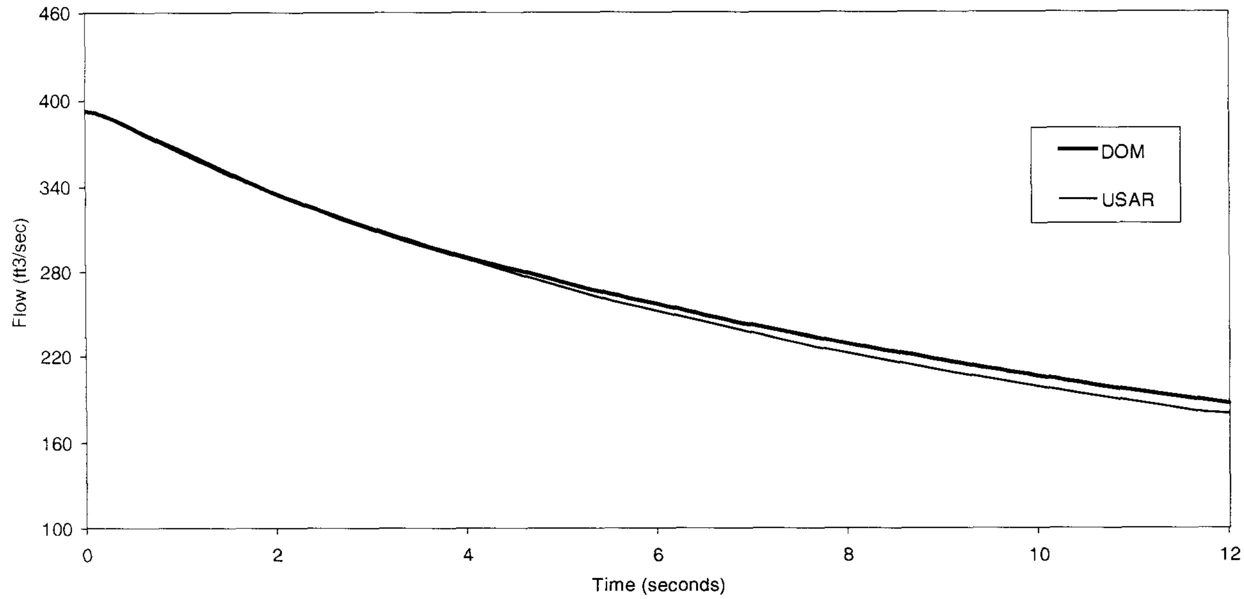


Figure 4.6-2 LOF - Nuclear Power

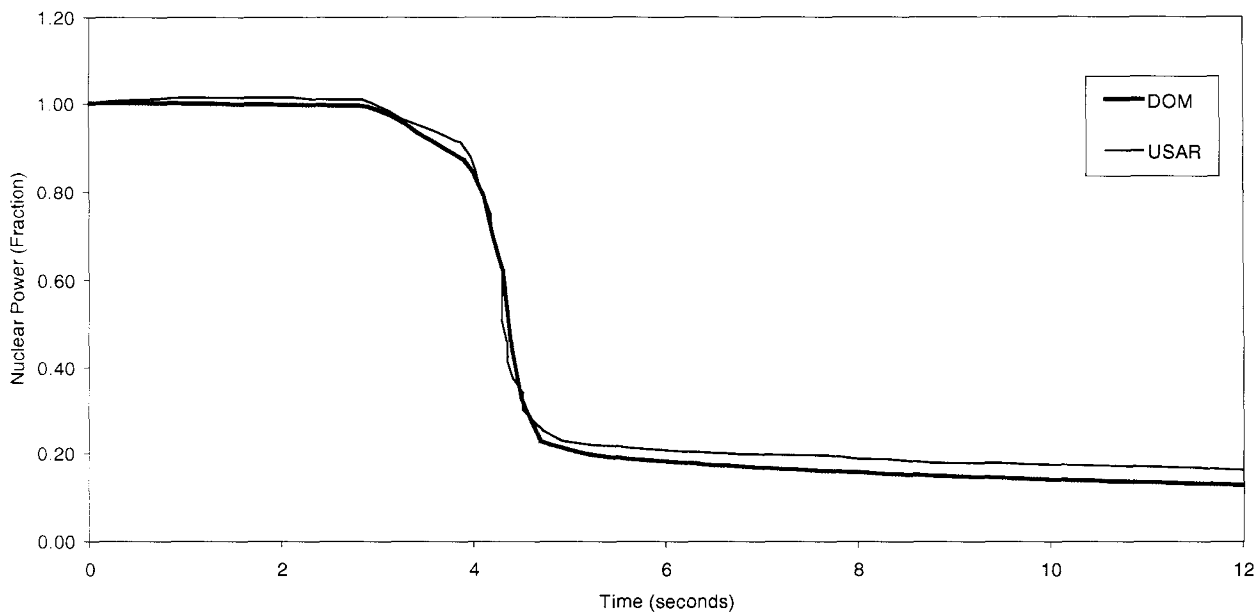


Figure 4.6-3 LOF - Core Average Heat Flux

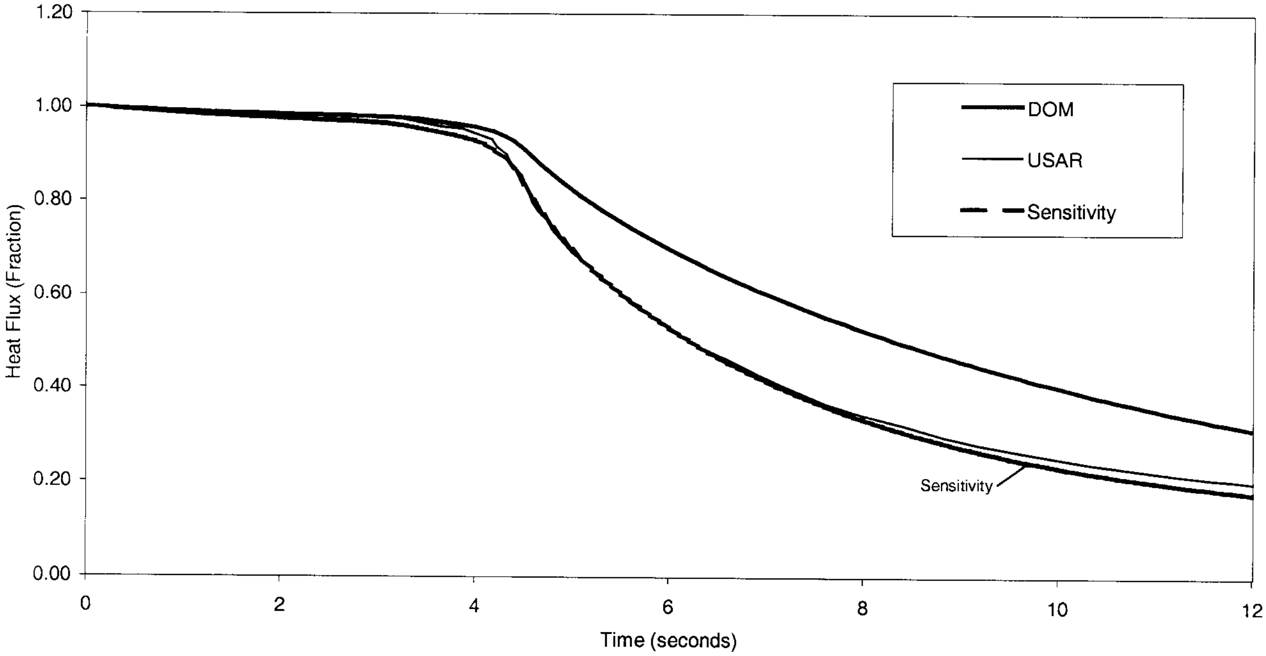


Figure 4.6-4 LOF - RCS Loop Temperature

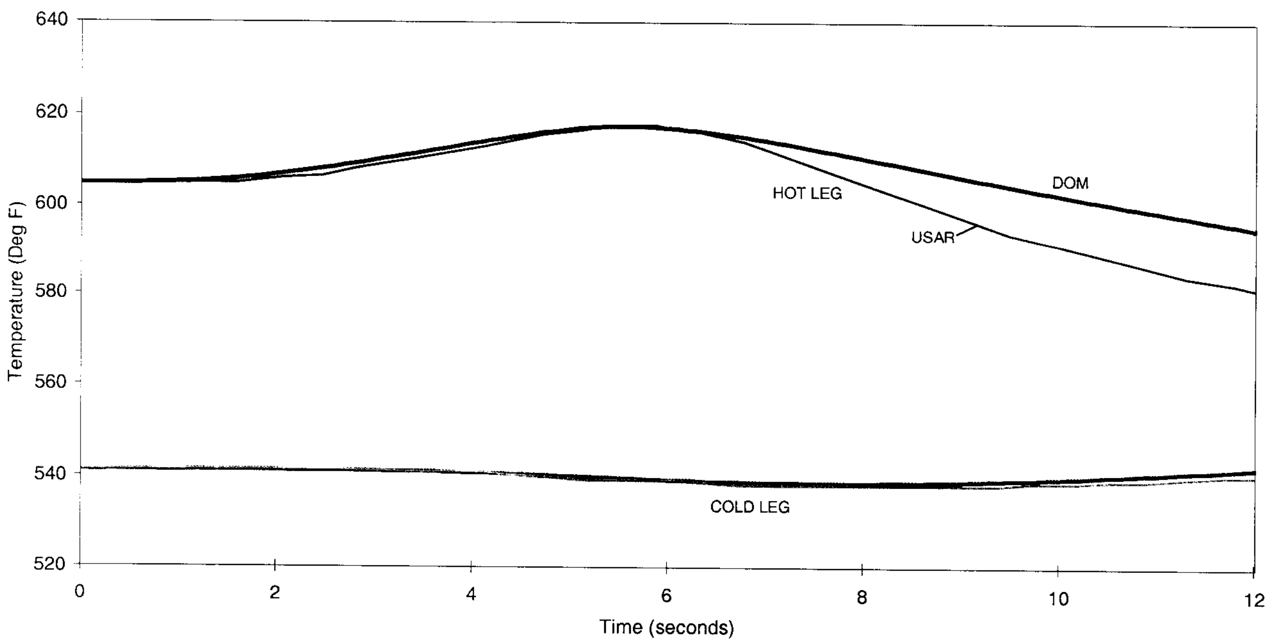
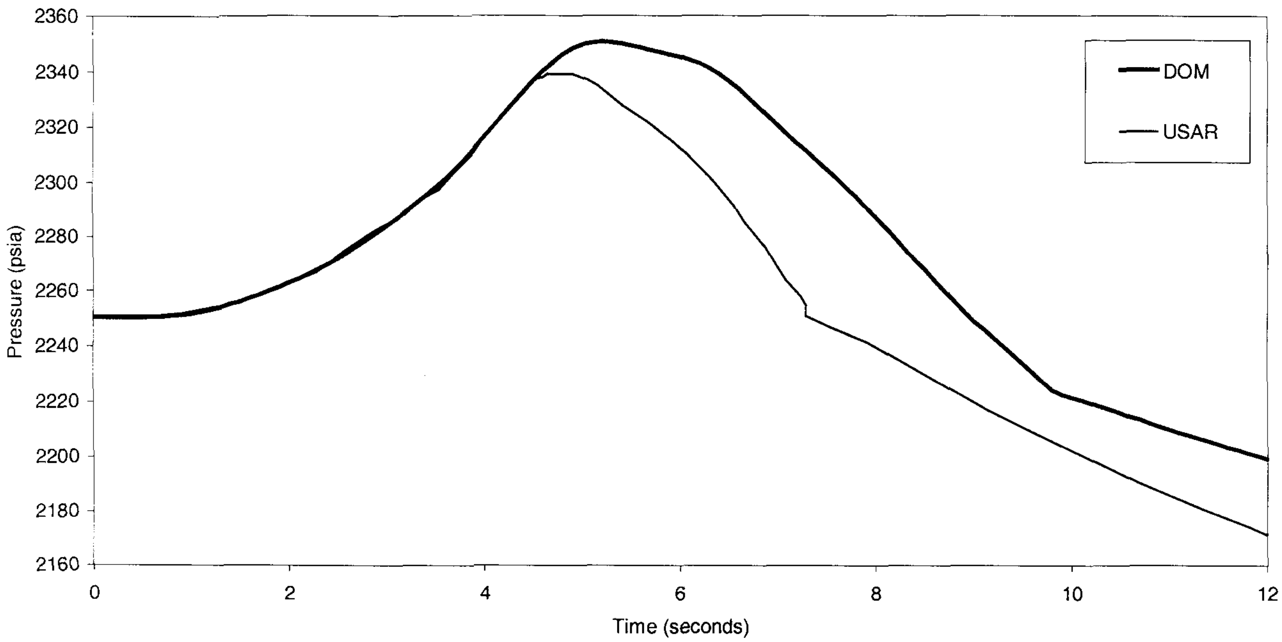


Figure 4.6-5 LOF - Pressurizer Pressure



5.0 Conclusions

This report presents demonstration transient analyses performed with the KPS RETRAN model developed in accordance with VEP-FRD-41. These analysis results are compared with current Kewaunee USAR results. The following conclusions are drawn based on these analyses.

1) This report demonstrates that the Dominion RETRAN-02 model and analysis methods can predict the response of transient events with results that compare well to USAR results.

2) Where there are differences between the Dominion results and the USAR results, they are understood based on differences in noding, inputs, or other modeling assumptions.

3) The Dominion Kewaunee RETRAN-02 model is consistent with current Dominion methods (Reference 1). These methods have been applied extensively for Surry and North Anna licensing, engineering and plant support analyses.

4) The RETRAN comparison analyses satisfy the DOM-NAF-5 applicability assessment criteria and provide further validation of the conclusion that Dominion's RETRAN analysis methods are applicable to Kewaunee and can be applied to Kewaunee licensing analysis for reload core design and safety analysis.

6.0 References

1) Topical Report, VEP-FRD-41, Rev. 0.1-A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code," June 2004.

2) Topical Report, VEP-NFE-2-A, "VEPCO Evaluation of the Control Rod Ejection Transient," December 1984.